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Application of the Concept of Clearance

Draft Safety Guide
DS500 (Revision of Safety Guide RS-G-1.7)

IAEA
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FOREWORD

By
Director General

[standard text to be added]

PREFACE

In 2014, the Agency published the safety requirements *Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards* (IAEA Safety Standards Series No. GSR Part 3) (the BSS), jointly sponsored by EURATOM, FAO, IAEA, ILO, OECD/NEA, PAHO, UNEP and WHO. That publication sets out the requirements that are designed to meet the fundamental safety objective and to apply the principles specified in the Fundamental Safety Principles (IAEA Safety Standards Series No. SF-1).

The establishment of safety requirements and guidance on the application of the concepts of exemption and clearance is a major component of the support for radiation protection and safety provided by the IAEA to its Member States. The objective of this Safety Guide is to promote an internationally harmonized approach to clearance, through detailed guidance on clearance levels and their application in practices, which is aimed to contribute to optimizing protection and safety and to application of the graded approach to regulation of materials, waste and objects containing radionuclides with low activity concentrations.

This Safety Guide updates part of the guidance related to clearance, that was provided in the previous safety guide: Application of the concept of Exclusion, Exemption and Clearance (IAEA Safety Standards Series No. RS-G-1.7), which is hereby superseded along with a parallel safety guide (DS499) that updates part of the guidance relevant to the concept of exemption.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Requirement 8 of IAEA Safety Standards Series No. GSR Part 3, *Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards* [1] makes provision for exemption and for the clearance of sources within notified or authorized practices, consistent with the use of a graded approach.

1.2. Exclusion, exemption and clearance are used as part of the process to determine the nature and extent of regulatory control as it applies to planned exposure situations. Exclusion applies to those planned exposures that are not deemed amenable to control, regardless of the magnitude of the exposures in question. Exemption refers to the determination by a regulatory body that a source or practice need not be subject to some or all aspects of regulatory control. Clearance is the removal of radiological regulatory control from radioactive material or radioactive objects within notified or authorized practices.

1.3. The process of clearance is a regulated activity and is carried out in accordance with the regulatory regime for the authorized activity and, hence, the procedures and processes leading to the act of clearance need to be well defined.

1.4. This Safety Guide provides guidance on the clearance process and on the application of the clearance levels, in particular on the organisation and regulation of the process, and its verification. It contains guidance on the clearance process; establishment of national regulations; planning, organization and implementation; technical and safety implications; and resources needed to implement the clearance process. It also addresses conditional clearance, the use of surface contamination levels, and the concept of clearance for liquids and gases, explaining the boundary between clearance and discharges. It also discusses whether the existing clearance levels for solid material could be relevant to liquids and gases. Finally, involvement of interested parties and enhancement of public understanding is addressed.

1.5. This Safety Guide is one of the documents supporting the GSR Part 3 *Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards* and *GSR Part 6 Decommissioning of Facilities*. This safety guide addresses application of the concept of clearance. The Safety Guide on the Application of the Concept of Exemption (DS499) [2] addresses application of the concept of exemption.

Together, these two Safety Guides supersede the Safety Guide on Application of the Concepts of Exclusion, Exemption and Clearance, issued in 2004.[1]

1.6. GSR Part 3 [1] provides mass specific values that can be used for clearance of bulk quantities of solid material. Values are provided for both radionuclides of natural origin and artificial radionuclides and these are the values originally provided in RS-G-1.7. The models used in the calculations of individual dose for artificial radionuclides are described in the Safety Report SRS-44 [3].

1.7. The calculation scenarios and models described in the Safety Report SRS-44 [3] are still valid and therefore there is no need to repeat this information in this guidance document.

1.8. It is recognized that the values for clearance, currently defined for artificial radionuclides, are based on exposure scenarios that are highly conservative. This Safety Guide provides guidance on how to avoid additional layers of conservativism in the other steps of the clearance process. It also reflects the use of the graded approach, in the light of the conservative nature of the values.

OBJECTIVE

1.9. The objective of the Safety Guide is to provide detailed guidance on the application of the concept of clearance for materials and buildings that are to be released from regulatory control in the framework of planned exposure situations, as specified in GSR Part 3 [1]. That guidance addresses regulatory framework for clearance, clearance process, derivation of clearance levels, application of clearance to solid, liquid and gaseous materials, unconditional (general) clearance and conditional (specific) clearance for both mass specific and surface specific clearance criteria. It also provides guidance on involvement of interested parties.

1.10. The This safety Guide is mainly intended for regulatory bodies in Member States to assist them in the application of the GSR Part 3 requirements on the clearance of materials and objects from regulatory control. It will also be of interest to authorized parties.

SCOPE

1.11. The scope of this Safety Guide covers the following aspects:

- Responsibilities of the authorized party (registrant or licensee) and the regulatory body;

[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).

- All relevant steps of the clearance process including characterization, determination of the nuclide vector, sampling, measurement techniques, and management of the clearance process;
- Mass specific and surface specific clearance criteria for unconditional clearance;
- Concept of conditional clearance and guidance on its application;
- Examples of derivation of mass specific and surface specific clearance criteria for conditional clearance (actual values would depend on specific conditions applied, so no universal set of values could be proposed);
- Case by case approach which can be used for small quantities of material, or for other situations where the assumptions for the generic derivation of clearance levels do not apply (e.g. where the water pathway is not relevant), or for radionuclides for which clearance values have not been given in the GSR Part 3 [1], or e.g. for cases where it is proposed that the rounding procedure or other features from the model in Safety Report SRS-44 [3] are not applied or are modified;
- Clearance of material and waste associated with planned activities in an area affected by consequences of a nuclear or radiological emergency;
- Considerations of clearance of liquids;
- Consideration of clearance of gases;
- Additional requirements for building materials containing radionuclides of natural origin;
- Considerations of averaging masses and averaging areas;
- Discussion of the scenarios underpinning calculation of the clearance levels and the implications for application of the clearance levels;
- Involvement of interested parties.

1.12. The guidance provided in this Safety Guide is especially applicable during decommissioning of facilities, to assist in the minimization of waste that will require disposal as radioactive waste. However, the guidance is also applicable for removal of regulatory control by the regulatory body from radioactive material or radioactive objects within other notified or authorized facilities and activities, such as releasing material for unconditional reuse/recycling or for non-radiological disposal during the normal operation of a facility.

1.13. The information presented in this Safety Guide is applicable to facilities that use, manufacture, process or store radioactive material. The types of facilities considered include nuclear power plants, research reactors, other nuclear fuel cycle facilities, industrial plants, medical facilities, research facilities

and accelerators. It also applies to industries processing material containing radionuclides of natural origin (NORM industries). NORM industries are industries where these materials are processed but not for their radioactive, fertile or fissile properties. Examples of NORM industries are production of oil and gas, manufacture of titanium dioxide pigments, extraction of rare earth elements and alloys, production of metals (aluminium, iron, steel), use of thorium in gas mantles.

1.14. The aspects of exemption are outside the scope of this Safety Guide, as they are addressed in the Safety Guide DS499 [2]. The concept of exclusion is described in the section 2, as well as in DS499.

1.15. The aspects related to the control of contaminated non-food commodities that can be traded freely are outside the scope of this Safety Guide and will be addressed in a separate publication.

1.16. The aspects related to release of sites from regulatory control are outside the scope of this Safety Guide, as they are addressed in the Safety Guide WS-G-5.1 [4].

1.17. The aspects related to managing radioactive waste in an emergency are outside the scopes of this Safety Guide, as they are addressed in the Safety Guide GSR Part 7 [5] and GSG-11 [6].

STRUCTURE

1.18. Section 2 gives an overview of the regulatory framework for clearance, including basic definitions and concepts of exclusion, exemption and clearance, general clearance criteria, and the responsibilities of different parties involved. Section 3 addresses the general aspects of clearance, such as the overall process and its management. Section 4 deals with clearance of solid material, discussing mass specific and surface specific criteria for clearance, case-by-case approach, averaging masses and areas, implementation of clearance measurements, uncertainty considerations and aspects related to use of mixing and dilution. Sections 5 and 6 provide considerations related to clearance of liquid and gaseous materials, respectively. Concept of conditional clearance is discussed in Section 7. Section 8 addresses the involvement of interested parties and the enhancement of public understanding in relation to clearance.

1.19. The Appendix provides an example of application of screening levels for recycling or disposal on conventional landfills of material and waste generated in a post-emergency situation.

1.20. Annex I discusses the dosimetric modelling for derivation of radionuclide specific values for clearance based on surface contamination measurements. Annex II provides examples of surface specific values for unconditional clearance. Annex III provides examples of mass specific values for conditional clearance. Annex IV provides an example of the application of clearance in small medical facilities.

2. REGULATORY FRAMEWORK FOR CLEARANCE

GENERAL

2.1. In this Safety Guide term "clearance" is used in accordance with the definition from the IAEA Safety Glossary, 2018 Edition [7]: "*Removal of regulatory control by the regulatory body from radioactive material or radioactive objects within notified or authorized facilities and activities. Removal from regulatory control in this context refers to regulatory control applied for radiation protection purposes.*". The term "clearance" is used in relation to materials, waste and movable objects, while for removal of regulatory control from buildings and sites, the term "release from regulatory control" is used. Usually, there are different radiological criteria for clearance of materials and for release of sites, but in some Member States the same criterion is used for both materials and sites, so the term "clearance of buildings and sites" is also in use.

2.2. The Safety Guide No. 89 (1988) [8] described the original basis for exemption and the derivation of the 10 µSv/a dose criterion and the collective dose criterion of 1 man Sv per year of operation. This concept was taken forward in the Basic Safety Standards 115 (1996) [9] as a basis for exemption and clearance.

2.3. In 2014, the International BSS (GSR Part 3 [1]) introduced clearance levels based on criteria for individual exposure. The collective dose criterion is no longer considered as part of the clearance and exemption concepts.

2.4. GSR Part 3 [1] defines the concept of clearance as "*The removal of regulatory control by the regulatory body from radioactive material or radioactive objects within notified or authorized practices*", where a practice refers to "*Any human activity that introduces additional sources of exposure or additional exposure pathways, or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people or the number of people exposed.*"

2.5. The requirement 10 (Justification of practices) of GSR Part 3 states that: "The government or the regulatory body shall ensure that only justified practices are authorized". So, it is implicit that a practice has to be justified[2] in order to be subject to notification or authorisation.

[2] Definition of **justification** [1]: The process of determining for a planned exposure situation whether a practice is, overall, beneficial; i.e. whether the expected benefits to individuals and to society from introducing or continuing the practice outweigh the harm (including radiation detriment) resulting from the practice.

2.6. In addition, GSR Part 3 defines a clearance level as "*A value, established by a regulatory body and expressed in terms of activity concentration, at or below which regulatory control may be removed from a source of radiation within a notified or authorized practice.*" Therefore, the process of clearance has to be a regulated process, established by the regulatory body.

2.7. The clearance criteria are described in Schedule I, Paras I.10 to I.12 of the GSR Part 3 [1], stating as a general rule 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."

Radioactive material may then be cleared without further consideration provided that the expected effective dose incurred by any individual is of the order of 10 μSv or less in a year for a realistic scenario and does not exceed 1 mSv in a year for low probability scenarios. The corresponding clearance levels (derived limits of activity concentration) for solid material with radionuclides of artificial origin are listed in Table I.2 of [1] and for solid material with radionuclides of natural origin in Table I.3 of [1].

2.8. For radionuclides of natural origin in residues that might be recycled into construction materials, or for which the disposal is liable to cause contamination of drinking water supplies, the activity concentration in the residues should not exceed specific values derived to meet a dose criterion of the order of 1 mSv in a year [1]. This means that the values in Table I.3 of [1] are not relevant in these cases. The regulatory body will therefore need to stipulate appropriate values taking into account these considerations. Further guidance is given in section 4. Examples for building materials are provided in the EC BSS (2013/59/EURATOM) [10] and EC RP112 [11]. The drinking water exposure pathway is addressed in EC RP122 Part 2 [11], and therefore these numbers can be used in situations where disposal of material could cause contamination of drinking water supplies.

2.9. Conditional clearance may be granted by the regulatory body for specific situations (para I.13 of the [1]), on the basis of the dose criteria listed in para 2.8. Further details are developed in section 7 of this Safety Guide.

2.10. Clearance is, in principle, applicable to solid, liquid and gaseous materials. Once the clearance process has taken place, the waste or material that meets the clearance levels is no longer considered radioactive material and can be used, recycled or disposed of without further regulatory consideration regarding the radiological aspects. Hence, the procedures and processes leading to the act of clearance

should be well defined. In particular, the respective responsibilities of the regulatory body and of the authorized party should be clearly established.

2.11. According to para 3.12 of GSR Part 3 [1] (Requirement 8: Exemption and clearance)

> *"The regulatory body shall approve which sources, including materials and objects, within notified practices 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."*

This means clearance regulations should be embedded into a regulatory framework specifying that cleared materials are no longer radioactive in a legal sense or, equivalently, that the residual activity of cleared materials may be disregarded. In this way, any discrepancy between the fact that cleared material may still bear activity in a physical sense, and the fact that this activity does not play any role in a legal sense can be avoided. This applies to both the materials released under either unconditional clearance or conditional clearance.

2.12. As clearance is applicable in planned exposure situations, it is also applicable to management of material originating from remediation activities or from post-emergency situations. If the concept of clearance is applied to material arising from such situations, the dose criterion remains the same as for application of clearance to material from planned exposure situations, as specified in paragraphs I.10 – I.12 of GSR Part 3 [1]. Other approaches can also be taken, based on reference levels for existing exposure situations. More detailed explanation and examples are given in the Appendix.

EXCLUSION

2.13. According to paragraph 1.42 of the GSR part 3 [1], the requirements of the GSR Part 3 apply to all situations involving radiation exposure that are amenable to control. Exposures deemed not to be amenable to control are excluded from the scope of the GSR PART 3 and thereby from the scope of an instrument of regulatory control from a radiological point-of-view.

2.14. For example, it is not feasible or practical to control ^{40}K in the human body or cosmic radiation at the surface of the Earth (Footnote 8 of the GSR Part 3 [1]). Other examples of excluded exposures are: (a) unmodified soil concentrations (concentrations of radionuclides of natural origin in normal soil material), including unmodified high natural background radiation areas including unmodified soil concentrations in high natural background radiation areas, and any other unmodified primordial

radionuclides present in nature at (extremely) low activity concentration levels (e.g. ^{87}Rb, ^{138}La, ^{147}Sm, ^{176}Lu), and (b) global fallout coming from weapon testing (pre-1960s) and from nuclear accidents.

2.15. Excluded exposures are such that control measures are not possible to be taken by means of regulatory action, regardless of their magnitude. Therefore, sources leading to such exposures are, by their nature, excluded from regulatory control and are out of the scope of the requirements of the GSR PART 3 [1].

RESPONSIBILITIES OF THE REGULATORY BODY

2.16. To meet the Requirement 8 of the GSR PART 3 [1], described in para 2.12 of this Safety Guide, the regulatory body should put in place a framework for clearance of material, including clearance levels to be used, in agreement with the clearance criteria defined in the GSR PART 3 [1].

2.17. As part of the clearance process, the regulatory body should review the results of the characterisation programme implemented by the authorized party (described in para 2.35) to define the radionuclide inventory subject to clearance.

2.18. For clearance of bulk material from regulatory control, the regulatory body should refer to the derived clearance levels for solid material with radionuclides of artificial origin and of natural origin, listed in the Table I.2 and Table I.3 of [1] since they are based on the clearance criteria defined in [1] as discussed in para 2.8.

2.19. For clearance of surface contaminated material, the regulatory body should promote the use of radionuclide specific clearance levels, derived in an analogous way as the clearance levels for bulk materials. This topic is discussed further in section 4, paras 4.16-4.26.

2.20. When establishing clearance levels, the regulatory body should be aware of other regulatory requirements that could also apply, such as environmental limits and, to the extent possible, harmonize these requirements.

2.21. If the regulatory body allows authorized parties to propose their own derived values on the basis of the clearance criteria defined in [1], the regulatory body should require authorized parties to demonstrate that their own derived values will provide an equivalent level of protection and safety, and should approve these values. In that case, the implications of the selected values should also be explained to the relevant interested parties and verified against other regulatory requirements e.g. dealing with environmental limits.

2.22. In addition to defining clearance levels in terms of activity concentration, the regulatory body should also specify averaging masses, volumes or areas of material to be monitored for clearance, along

with acceptable confidence levels for false negatives. The regulatory body should also approve or specify additional monitoring criteria to identify presence of "hot spots" (a non-uniform distribution of activity concentration with some values above the clearance levels) in the material considered for clearance. Further details are discussed in section on averaging masses and areas (section 4, paragraphs 4.34-4.47).

2.23. The regulatory body should specify that deliberate dilution and/or mixing with clean material to reach the activity concentration values prior to release of material from regulatory control is not an acceptable practice, unless a permission is obtained from the regulatory body for such an action. More detailed explanation on this point is provided in section 4 (paragraphs 4.90-4.95).

2.24. According to GSR-Part3 [1], Requirement 14: Monitoring for verification of compliance:

> *(3.37) 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.*

2.25. According to the national framework, the regulatory body should agree / approve the monitoring process of the authorized party (registrant and licensee) for verification of compliance with clearance levels. Based on the results of the monitoring process, the authorized party should decide whether material complies with the clearance levels. For the decision as to whether specific material is suitable for clearance, the regulatory body should base its approval on the monitoring results, according to the national framework. In case of the use of statistically based methods by the authorized party, the approach should be fully documented and approved by the regulatory body prior to its implementation.

2.26. Since decisions made on the basis of the monitoring results have important regulatory, public health and societal implications, the quality management system implemented and used by the authorized party should satisfy the requirements established by the regulatory body and international standards. The regulatory body could also undertake its own independent verification programme, as additional assurance that the monitoring programme is being carried out adequately.

2.27. In the case of conditional clearance (specific clearance), the regulatory body should establish a mechanism to demonstrate compliance with the conditions attached to the process, e.g. that the metal will only go to a recycling facility and will be melted rather than reused directly.

2.28. According to GSR-Part3 [1], Requirement 3: Responsibilities of the regulatory body,

> *"2.35. The regulatory body shall make provision for establishing, maintaining and retrieving adequate records relating to facilities and activities."*

2.29. In the case of clearance of material from regulatory control, the regulatory body should define the format for the records and the documentation of the to demonstrate compliance of the monitoring process with the requirements.

2.30. In addition, the regulatory body should define the period of time required for keeping the records and documentation (depending on the history, hazard and characteristics of the material) after the material has been released.

2.31. Material cleared from radiological regulatory control could still be subject to other non-radiological regulatory controls. Therefore, the relevant regulatory bodies should coordinate their activities, share their concerns, and communicate their regulatory strategies and their implementation in order to build confidence in the clearance process and to ensure smooth management of the material after clearance. In the case of transport across the national borders, this coordination should involve regulatory bodies from the involved countries. This can be accomplished through transparency, disclosure and use of international standards and procedures.

2.32. Regulatory body should get involved in consultations with interested parties in the development of the regulatory framework for clearance and in enhancing public understanding.

RESPONSIBILITIES OF THE AUTHORIZED PARTY

2.33. According to GSR Part 3, Requirement 14: Monitoring for verification of compliance:

(3.38). 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 these Standards;

(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 these Standards;

(e) The results of monitoring and verification of compliance are shared with the regulatory body as required.

2.34. As part of the clearance process, the authorized party should perform a radiological characterization of the material to be cleared, comprising the determination of the radionuclide vector to be considered

and the spatial distribution of the radiological activity. The results should be submitted to the regulatory body according to the national framework.

2.35. The authorized party should set up the clearance process, making the measurements and verifying compliance with the clearance criteria, including selecting proper equipment and place for clearance measurements, calibration of equipment, training of staff, development of procedures and documentation, and interfacing with the regulator, according to the national framework.

2.36. The process of clearance of material from regulatory control should be an integral part of the integrated management system . The assurance of the quality of results obtained and used during the clearance process is critical for ensuring and demonstrating that the established activity values have been met and to build confidence in the use of the data, the equipment and the methodology. The authorized party should design and implement a quality management programme during monitoring for compliance with clearance levels through formally documented and controlled procedures and working instructions. This quality management should satisfy the recognized standards established by the regulatory body and international standards.

2.37. The authorized party is responsible for the reliability of the results of the monitoring programme and should not rely on the regulatory body to point out unexpected deficiencies in their work. Any verification programme carried out by the regulatory body should not be considered as a substitute for the quality control/assurance programme established by the authorized party.

2.38. The authorized party should communicate the results of its clearance and monitoring programme to the regulatory body in a transparent way to obtain regulatory approval for the clearance of material.

2.39. The authorized party is responsible for the clearance activity and should retain key records from the monitoring to demonstrate that clearance has been carried out appropriately. These records should be developed and preserved in the appropriate formats, as defined by the regulatory body. Documentation should be stored for a defined period of time, as specified by the regulatory body.

2.40. The authorized party should engage with interested parties to discuss the application of the concept of clearance.

ORGANIZATION AND IMPLEMENTATION OF THE CLEARANCE PROCESS

2.41. The organisation and implementation of the clearance process will be dependent on the chosen approach of the regulatory body for this matter. The minimal set of requirements for the clearance process should consist of:

- Defining roles and responsibilities of authorized party, contractors and regulatory bodies and foreseeing adequate resources (in number and competence);
- Establishing an appropriate quality management programme;
- Organizing involvement of interested parties prior to implementation of the process.

2.42. The clearance levels to be used could either be defined by the regulatory body (e.g. generic clearance levels as defined in the GSR Part 3 Tables I.2 and I.3 [1]), or proposed by the authorized party and approved by the regulatory body, or a combination of the two approaches.

2.43. In any case, the overall clearance process requires a structured approach both by the regulatory body and by the authorized party. The regulatory body should clearly define the different steps in the process and specify hold points if applicable. Arrangements should be put in place for timely discussions between regulator and authorized party as an important part of the clearance process. In the cases where authorisation or licensing for clearance is required, the requested data and the level of detail should be specified by the regulatory body.

2.44. In order to verify compliance with clearance levels, the authorized party should put in place an appropriate monitoring programme, based on a reliable characterisation and a good definition of the source term. The monitoring programme should be submitted to the regulatory body for approval, according to the national framework, before the start of the clearance process.

2.45. Clearance occurs at the point at which regulatory control due to radioactivity of the material is removed. This will occur at a location within an authorised facility and might involve independent verifications by the regulatory body. Additional considerations for the point at which clearance occurs in case of conditional clearance are addressed in section 7.

GRADED APPROACH

2.46. The clearance process provides an opportunity to apply a graded approach to management of material and waste, by applying the level of regulatory control that is commensurate to the level of radiological hazards and risks associated with the material and its intended use (Requirements 3 and 6 of the GSR Part 3 [1]).

Requirement 3: Responsibilities of the Regulatory Body

"2.31. 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."

Requirement 6: Graded approach

"The application of the requirements of these Standards 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."

2.47. The application and implementation of the clearance concept relates to a large range of authorized activities, for example, operation of facilities, decommissioning, and management of radioactive waste. International consensus has been achieved on activity concentration values (Bq/g) below which material does not require regulatory control [1]. The practical implementation of these clearance levels should also consider a graded approach.

2.48. The application of the graded approach to the clearance process should take into account aspects such as the size and complexity of the facility or project (e.g. nuclear power plant versus research laboratory, decommissioning versus operational activities), operational history, the national regulatory framework and general economic factors.

2.49. In the case where the history and provenance of the material is well known and shows evidence for no activation and little or no contamination, the complexity of the monitoring process (number of samples and measurements, type of analyses) should be in an agreement with this information. In any case, the reasoning for the selected approach needs to be documented.

2.50. Also, if the material has a uniform radionuclide vector or a uniform level of contamination then fewer measurements are required to characterise it and conduct the clearance process.

2.51. Regardless of the size of the project, adequate monitoring of the material to be released is required to demonstrate that the requirements of the regulatory body are met, but the level of effort put into quality management needs to be commensurate with the scope and complexity of the monitoring process.

2.52. Further discussion on the management of the uncertainties in the clearance process is given in section 4.

2.53. The concept of conditional clearance, otherwise referred to in GSR Part 3 as *"clearance granted by the regulatory body for specific situations"*, is also an example of a graded approach. This is described further in the section 7.

3. GENERAL ASPECTS OF CLEARANCE

CLEARANCE PROCESS

3.1. The operation and decommissioning of facilities results in the generation of radioactive waste or material. Some of this radioactive waste or material will have a sufficiently low activity concentration that meets the criteria for clearance as described in Paras I.10 to I.12 of the GSR Part 3 [1] and is therefore suitable for clearance. Most of the radioactive waste and material with low activity concentrations will be solid, but there are situations when liquids (and even gases) may also be suitable for clearance. The discussion on general aspects of clearance therefore focuses on aspects relevant to solids. If not specified otherwise, it is also applicable to liquids. Specific considerations for liquids are addressed in Section 5 and for gases are addressed in Section 6.

3.2. The clearance process results in a decision as to whether the radioactive waste or material can be released from further regulatory control regarding its radiological properties. Other properties, e.g. the hazardous properties of the wastes, will determine whether other controls remain or become appropriate.

3.3. Clearance is undertaken in the authorized facility and therefore should comply with the regulatory regime applicable to the authorised practice.

3.4. As part of the clearance process, the radionuclide content of the material should be determined through a process of characterisation (more detailed guidance is provided in the following sub-section). The resulting list of the radionuclides present and the fraction, or percentage, of the activity concentration contributed by each radionuclide is known as the radionuclide vector. The different processes that have contributed to the presence of radionuclides in the waste (e.g. fission, activation, contamination, fuel fabrication) should be identified in order to ensure that the radionuclide vector is comprehensive.

MATERIALS CONTAINING MORE THAN ONE RADIONUCLIDE

3.5. GSR Part 3 [1] specifies clearance levels for use in the clearance of solid material. The mass specific clearance levels for individual radionuclides of artificial origin are listed in Table I.2 of [1] and the mass specific clearance levels for individual radionuclides of natural origin are listed in Table I.3 of [1].

3.6. The clearance levels specified in GSR Part 3 [1] apply to individual radionuclides. If wastes or material contain more than one radionuclide, the process of clearance should take into account the contributions of each of the radionuclides to the dose. Therefore, a clearance level should be determined for the specific radionuclide vector in the material that is being considered for clearance. The approach

for materials containing more than one radionuclide depends on whether the radionuclide is of artificial or of natural origin.

3.7. The approach for materials containing more than one radionuclide of artificial origin is often referred to as the 'sum of fractions' approach and the summation rule is described in para I.14 of [1]:

For clearance of radioactive material containing more than one radionuclide of artificial origin, on the basis of the levels given in Table I.2 (p. 124), 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 (Xm), determined as follows:

$$X_m = \frac{1}{\sum_{i=1}^{n} \frac{f(i)}{X(i)}}$$

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; and

n is the number of radionuclides present.

3.8. The mass specific clearance levels for radionuclides of artificial origin, given in Schedule I, Table I.2 of [1], also take into account dose contributions from relevant progeny radionuclides. The weighting factors applied for the activity concentrations of the progeny are given in Table II-1 of the Appendix II of the SRS-44 [3]. Parent radionuclides, and their progeny whose dose contributions are taken into account in the dose calculations in SRS-44 (thus requiring only the clearance level of the parent radionuclide to be considered), are listed in the footnote to Table I.2 in GSR Part 3 [1]. The weighting factors account for secular equilibrium and also for the relevant contribution of longer-lived progeny to the exposure resulting from the parent radionuclide.

3.9. Therefore, a progeny radionuclide that is listed in the footnote to Table I.2 in GSR Part 3 [1] should not be included in the summation rule if it is present at an activity concentration that is equal to or lower than that corresponding to the weighting factor from the Table II-1 of Appendix II of the SRS-44 multiplied by the activity concentration of the parent radionuclide. If this is not the case, the activity concentrations of parents and progeny radionuclides should be considered (unmodified) in the summation rule.

3.10. The mass specific clearance levels for radionuclides of natural origin apply to each individual radionuclide. Hence, the sum of fractions approach is not appropriate for clearance of materials containing

only radionuclides of natural origin. GSR Part 3 [1] specifies in para I.12(b) that the material can be cleared *provided that the activity concentrations of radionuclides of natural origin do not exceed the relevant level given in Table I.3.* The activity concentration of each radionuclide of natural origin should therefore be compared with the clearance level and, if each one is less than or equal to the clearance level then the material can be cleared. For example, for wastes containing radionuclides from the ^{238}U- decay chain, the mass specific clearance level of 1 Bq/g would apply to each member of the ^{238}U- decay chain present in the waste.

3.11. Radionuclides of natural origin that arise as a result of the practice are treated in the same way as artificial radionuclides. Hence, the mass specific clearance level of 1 Bq/g does not apply in these cases.

3.12. For clearance of solid bulk material containing a mixture of radionuclides of natural origin and radionuclides of artificial origin, GSR Part 3 [1] specifies that *the conditions given in paras I.12(b) and I.14 of [1] both have to be satisfied*. Therefore, the radionuclides of artificial origin in the waste or solid material should be considered separately from the radionuclides of natural origin when making the decision as to whether the waste can be cleared. The clearance decision should therefore contain the following steps:

a) apply the summation rule to the radionuclides of artificial origin in the waste or material;

b) apply the clearance levels for radionuclides of natural origin to the radionuclides of natural origin individually;

c) if both the radionuclides of artificial origin and the radionuclides of natural origin meet the clearance criteria then the material can be cleared.

3.13. Note that radionuclides of natural origin that arise as a result of a practice are included in the summation rule under (a), as discussed in paras 4.7-4.9. Radionuclides of natural origin not arising from a practice, e.g. present in construction materials, are considered under (b) above.

CHARACTERISATION OF THE MATERIAL TO BE CLEARED

3.14. The objective of radiological characterization of the material to be cleared is to provide a reliable database of information on quantity and type of radionuclides, their spatial distribution and their physical and chemical states. The database should then be used by the authorized party to clearly define the material to be cleared, and to select the optimum monitoring strategy (compliance monitoring) for the clearance process. The database should also be used to assess various options for the clearance process and their consequences, for example, batch monitoring tools and techniques, destinations for the cleared material, conditional clearance or unconditional clearance, radiological protection of workers, general public and

environment, and resulting costs. The level of detail should be proportional to the complexity of the situation in accordance with the graded approach.

3.15. Characterization requires a logical and systematic approach. A comprehensive characterization programme comprises the following steps: (a) review of historical information; (b) activation and decay calculations; (c) preparation of the sampling and analysis plan based on an appropriate statistical approach; (d) performance of in situ measurements, sampling and analyses; (e) review and evaluation of the data obtained; and (f) comparison of calculated results and measured data. As discussed below, it is an iterative process.

3.16. The characterisation process should collect information on the following aspects: origin of the material within the facility, location of the originating facility, type of originating facility, period of operation, operational history (including incidents and post-incident remediation) and radionuclides associated with operations; size, type and quantities (total and rate of production) of material; radionuclides present in the material; expected levels of contamination or activation of each type of material; type of contaminant (fixed or non-fixed surface contamination, bulk contamination); homogeneity of contamination (identification of hot spots on the surface or within the volume); other hazards associated with the material; time frame for the clearance process and clearance monitoring throughput required. Further information on characterization is provided in Safety Report 67 [12].

3.17. The characterisation process will generate a large amount of data in different formats (e.g. paper, digital) and therefore the authorized party should have a suitable records and data management system, which should be integrated with the overall information management system of the facility. Examples of such systems to support decommissioning are described in the references [13,14].

Historical information

3.18. Detailed information on the history of the material to be cleared should be collected as the first step in the characterisation process. This information should then be used to develop the other steps in the characterisation process. Information should be obtained from various sources such as: historical records; knowledge of the types of processes involving the material; experience gained elsewhere; public or institutional memory; recollections from workers.

3.19. This detailed history should include information on: the processes or activities during the operation of the facility; location of controlled and supervised areas; description of the facility and equipment; type and form of the radioactive material used during operations; whether the radioactive contaminants have been enclosed within specific areas; whether the material has been potentially activated by neutron exposure; whether the material has been contaminated as a consequence of an accident or spill; whether

the building or equipment has been refurbished or modified; whether the building, equipment and areas have been decontaminated; time at which contamination or activation of the material occurred; results of any past characterisation or monitoring of the material.

3.20. Establishing the historical information relevant to the material to be cleared should be straightforward for most of the facilities, such as nuclear power plants, but might be more complicated for nuclear research facilities where different types of activities, such as experiments, chemical processes and others, were carried out, or for novel nuclear power plants.

3.21. The detailed information on the history of the material to be cleared should be used to determine an initial estimate of the radionuclide vector for the material, and this initial estimate should be used to develop and implement steps b), c) and d) described in para. 3.15 (calculation, sampling plan, in situ measurements, sampling and analysis). Initial in situ measurements provide useful information that can be used to guide the sampling plan, e.g. by defining zones.

3.22. Steps e) and f) (review and evaluation of data, and comparison of calculated and measured results) should be carried out as early as possible and continue during sampling and analysis. 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 measurements made indicate that the sampling plan in use will not give the required information for planning. The historical information may also need to be revisited if additional radionuclides are identified in steps b), c) and d) and thus the characterisation process should be viewed as an iterative process. One of the important outputs from the characterisation process is a credible radionuclide vector or vectors for the materials to be cleared.

3.23. Two main types of in situ measurement are relevant for characterization of solids for clearance: measurements of the surface contamination (fixed or removable), and bulk activity measurements, generally based on gamma spectrometry or total gamma measurements, but also including alpha and beta measurements. In each case, particular attention should be paid to ensure that the methods of measurement take into account the geometry, the surface conditions and the nature and extent of the radioactive contaminants. It is unlikely that dose rate measurements unsupported by spectrometry will provide useful information for characterization for clearance. Further information on in situ measurement techniques is available [12,15,16].

3.24. The sampling and analysis programme should ensure that representative samples are taken from the material to be characterized. The sampling and analysis techniques determine the constituents and their radioactivity in selected locations. Further information on sampling and analysis techniques is available [12,15,16] (SRS-67, TRS389, MARSSIM). The sampling and analysis techniques should be selected and applied in accordance with the graded approach.

Establishing the nuclide vector selecting from all radionuclides that have been identified

3.25. Material for clearance usually contains 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 and the ratios (also called correlation factors) between the different radionuclides that are used in derivation of radionuclide vectors. Then, a limited number of thorough measurements can be used to determine whether difficult-to-measure (DTM) radionuclides are found to be roughly in a fixed ratio with easy-to-measure (ETM) radionuclides. If this is the case, then a stable radionuclide composition exists and the measured correlation factors could be used to estimate the activity of the DTM radionuclides, in the material to be cleared, from the measurements on ETM radionuclides. An example is the use of ^{60}Co to monitor the wide range of DTM radionuclides associated with activation products and corrosion products associated with the operation of reactors.

3.26. Correlation factors for DTM radionuclides should be used carefully and the stability of the radionuclide vector should be reviewed frequently. In some facilities, one set of correlation factors can apply 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. Radionuclide composition will also vary where the radioactivity is generated by neutron activation and the concentration of impurities in the material play a significant role (e.g. variations in the cobalt content of steel).

3.27. The selection of the significant radionuclides to be evaluated for clearance is a kind of screening process. As such, it is based on an initial estimate of the activity concentration (C) of the radionuclides in the material. Since considerations of the uncertainty in the activity concentration (C) are addressed in the final process for clearance through the compliance measurements, the uncertainty in the value of C should not be considered in the screening process.

3.28. In practice, all radiation monitoring equipment has a response which depends on radiation type, energy and geometry. The response of the equipment that will be used will have to be calculated for the mix of radionuclides and their relative proportions. This involves selection of key radionuclides to be measured based on their emission properties, the ease and efficiency of detection (particularly whether the required detection limit can be achieved) and considering their contribution to the summation rule. Although selection of radionuclides which contribute a higher fraction towards the clearance level (CL) of the material, i.e. a high value of C/CL, where C is the activity concentration of the radionuclide would be preferable, in many cases it will be necessary to select a radionuclide which contributes a lower fraction

because it is easier to measure. An example for the selection method proposed in Japan for solid materials consists of two steps (Fig. 3.1):

(a) In the first step, a key radionuclide is selected that gives relatively high values of C/CL among the radionuclides considered and is easily measurable. Then a measure of significance is introduced which is a relative ratio defined by $(C_j/CL_j)/(C/CL)_{key}$, where $(C/CL)_{key}$ means the ratio for the key radionuclide, and significant radionuclides are selected as they satisfy the inequality $(C_j/CL_j)/(C/CL)_{key} > 0.01$.

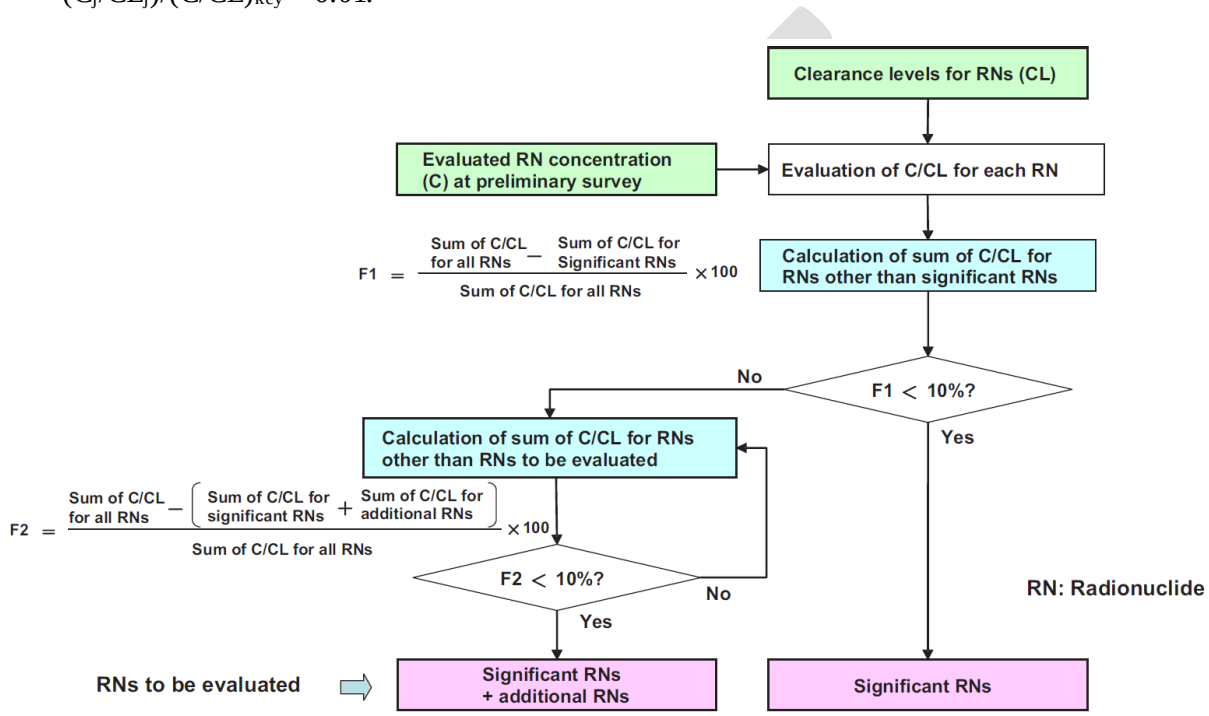

Fig. 3.1. An approach to selection of significant radionuclides to be evaluated.

(b) In the second step, the sum of C/CL for the all radionuclides considered and the sum of C/CL for significant radionuclides are calculated. Then, if the difference between the two sum values, F1, is less than 10%, the significant radionuclides are recognized as the ones to be evaluated. However, if not, the sum of C/CL for the significant radionuclides and for one additional radionuclide is calculated, and this process is repeated until the difference F2 is less than 10%.

3.29. The response of the monitoring system can then be calculated in terms of the radionuclide composition. This approach can also allow a calculation of the likely variation in response of contamination monitoring equipment with the surface contamination. If the equipment, for example, has a good response over a wide range of beta energies, then the response will change quite quickly 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, e.g. with cling film, so that the variations are eliminated.

3.30. As mentioned above, the radionuclide composition (and the correlation factors) should be re-evaluated as monitoring of material proceeds, particularly on old, complicated facilities that cannot be characterized in detail before the clearance process begins. Quite simple means can sometimes be employed to check on radionuclide composition stability, 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 cheap and easy process that can be employed to check on the photon emitting component. A combination of gamma spectrometry and gross beta measurement can also demonstrate stability where the main contaminants are 137Cs + 137mBa, a gamma and medium energy beta emitter, and ^{90}Sr + ^{90}Y, a medium and a high energy beta emitter.

3.31. The following example demonstrates how to go from information on the radionuclide vector and the individual clearance levels for radionuclides present to identifying 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 to the total activity with 75% and 25%, respectively. The derived clearance level for a mixture of radionuclides in this particular example is the following[3]:

$$\frac{1}{CL_{\text{eff}}} = \frac{0.75}{1\ \text{Bq/g}} + \frac{0.25}{0.1\ \text{Bq/g}} \quad (1)$$

$$CL_{\text{eff}} = 0.31\ \text{Bq/g} \quad (2)$$

In order to demonstrate compliance with this effective clearance level for a mixture of radionuclides, one or more easy to measure radionuclides should 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 nuclide in this given mixture, is then calculated by multiplying the CL_{eff} 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 activity of ^{60}Co below this level can be cleared.

3.32. Following characterisation, 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 correlation factors between different radionuclides. This will

[3] CL is used in this context to represent Xm from the equation I.2 of the GSR PART 3.

then feedback into additional characterisation work, followed by a revised monitoring scheme for the clearance process.

MANAGEMENT OF THE CLEARANCE PROCESS

3.33. The clearance process requires careful planning and implementation in order to achieve optimum performance. This section describes good practice concerning the management of the clearance process in situations where this a regular process and where the material throughput is substantial, i.e. where the clearance process is not applied on a sporadic basis or for small quantities. Some aspects are still valid for small quantities of cleared material and should be applied in accordance with the graded approach.

Assignment of responsibility

3.34. Responsibility for the clearance process usually lies with the radiation protection department of a facility, in which radioactive material is handled. The staff implementing the clearance process should clearly be allocated and properly trained. The number of staff needs to be commensurate with the required measurement capacity and the quantities to be handled.

3.35. Additional staff are required to keep track of the material undergoing the clearance process by updating databases on the material and maintaining the documentation. Staff will also be needed to ensure continued movement of material and monitoring segregation of material that has been cleared.

Prerequisites for the clearance process

3.36. The implementation of the clearance process will require sufficient and adequate equipment to perform the radiation monitoring and equipment to handle the material. The area where clearance measurements are being performed should be cleaned previously and should have a low radiation background.

3.37. Prerequisites for the clearance process are a thorough radiological characterisation, where sampling has been performed in a representative way for to the facility or facility section, taking account of the operating history, and an adequate analysis of the radionuclide mixtures (including specification of the key nuclides for the facility or facility section) has been carried out.

3.38. The results of the radiological characterisation serve as the basis for forming appropriate batches of material in the clearance process. Using batches of materials with similar characteristics enables the clearance process to be more efficient than using batches of highly heterogeneous material, as the relevant settings of the measuring instruments are very similar for material with similar characteristics.

3.39. A further prerequisite for the clearance process is a database system in which the identification and location of the material and the results of the clearance measurements can be updated to reflect the current situation at any time.

3.40. The clearance process for materials is most effectively implemented if clearly assigned areas exist for material transfer, buffer storage, surface-related and mass-related measurements, and staging areas where the cleared material can be placed until it can be removed from the facility. Since the clearance of building structures, sites and floor slabs will be performed in situ, this is not practicable, hence appropriate processes should be developed taking similar aspects into consideration.

3.41. The following description refers to an idealized clearance process for solid materials. In practical cases, individual steps can be omitted or carried out in a different sequence.

- 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. The individual parts usually have a size that enables them to be handled with a small electric hoist or by hand.
- On the buffer storage area, the material is assembled into batches depending on its origin and characteristics, in particular the technical system, the operation history, the radiological properties, and others. Batches entering the clearance process will therefore consist of material with similar characteristics.
- The next step in the clearance process is a measurement of surface specific contamination, including contamination of inner surfaces, if such measurements are required. This is carried out in an area dedicated for this purpose. The individual parts are put on tables or racks where the total surface can be accessed with contamination monitors.
- The readout of the surface-related measurements is evaluated against the surface-related clearance levels to be complied with (if any), taking the averaging area, nuclide vectors and other specifications of the process into account. If the surface-related clearance levels are complied with, the material can be moved on to the next station; if not, additional decontamination may be necessary and the material is sent to a controlled area for further treatment or for management as radioactive waste.
- Following demonstration of compliance with the surface-related clearance levels (if this step is required), the material is then moved to the next buffer storage area, awaiting measurements for determination of the mass-related activity.
- The next step is usually the measurement of the bulk activity to be compared with the mass-related clearance levels. 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 are used for this step (in such devices, the mass of material per measurement is usually in the range between a few 10 kg and a few 100 kg). In other cases, the bulk activity has to be determined from sampling, from surface-related measurements or from other measurement methods.

- The readout of these measurements is evaluated against the mass-related clearance levels, taking the averaging mass, nuclide vectors and other specifications of the process into account. If the mass-related clearance levels are complied with, the material has successfully passed all measurements; if not, a decision on alternative waste routes has to be taken.
- Before the material is released from the site, verification measurements by or on behalf the regulatory body may be required. In this case, the material is brought to a further buffer storage site outside or at the border of the controlled or supervised area, where these checks can be performed. If compliance with clearance levels has been confirmed, the material may be released.
- Finally, the material is moved to a place where it can be handed over to a (conventional) waste management company (e.g. a scrap dealer or a recycling company for building rubble) that will take care of the material in accordance with any conditions that may be posed by the conditional clearance option, if used.

3.42. Once a certain batch of material has completed the clearance process, the database and the documentation are updated and archived accordingly.

Practical considerations regarding a smooth implementation of the clearance process

3.43. Practical experience from numerous decommissioning projects involving clearance of large amounts of materials have shown that the following considerations are beneficial for a smooth implementation of the clearance process:

- Moving the material in suitable containers like boxes (e.g. 1 m^3) or drums (e.g. 200 l) instead of as single items has the advantage that material of similar origin is always kept together, that the material can be traced easily via the identifier of the box and that bulk measurements can be performed directly for these containers.
- Planning for buffer storage areas of sufficient size between the various steps of the clearance process enables a smooth material flow even if there are delays (e.g. due to temporary unavailability of a measuring instrument) at one step.
- Having separated buffer storage areas between the individual steps avoids unintentional mixing of material or cross-contamination between steps and prevents material from skipping a step in the process and being unintentionally cleared. The buffer storage areas will also allow segregation of the material according to nuclide vectors, origin, material type or other criteria.

- Traceability of the material at any time and a 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 after many years.
- Undertaking the measurements in areas with a low background dose rate will enable high quality measurements and decision thresholds that are appropriately below the clearance levels.

3.44. If a facility is too small to allow for adequate space for the clearance process, it may be advisable to construct a separate building where the process can be implemented. As the material undergoing the clearance process will have residual activities in the range of clearance levels, it will pose only trivial radiological risk, even if it turns out that part of the material does not comply with clearance levels. The separate building may therefore be of simple design without extensive demands for shielding or ventilation. Larger decommissioning projects have erected separate buildings dedicated to clearance only.

4. CLEARANCE OF SOLID MATERIAL

OVERVIEW

4.1. The following diagram provides an overview of the available clearance options for solid materials addressed in this document.

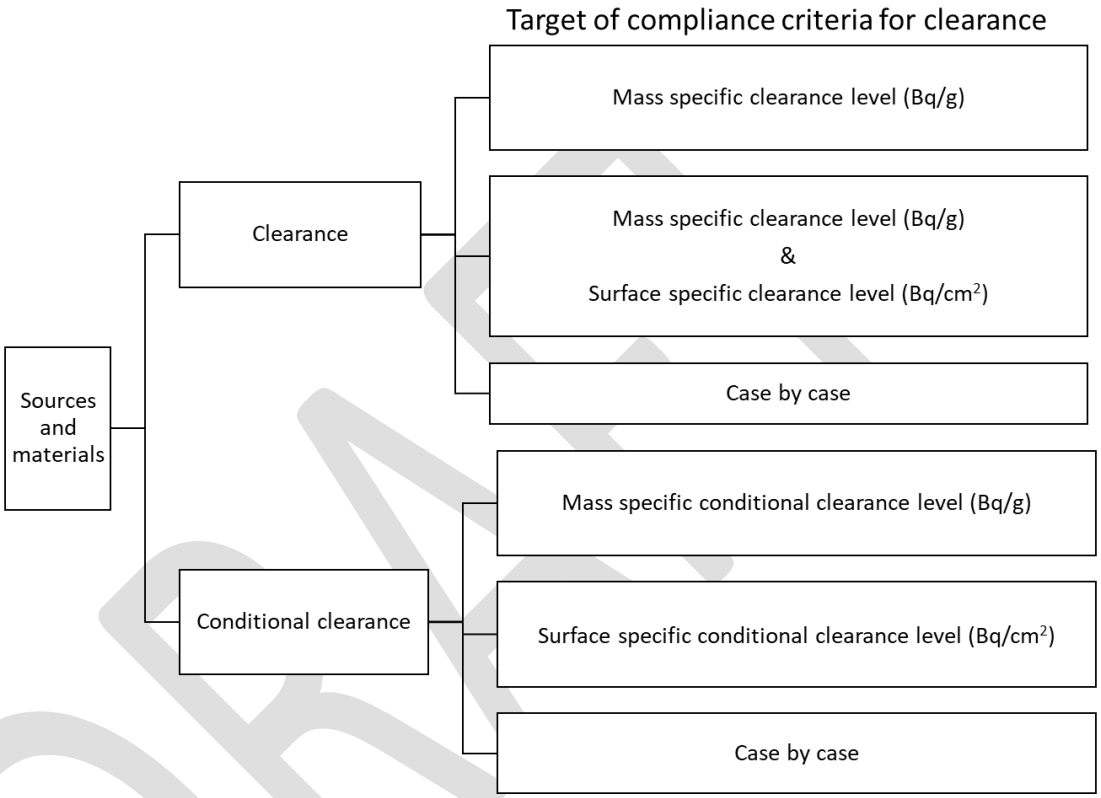

Fig. 4.1. Types of clearance options for solid materials

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

- details on the mass specific and surface specific clearance criteria that can be applied;
- considerations pertaining to averaging criteria and to aspects related to situations where mixing is part of the material management process after clearance;
- description of the implementation of clearance measurements and considerations of related uncertainty.

MASS SPECIFIC CRITERIA FOR CLEARANCE

4.3. The mass specific clearance levels specified in Tables I.2 and I.3 of [1] apply to solid materials, e.g. waste material comprising contaminated or activated structural materials, and contaminated excavated soils. They are not applicable to foodstuffs, drinking water, animal feed or any material intended for use in food or animal feed. They apply to bulk quantities of solid waste or material.

4.4. The methodology used to calculate these clearance values is described in SRS-44 [3]. For each radionuclide of artificial origin, the mass specific clearance level was determined on the basis of a set of exposure scenarios encompassing external irradiation, dust inhalation and ingestion (direct and indirect). The clearance levels were derived as the lower of the values obtained from:

(a) The use of realistic parameter values applying an effective dose criterion of 10 μSv/a;

(b) The use of low probability parameter values applying an effective dose criterion of 1 mSv/a and a skin equivalent dose limit of 50 mSv/a.

The derived results from the scenario calculations were then rounded to the nearest power of 10 using a near logarithmic rounding approach [3].

4.5. Mass specific clearance levels are specified in [1] for over 250 radionuclides of artificial origin. Values for other radionuclides of artificial origin should be derived using the models and approach described in SRS-44 [3]. Examples of values for other radionuclides can be found in regulations of some Member States [17,18] (regulations Germany, Switzerland]

4.6. A scenario-based approach was not used in the case of material that contains radionuclides of natural origin. Instead, the mass specific clearance levels given in Table I.3 of the GSR Part 3 [1] were derived using a pragmatic approach that involved consideration of the worldwide distribution of the concentration of radionuclides of natural origin present in material that is found in the environment. Values are given for all radionuclides of natural origin in the Uranium-238 decay chain and the Thorium-232 decay chain. The same pragmatic approach should be used to determine the mass specific clearance levels for other radionuclides of natural origin, e.g. primordial radionuclides. A mass specific clearance level of 1 Bq/g should be used for these primordial radionuclides pending establishment of specific values for these radionuclides on the basis of worldwide distribution.

4.7. Clearance of materials containing radionuclides of natural origin that arise from practices, e.g. ^{238}U in waste arising from nuclear fuel fabrication facilities, uranium enrichment, uranium conversion, should be subject to the clearance criteria given in para I.11 in GSR Part 3 [1]. Therefore, the clearance levels that are applied to these radionuclides of natural origin arising from practices should be derived using the

methodology described in SRS-44 [3]. These values should then be included in the summation rule when considering a mixture of radionuclides.

4.8. GSR Part 3 [1] also specifies that the mass specific clearance levels given in Schedule I, Table I.3 in [1] *may also be applied for the clearance of materials arising from practices subject to the clearance criteria given in para. I.11, pending establishment of radionuclide specific values for the radionuclides of natural origin given in Table I.3.* The member state should develop a programme for establishing these radionuclide specific values.

4.9. When establishing clearance levels for radionuclides of natural origin arising from practices, the following aspects should be considered:

a. The methodology described in SRS-44 [3] should be used;

b. The dose contribution from progeny radionuclides should be included in the calculations in order not to underestimate doses. Following the approach taken in SRS-44 [3], this is ensured by adding the dose coefficients of the progeny radionuclides to the dose coefficients of the parent radionuclides, using the appropriate weighting factors for the dose coefficients of the progeny radionuclides. For all pathways except the water pathway, the weighting factors for the progeny nuclides are taken as the maximum activity ratio that the respective progeny radionuclides will reach during a time span of 100 years. The time span of 100 years ensures that material that does not exceed the activity concentration values at a certain time will also not do so at any later point in time, within a reasonable time frame. For the water pathway, the calculations should consider the peak dose calculated over time i.e. there is no specified cut-off time.

c. Mass specific clearance levels should clearly specify the radionuclides in the decay chain that have been included in the calculations. An example of an approach to radionuclide chains is given in Schedule I of GSR Part 3 [1] for the derivation of exemption levels for moderate amounts. If all the radionuclides in the decay chain present in the waste have been considered in secular equilibrium in the calculations then only the clearance level of the parent radionuclide needs to be considered. Otherwise, the sum of fractions rule needs to be applied to ensure that all the radionuclides in the decay chain are considered. For example, for wastes containing ^{226}Ra and progeny, if the calculations considered the decay chain in secular equilibrium (^{226}Ra and all progeny as listed in Schedule I of GSR Part 3), then only the clearance level of ^{226}Ra needs to be considered. However, if the waste contains the entire ^{238}U decay chain in secular equilibrium then the pre-cursers ^{238}U, ^{234}Th, ^{234m}Pa, ^{234}U, ^{230}Th also need to be considered, using the sum of fractions rule. Using the nomenclature given in Schedule I of GSR Part 3, the clearance level for the ^{238}U chain in secular equilibrium would be derived from applying the sum of fractions rule to the

values for ^{238}U, ^{234}U, ^{230}Th, and ^{226}Ra only because the other radionuclides in the decay chain are already included in the calculations. Alternatively, calculations could be performed for ^{238}U-sec which explicitly includes all the progeny in secular equilibrium.

4.10. The methodology in SRS-44 [3] focusses on the handling (transport, trade, use or disposal) of the material outside the facilities in which they arise (reactors, accelerators or laboratories). The scenarios used to derive the mass specific 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). The scenarios relevant to direct handling of the cleared material (without a significant decay time prior to start of the exposure) should be added to the methodology used in the SRS-44 [3] when calculating activity concentration values for very short-lived radionuclides (fraction of a day or less).

4.11. Direct handling of the material is considered in the scenarios used to determine the exemption levels for moderate quantities given in Table I.1 in [1]. Hence, for those short-lived radionuclides where mass specific exemption levels are given in Table I.1 but there is no clearance level in Table I.2, the following alternative approach could be taken:

a. Use SRS-44 [3] methodology to obtain mass specific activity concentration that meets the clearance criteria;

b. Identify the mass specific level for moderate quantities from Table I.1 of the GSR Part 3 [1] that meets the clearance criteria;

c. Take the lesser of the two results as the clearance level.

CONSERVATISM IN THE DERIVATION OF CLEARANCE LEVELS FOR UNCONDITIONAL CLEARANCE

4.12. The derivation of clearance levels for unconditional clearance as performed in IAEA SRS-44 [3] includes a number of conservative assumptions that were deliberately taken to encompass a large variety of exposure situations that could arise as a consequence of clearance from all types of materials. This general approach is explicitly stated in SRS-44 [3]: *"The approach to encompass the variety of situations that may be found in Member States around the world necessarily requires a degree of conservatism"*. Nevertheless, several methods have been applied to keep the overall degree of conservatism at a reasonable level:

- Two sets of scenarios have been used in parallel, one applying so-called "realistic scenarios" for an effective individual dose limit of 10 µSv/a, and one applying so-called "low probability

scenarios" for an effective individual dose limit of 1 mSv/a. In this way, parameter values in general could be chosen on the less conservative side for the "realistic scenarios". This approach fully satisfies the criterion for clearance as defined in para. I.10 and I.11 of GSR Part 3 [1]: *"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"*.

- The scenarios for workers and members of the public have been devised in such a way that those exposure pathways that could occur simultaneously (e.g. external irradiation and inhalation) have been analysed together and their dose contributions have been added. In this way it was possible to formulate the entire scenarios in a conservative way and not necessary each exposure pathway individually. As radionuclides generally have only one dominating exposure pathway, this reduces conservatism.

4.13. The following points, on the other hand, show introduction of conservatism in the model:

- The application of the summation rule for cases where there is more than one radionuclide present is inherently a conservative approach since the pathways of exposure of the critical group of exposed individuals are not necessarily the same for each nuclide, because of partitioning or separation of nuclides by processes. A less conservative, but impractical approach would be to sum the contributions of the radionuclides in the radionuclide mixture for each scenario and each exposure pathway first and evaluate then the activity value leading to full utilisation of the dose criterion 10 µSv/a.
- The dose contribution from progeny radionuclides is always included together with the parent radionuclide with a percentage that corresponds to the highest ingrowth with a time span of 100 years after clearance. This leads to an overestimation of the dose coefficients for the mixture of parent and progeny nuclides in such situations.

4.14. Many individual parameter values have been chosen on the conservative side, mainly for the so-called "low probability scenarios". Examples:

- In many so-called "low probability scenarios", absolutely bounding values have been assumed
  - for exposure times (8,760 h/a for the full year, 1,800 h/a for the full working year),
  - for dilution (factor 1, i.e. no dilution),
  - for decay time prior and during the scenario (1 d / 0 d corresponding to virtually no decay at all),

- for unfavourable exposure conditions.

- The groundwater model contains a number of highly conservative assumptions, such as:
  - The model assumes conservatively that the whole inventory of radionuclides in the material is available for migration.
  - The Kd values have been selected conservatively from the values published in literature for different elements.
  - The private well from which groundwater is abstracted for several uses has been assumed very close to the deposited material, thus reducing the effect of radioactive decay significantly.
- Skin contamination: Dose coefficients for the skin relate the skin equivalent dose to the concentration of radionuclides on the skin. The skin dose coefficients were taken conservatively for a skin surface weight of 4 mg/cm², while contamination would predominantly occur on the hands where the skin surface weight is significantly higher.

4.15. Less conservative parameter values have been applied in the so-called "realistic scenarios". In summary, the approach adopted in IAEA SRS-44 [3] follows international good practice in deriving clearance levels.

SURFACE SPECIFIC CRITERIA FOR CLEARANCE

4.16. For surface contaminated items where radioactivity may be concentrated on surfaces, compliance with the mass specific clearance level (activity concentration per unit mass) may not be sufficient in all cases because there are additional considerations relating to the handling of the material. Examples are surface contaminated items with a large ratio of surface area to volume, such as paper, card, plastic sheeting and clothing, and glass and thin metal sheeting of low to moderate density. In these cases, surface specific clearance levels should be agreed with the regulatory body. The authorized party should then comply with these surface clearance levels, in addition to complying with the general (unconditional) clearance levels expressed as activity concentration per unit mass.

4.17. The radioactivity inside and on the surface of the cleared material has to be appropriately limited to guarantee compliance with the constraint for the effective individual dose of 10 µSv per year. An example of potential outcomes when applying both surface and mass specific clearance levels is given in Table 1. For items and bulk material, this is usually accomplished by limiting the mass-specific activity, e.g. as provided in Table I.2 of GSR Part 3 [1]. However, when mass-specific clearance levels cannot be

applied or are not sufficient as the sole criterion, the surface-specific activity should be appropriately limited. An example of a situation where surface-specific activity is limiting is the clearance of contaminated pipes and beams. In particular, surface-specific clearance levels limit the contamination that is directly accessible and could be mobilised during handling. They also limit the contamination on larger areas from which direct exposure by external irradiation could result.

Table 1. Example of potential outcomes when applying both surface and mass specific clearance levels (adapted from UK Good Practice Guide [19])

Surface Specific Clearance Levels[4] [Bq/cm^2]	**Mass Specific Clearance Levels [5] [Bq/g]**	**Outcome**
Average < relevant limits	Average < relevant limits	No radiological requirement to undertake separation and segregation prior to clearing waste.
Average > relevant limits	Average < relevant limits	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 10 µSv per year.
Average < relevant limits	Average > relevant limits	Unless commercial considerations (e.g. recycling or re-use 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 > relevant limits	Average > relevant limits	Manage as radioactive waste in accordance with national strategy for management of radioactive waste.

[4] For example, paint, laminate or region of increased radionuclide concentration

4.18. Surface-specific clearance levels have not yet been provided in guidance issued by the IAEA. However, a number of international studies and recommendations is available that use dosimetric modelling to establish a link between the surface specific contamination and the resulting annual dose to an individual. Examples of dosimetric models are given in Annex I.

4.19. International studies and recommendations are available that provide surface-specific clearance levels applicable to clearance of items made of metal and other materials. Recommendations RP 101 [20] in combination with RP 89 [21] issued by the European Commission contain surface-specific clearance levels for metallic items both for direct reuse and for recycling of metallic material by melting. The set of values recommended for direct reuse of items can be considered as surface specific-clearance levels for unconditional clearance of objects of all kinds, because the exposure scenarios considered are independent of the type of material. The examples of surface-specific clearance levels applicable for unconditional clearance are given in Annex II. In general, calculations of these surface specific clearance levels consider both the fixed and removable activity on the surface of materials.

4.20. The numerical values for some radionuclides differ among different international studies and recommendations. The differences are due to different conditions and parameters assumed in derivation of surface-specific clearance levels (material, size of the item, geometry, exposure scenarios, and other aspects). Hence, it is to be expected that various studies will determine different surface specific clearance levels that comply with the same dose criteria. Therefore, application of a set of existing values (derived for a particular situation) to a different situation should be done with care, taking into account adequacy of assumptions, characteristics of the material, exposure scenarios used, and other aspects. For example, applying surface contamination levels derived for clearance of large objects would be too strict and conservative for small objects.

4.21. This Safety Guide does not provide a single set of radionuclide specific values, but offers to the Member States a range of examples for selection in accordance with needs and prevailing circumstances.

4.22. Para 106 of SSR-6 [22] states that the transport regulations apply to radioactive material, where radioactive material is defined as material that exceeds the mass specific exemption values and total activity exemption values defined in the Schedule I, Table I.1 of the GSR Part 3 [1]. Hence material that meets these mass specific exemption levels is not subject to transport regulations.

4.23. Para 107 (f) of SSR-6 [22] states that the transport regulations do not apply to natural material and ores containing naturally occurring radionuclides, which may have been processed, provided the activity

[5] For example, brick, blockwork or metal structure

concentration of the material does not exceed 10 times the values specified in Table 2 of SSR-6 [22], or calculated in accordance with paras 403(a) and 404–407 of SSR-6 [22]. For natural materials and ores containing naturally occurring radionuclides that are not in secular equilibrium the calculation of the activity concentration are required to be performed in accordance with para. 405 of SSR-6 [22]. Hence, material that meets these levels is not subject to transport regulations.

4.24. Any material that has been cleared on the basis of mass specific clearance levels established in the Schedule I, Table I.2 and Table I.3 of the GSR Part 3 [1] will have mass concentrations that meet (are equal or below) the mass specific exemption levels. Hence, this cleared material is not radioactive material anymore [7] (IAEA Safety Glossary), and therefore not subject to transport regulations. Similarly, material cleared on the basis of surface specific clearance levels is no longer defined as radioactive material with surface contamination, and thus not subject to transport regulations.

4.25. Surface specific clearance levels have not been established in all Member States yet. In such a case, compliance with mass specific clearance levels has to be demonstrated for materials with surface contamination. This can be achieved by converting the total activity on the surface to a mass specific activity (Bq/g) taking account of the total mass of the material below the surface (i.e. the mass specific activity should not be calculated by just using the thickness of the thin surface contamination layer). In this process, considerations related to the radiological models used for deriving the clearance levels need also to be taken into account, particularly the averaging mass (see para 4.34-4.36).

4.26. If the contamination has penetrated through the surface and into the volume, a prudent approach is to estimate the total activity using the sum of the contamination present directly on the surface and the contamination inside the volume beneath the same surface area. For comparison with mass specific clearance levels, this activity should be divided by the total mass below the surface. In this process, considerations related to the radiological models used for deriving the clearance levels need also to be taken into account, particularly the averaging mass (see para 4.34-4.36). This approach also applies to materials with activation inside the volume of the material.

CASE-BY-CASE APPROACH

4.27. International and also national guidance on clearance and clearance levels are based on the application of generic methods for evaluation of the radiological consequences of clearance. This is usually accomplished by using generic models that describe possible exposure scenarios caused by clearance in a generic and enveloping way. These generic models need to be biased to the conservative side so as not to underestimate the possible exposure in all relevant circumstances. An example for such a model can be found in SRS-44 [3].

4.28. There are, however, situations where the 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. A case-by-case approach should then be used, in which a radiological model is developed specifically comprising the relevant exposure scenarios and parameter values for this case. Key parameters where significant deviations from generic values are relevant are likely to include exposure times, distances on which dose rates from external irradiation are based, shielding geometries, concentration of contaminated aerosols or quantities of materials to be cleared.

4.29. Furthermore, the analysis of a specific situation may also show that certain scenarios having been included in the corresponding generic model are not relevant to this particular case. These scenarios should then be left out of further consideration in the analysis of this specific situation.

4.30. The radiological calculations should follow the same principles as the generic models, e.g. the one described in SRS-44 [3]. This means that all exposure pathways (external irradiation, inhalation of contaminated aerosols, direct ingestion of small quantities, secondary ingestion of radionuclides via the food chain and skin contamination) should be adequately included in the scenarios. Although being based on the specific features of the situation to be analysed, the parameters describing the exposure situations should still be chosen in such a way that their possible variation is sufficiently encompassed. Example: While in a certain clearance practice the assumption of an exposure time of a full working year (e.g. 1,800 h/a) may be too high and measured real exposure times vary between 240 and 480 h/a, it would be prudent to use a value of 500 h/a for the exposure time in this case-by-case analysis to account for contingency.

4.31. The use of case-by-case approaches for clearance is generally encouraged, because in this way certain situations that are not adequately covered in generic international guidance can be analysed in a specific country, taking account of industrial, environmental, climatic and other features and regulatory requirements valid for this country. In this way, inappropriate use of generic clearance levels can be avoided.

4.32. Following the graded approach described in section 2 and the criterion for clearance specified in Schedule I paragraph I.10 (b), for activity concentrations that exceed the relevant general clearance levels given in Table I.2 by several times (e.g. up to ten times), 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. The mechanism for giving effect to such a decision will depend on the nature of the national regulatory infrastructure. In many cases, a decision will be made by the regulatory body on a

case by case basis, following notification by the authorized party to the regulatory body. This can be considered to be an example of conditional clearance, see Section 7 for further details.

4.33. It can generally be expected that a case-by-case approach will lead to less restrictive clearance levels, as such an approach will only be endeavoured if the characteristics of a specific situation have been identified to be less conservative than in the generic model. It should, however, be kept in mind that it might cause problems in international trade if material that has been cleared according to clearance levels based on a case-by-case approach in one country is then exported to another country where e.g. the internationally agreed unconditional clearance levels as provided in Table I.2 of GSR Part 3 [1] are valid. Therefore, a case-by-case approach should be limited to situations where international trade will not play a role or is at least highly unlikely, or where this is explicitly featured into the model, appropriately informing the target country about this clearance practice.

AVERAGING MASSES AND AREAS

4.34. The general clearance levels specified in Schedule 1 of GSR Part 3 [1] for artificial radionuclides are calculated using a set of scenarios, and these scenarios consider exposure to a large quantity of homogenous material. For example, the transport scenario considers a truck containing 10 t of material and the landfill scenario considers even larger quantities. When applying the clearance levels, the regulatory body should recognise that they were derived for bulk amounts and that the averaging should be done accordingly.

4.35. In this context the regulatory body should determine or approve appropriate averaging masses to be used as the decision units in the clearance compliance measurements, and averaging procedures used by the authorized party should take this into account. Examples of appropriate averaging masses are few hundred kilograms or a few tons. The regulatory body should ensure that the averaging procedure is not used to intentionally release material above the clearance levels (see para 4.38). The authorized party should make averaging procedures an integral part of the verification scheme, selected according to the type of material.

4.36. In the case of surface specific clearance levels, these are intended as an average over moderate areas and regulatory authorities should authorise, depending on the type of material, contamination and homogeneity of the contamination, averaging areas of several hundred square centimetres up to 1 square meter, in case of unconditional clearance[6]. For non-accessible surfaces for which some degree of surface

[6] Averaging areas for conditional clearance could be higher, for example 10 square meters for clearance of buildings for demolition.

contamination can be reasonably expected, the authorized party should make a conservative assessment of the surface activity for comparison with the clearance levels.

4.37. Averaging masses and areas for decision making on compliance with clearance criteria (decision units) should be distinguished from masses and areas used for actual measurements (measurements units). For example, multiple samples of 100 g of soil could be used to determine whether a mass of a few tons complies with the clearance levels. In any case, the measurement unit (sentencing unit) should be smaller than or equal to the decision unit.

4.38. In deciding on a measurement strategy, the authorized party should group the material so that it is as homogenous as possible in relation to both material and origin, and thus radionuclide spectrum and activity level. Variations of activity level within the averaging unit of mass or area for decision making should be allowed. For example, variations of up to a factor of 10 with respect to the average value for the decision unit are generally considered to be acceptable, whereas a greater variation would be acceptable if the overall average concentration was a very small fraction of the clearance level. Also, the maximum concentration in any measurement unit should not exceed ten times the clearance level as long as the average value over the decision unit does not exceed the clearance level (Appendix A of the IAEA TECDOC-1000 [23]).

4.39. The authorized party should make use of the maximum practicable and permitted averaging areas or masses when designing the monitoring regime as this improves the efficiency of the clearance process. The monitoring regime 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, appropriate use of 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.

4.40. When the authorized party makes a single measurement or multiple measurements to determine whether the material is in compliance with the clearance level, each measurement is based on a measurement unit that is defined by the chosen monitoring regime and instrumentation (e.g. contamination monitor, drum monitor, bulk monitor). The size of the measurement unit should be chosen based on practical considerations that reflect the size of an object and how the material will arise or be measured (e.g. a drum of waste, or an excavator bucket, or the actual geometry of the measurement system).

4.41. The authorized party should select measurement units and should propose decision units that are sufficiently representative of the material, with appropriate adjustments to satisfy homogeneity limitations and confidence level requirements for the clearance measurements. The measurement and decision units

should therefore usually be related to the same origin of material for clearance, or one of several origins of a very similar nature. In general, larger measurement and decision units are acceptable where the contamination in the material is reasonably uniform and smaller measurement units should be used where inhomogeneity is significant. The decision units should be agreed with the regulatory body and formally recorded by the authorized party as part of the clearance measurement process.

4.42. The decision units will put an indirect limitation on the size of "hot-spots" (see next sub-section), since the total activity per decision unit will be the maximum activity level of the hot-spot. The regulatory body should define a maximum value for a hot spot, that should be kept in mind when defining the size of a decision unit.

4.43. If the results of samples taken from the bulk waste or material are subject to considerable variability, then averaging over the whole waste or material mass (as a single decision unit) is unlikely to be acceptable without proper (documented) consideration of:

i. the practicability of segregation and separation;

ii. suitable revision of monitoring and numbers of samples;

iii. suitable reduction in the size of each measurement unit (sentencing mass or volume);

iv. whether it is practicable to make further measurements to identify each area or volume containing significant concentrations of radioactivity;

v. whether it is practicable to remove or segregate small areas or volumes containing significant concentrations of radioactivity (hot spots);

vi. the potential radiological significance of inhomogeneity.

Presences of hot spots and distribution of activity with depth and area

4.44. One of the most challenging tasks in the release of material from regulatory control 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 non-uniformity. Hot particles are generally small items which are not part of the material in which they are found, for example small metal flakes of high ^{60}Co content or small pieces of spent nuclear fuel which may be found in a cooling pond. These can be radiologically significant (giving doses that can lead to deterministic effects) and should be removed before the clearance process begins. It is important that during any decommissioning related survey the potential for hot particles is considered and, if found to be possible, that the monitoring and clearance process will identify their presence, rather than just considering them as contributors to the total activity of a large sentencing mass.

4.45. Hot spots, in terms of a local non-uniform distribution giving activity concentrations above the clearance levels, are to be expected. It is important that the range of activity concentrations in a sentencing volume (measurement unit) is reasonably restricted. The regulatory body should approve or specify additional monitoring criteria to the existing averaging criteria, in order to detect and manage any hot spots in the material considered for clearance.

4.46. In cases where the compliance with the surface specific clearance levels is demonstrated using instruments with much smaller surface area than the averaging area, information on the homogeneity and hence the presence of "hot spots" can be derived from the variation of the readouts of individual measurements. The final value of surface specific activity for comparison with clearance level is derived from an averaged result. It is generally possible to set a rate-based alarm related to the presence of "hot spots" for individual measurements, which will identify any particularly active areas.

4.47. For bulk material, many processes involving bulk measurement are based on scanning or multi-point measurements, both of which can be set up to identify particularly active volumes. Another approach to demonstrating compliance with the hot spot criteria [23] (TECDOC-1000) is to use monitoring techniques sensitive enough to detect 100% of the contamination in the 'worst case' 10% of the volume. For example, if measurements are taken on the outside of a drum, calculations to demonstrate compliance could assume that all of the contamination is located in the centre of the drum (surrounded by clean material), furthest from the detectors and shielded by the clean contents. This will result in higher efforts for measurement (e.g. longer counting times, more measurements or more sensitive detectors), but the additional cost may be small compared to using an additional sampling measurement to demonstrate compliance with the average and the hot spot criteria. This approach works well for a drum where the density is low and the nuclide emits very high energy gamma radiation, e.g. ^{60}Co in concrete rubble, and where the general level of activity is well below the clearance level. It is not recommended if the material itself provides effective shielding, the gamma energy is lower (e.g. metal contaminated with ^{241}Am), and there is significant bulk activity concentration, as it will lead to significant over-estimation of the radionuclide content.

IMPLEMENTATION OF CLEARANCE MEASUREMENTS

Monitoring programme and strategy

4.48. The monitoring programme to support clearance process should be based on the results of the characterization, where isotopic vector or key nuclides have been identified and the level and the location of contamination have been quantified, as described in Section 3 (Characterisation of the Material to be Cleared).

4.49. The monitoring programme should be managed as a material flow process that starts with well characterized material to be evaluated for clearance.

4.50. Material presented for clearance should be sorted into batches, consisting of the same type of material, same radionuclides, same history. This information from the characterization of the material should be used as a technical basis for the establishment of the monitoring programme.

4.51. Within the monitoring programme, 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 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 into the batch process and the output options, being cleared material or radioactive waste. The optimal strategy should be defined based on radiological criteria and cost-effectiveness.

4.52. In the definition or selection of batches, the spatial distribution of the contamination is an important selection criterion. Distinction should be made between bulk contaminated material and surface contaminated material.

4.53. The monitoring strategy should determine which of the three monitoring techniques (surface measurement, bulk measurement and sample analysis) is the most appropriate for a given batch and depending on the material being considered, a combination of techniques can be required.

4.54. The choice of the monitoring technique implies the selection of radiation monitoring equipment. The response of the equipment will depend on radiation type, energy and geometry. A good knowledge of the radionuclides to be measured is therefore crucial and should be determined prior to the monitoring programme. Depending on the case, key radionuclides should be defined in the radionuclide mixture and the contribution of other nuclides can be derived by the use of correlation factors to these key nuclides. Based on this information, the appropriate radiation measurement instrument should be selected for monitoring for compliance with the clearance levels, taking into account the level of activity concentration that has to be verified. Information on the selection of the instrument can be found in ref. [12] (SRS-67).

4.55. The response of the measurement equipment, expressed in operational units (e.g. counts integrated over a period of time), should be converted into activity values (Bq). The equipment might return a total number of counts over the whole energy range or provide a number of counts as a function of the energy. In the first case, identification of the measured radionuclide will not be possible, whereas in the latter, the spectral information will allow radionuclide identification. The choice will be part of the monitoring strategy and a sequential combination of both might be necessary.

Surface contamination measurements for compliance with mass specific clearance level

4.56. If the contamination in the materials for clearance is limited to the surface, i.e. for impermeable materials, surface contamination measurements need to be applied for compliance with mass specific clearance level instead of performing mass related measurements. If the materials are relatively small objects, of the order of the averaging mass, for example, iron plates or stainless steel pipes, it is easy to convert from total radioactivity obtained by the surface measurement to mass activity concentration taking account of their thickness, their densities and the number of the contaminated surfaces (one surface or two surfaces) [24] (DIN25457). In such a case, it may be useful to derive a surface criterion from the mass specific clearance level. This derived criterion can be used to demonstrate compliance with the mass-related clearance levels.

4.57. For the assessment of surface contamination, the principles and methods described in international standards (e.g. ISO standard on measurement and evaluation of surface contamination ISO-7503-2016 [25]) 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 and auditable for clearance measurement, then clearance measurements should use integrated counts over a defined time.

Measurement techniques

4.58. Special attention should be given to the condition of the surface to be measured when using a direct measurement technique. The ideal surface should be clean and flat. Cleanliness is required since dirt such as dust, grease, rust can mask the signal to be measured through absorption, especially for alpha contamination. It is therefore strongly recommended to clean the surface before measurement. This cleaning can be considered as a decontamination in case of the presence of non-fixed contamination. In addition, assessment of the removed fraction (e.g. by measurement of the cleaning tool) could give information on the nature of the contamination.

4.59. An uneven surface can occur e.g. after wall decontamination, which could affect the direct measurement due to unequal distance from surface to detector, affecting the detector response. On the flatness of the surface, the standard [25] (ISO -7503) states: "*Generally, it is applicable to well defined flat surfaces where direct methods are applicable, however, it can also be used for surfaces which are not flat and where indirect wipe tests would be appropriate*."

4.60. For total gamma measurements, the calibration procedure is generally complex and is described by the manufacturer of the instruments. The procedure can be greatly simplified by performing the calibration for a single radionuclide and deriving the calibration factors for other nuclides through calculations or through correlation factors as defined in the radionuclide vector.

4.61. For in-situ gamma spectrometry, the situation is even more complex since the response to individual radionuclides, in addition to the energy of radiation, also depends on the distribution over the surface and/or in the volume underneath the surface. Computer models are available that allow calculations of the calibration factors from a given radionuclide composition and spatial activity distribution. Details are supplied by the manufacturers of instruments and software for calibration calculations.

4.62. Samples should be taken through smear samples (wipe samples) in case of surface contamination, or through collection of a small fraction of the material itself. In case of smear samples, they should be analysed through indirect surface contamination assessment or be subject to sample preparation and measurements in a laboratory environment (for example, dissolution for tritium measurements). In case of material samples, they should be analysed in laboratories with specific equipment for spectral analysis. The laboratories should have a quality assurance system and should be properly accredited according to national requirements or international standards, for example [26] (ISO/IEC 17025:2005).

4.63. When sampling is used for compliance verification, additional issues should be addressed, such as sampling position, minimum sample size and number of samples. When the spatial distribution 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 required number of samples. The position of the individual samples should be properly recorded. The sample measurements should provide information on the activity distribution in the material as a whole, to be compared with the clearance levels.

4.64. The minimum number of samples to be taken should be determined by the median value and the standard deviation of the activity concentration distribution on the basis of a statistical compliance test. The number of samples should be increased if the results of the statistical analysis are not satisfactory with respect to median value and standard deviation. The decision on clearance of material will be based on a statistical test on the measured activity concentrations. For the selection of the proper test, guidance can be found in international references [16] (MARSSIM).

4.65. Each instrument has a threshold for detecting radiation of a specific type. In order for an instrument to be suitable for compliance verification with the clearance level for a specific radionuclide, this threshold should be well below the clearance level to account for uncertainties (see paragraphs 4.70-4.89). The threshold is usually referred to as the detection limit or the MDA (minimum detectable activity). It should be determined according to international standards (e.g. ISO-11929). The MDA does not only depend on the measurement technique, but also on the measurement conditions, such as background and measurement time, and the accepted level of confidence in the measurement. The complementary concept

to the MDA is the "maximum missable activity" [19], and this can be used in communication with the regulator and other interested parties.

4.66. In case of sampling, the minimum sample size should allow for providing a signal in the detector well above the detection limit when the activity concentration in the sample is a significant fraction of the clearance level. This fraction should be approved by the regulatory body. Possible loss of material in the sample preparation process should be taken into account when calculating the minimum sample size.

4.67. When using the concept of clearance of material, the background activity in the material prior to the planned exposure situation should be subtracted from the measured activity in the material. The concept of clearance for practices should be applied to the mass specific (or surface specific) activity level that is above the natural background level in the material, not to the total activity level in the material. Activity from sources other than the licensed practice itself, for example naturally occurring radionuclides in building material (^{238}U- and ^{232}Th- decay chains, ^{40}K) or fallout from nuclear weapon tests and nuclear accidents (e.g. ^{137}Cs), should be disregarded when performing clearance measurements. Cosmic radiation and naturally occurring levels of primordial radionuclides should also be disregarded when performing clearance measurements. For NORM industries, the clearance levels for radionuclides of natural origin apply to the total activity level in the material.

4.68. When determining what background needs to be subtracted during clearance measurements, variations in the background activity level should be considered. The regulator should approve the proposed level to be disregarded. Especially with total gamma measurements, the contribution from the activity that can be disregarded needs to be carefully established in order not to misinterpret the measurement signal from the activity undergoing clearance. The activity to be disregarded needs to be established using a suitable low percentile (e.g. 5 % percentile) from the distribution of activity values, thus preventing overestimating the signal to be subtracted. The distinction between these various contributions to the total activity can be significantly improved by using spectrometric information.

4.69. The requirement on the MDA for clearance verification has an impact on the acceptable background conditions during the measurement. They can be optimized by careful selection of the location, by adding shielding to the detector or by increasing the measurement time. The MDA also depends on the uncertainty level that is tolerable in comparing measured value with clearance level (see next section on uncertainty considerations).

UNCERTAINTY CONSIDERATIONS

4.70. The clearance process, in particular the measurements, involve a number of uncertainties that have to be properly taken into account, depending on the measurement techniques. These involve pure

statistical uncertainties of the counting process (so-called "type A uncertainties"[7]) and uncertainties relating to situations in the measurements process that can be determined as standard deviations of probability density functions based on experience or other information (so-called "type B uncertainties"[7]). The following list gives an overview of those type A and type B uncertainties that are most relevant for clearance measurement processes. These uncertainties are then addressed in the following sections in more detail.

- Statistical uncertainties of the counting process (type A);
- State of the surface of the measured material (type B);
- Fluctuation range of the geometry and the self-shielding of the measured material (type B);
- Fluctuation range of the activity distribution in the measured material (type B);
- Fluctuation range of the background effect (type B);
- Fluctuation range of the activity fractions of the radionuclides in the material in relation to the specified fractions in the nuclide vector (type B);
- Fluctuation range of the wiping efficiency during indirect surface activity measurement (type B);
- Fluctuation range of the content of natural radionuclides and other radionuclides to be disregarded in the measured material (type B);
- Measuring uncertainty during the calibration used (e.g. reference measured material for the total gamma activity measurement) (type B).

4.71. Taking the influence of all Type A and Type B uncertainties into account, it can be determined whether a certain measurement technique including all relevant parameters, such as efficiency, calibration and measurement time, qualifies for demonstrating compliance with clearance levels. One prerequisite will be that the detection limit determined for this measurement technique will be below the clearance levels. Details on this process are provided in the document "Evaluation of measurement data — Guide to the expression of uncertainty in measurement"[7].

4.72. When performing actual clearance measurements, the upper confidence level of the measurement effect must be below the clearance level (expressed in the same unit), taking all relevant uncertainties into account. Examples for this are provided in [16] (MARSSIM), [24] (DIN25457), [27] (ISO-11929), etc.

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[7]) Type A and Type B uncertainties are defined in the guidance document "Evaluation of measurement data — Guide to the expression of uncertainty in measurement" ("GUM"), JCGM 100:2008, of the Joint Committee for Guides in Metrology

Examples of linking the measurement uncertainty to the detection limit are provided in sections 5.1-5.3 of the Safety Report Series 67 [12]. However, care should be taken not to apply too much conservatism when considering all these sources of uncertainty. For example, if one of the uncertainties is biased to a very high level, then fluctuations relating from the other uncertainties are less important.

Treatment of statistical uncertainties of the counting process

4.73. Nuclear radiation measurements used for clearance involve counting of events (detection of photons, beta particles, alpha particles and others in monitors that either count the total effect or that have spectrometric capabilities. Examples for such instruments are contamination monitors with proportional counters or scintillation detectors, bulk monitors used on packages with up to several 100 kg of material, in situ gamma spectrometers used on packages or on surfaces, laboratory gamma spectrometers used for samples, liquid scintillation counters used on specifically prepared samples, and others. When these instruments count events for a certain period of time repetitively, the count rates will scatter around a best estimate (which is associated with the "real" activity value), usually following a normal distribution. This deviation between a single counting result and the (unknown) best estimate is a purely statistical effect and gives rise to a type A measurement uncertainty.

4.74. In addition, such measurement techniques to determine whether activity concentration values are in compliance with clearance levels rely on the performance of the instrument or the condition of measurement, for example, background level and measurement time.

Treatment of uncertainty related to the state of the surface of the measured material

4.75. The state of the surface influences the emission efficiency of the measured material for alpha and beta emitting radionuclides. In the case of metals, dirt or oxidation layers on the surface of the measured material typically influence the emission efficiency. The emission efficiency with respect to contamination monitors with proportional counters (sensitive for beta and alpha radiation) is strongly influenced by the layer thickness. A deeper penetration of the contamination is to be expected with porous materials, for example concrete and wood, which can be considered by separate tests. Up to a certain thickness of the layer, the effect can be taken into account by adjusting the value of the surface emission efficiency. If the layer becomes too thick, no meaningful measurement is possible anymore.

4.76. This uncertainty is relevant for all surface related measurements with beta and alpha sensitive instruments and needs to be included in the analysis of type B uncertainties. It is of minor relevance for gamma sensitive instruments (contamination monitors with scintillation counters), in situ gamma spectrometers or measurement of samples.

Treatment of uncertainty related to the geometry and the self-shielding of the measured material

4.77. When performing clearance measurements, the instrument is calibrated for certain geometries of the measured material, including assumptions on self-shielding. For example, plane surfaces with a certain distance between the surface and the window of the contamination monitor (e.g. a few mm) may be 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. In such a case, there are differences between the calibration geometry and the real measurement geometry, reducing the efficiency of the measurement process. This can be taken into account by correction factors or by using multiple calibration geometries that cover all conceivable geometrical situations.

4.78. In a similar way, the self-shielding of a large quantity of material measured in bulk monitors needs to be taken into account in the calibration process. In real measurements, there may still be deviations from the calibration, e.g. because the material is more densely packed. This effect can be evaluated e.g. by numerical simulations.

4.79. In all such cases, the possible variation of differences between real measurement situations and the calibration need to be evaluated and included in the analysis of type B uncertainties.

Treatment of uncertainty related to the activity distribution in the measured material

4.80. During calibration of surface measurements or of bulk measurements, certain assumptions have to be made with respect to the spatial distribution of activity on the surface or in the bulk of the measured material, respectively. Often calibration of surface measurement instruments is performed with homogeneous thin-layer sources of known activity and surface emission rate while real surfaces may exhibit localised rather than homogeneous contamination. Likewise, calibration of bulk monitors may be performed with dummies with homogeneous contamination while real boxes with scrap of building rubble have localised contamination. In both cases, the measurement result needs to be corrected for the difference in spatial activity distribution between calibration and measurements. This correction therefore needs to be included in the analysis of type B uncertainties.

Treatment of uncertainty related to the background effect

4.81. Any measurement process is influenced by photons or particles that do not originate from the material to be measured but have their origin elsewhere, such as other material, terrestrial or cosmic radiation. This is summarised by the background effect which needs to be measured separately for subtraction from the gross measurement effect as well as for determination of the detection limit. Although the background effect is regularly measured (e.g. before and after a measurement campaign

during the working day), variations of the background during the measurement campaign will occur, thus leading to differences between the previously determined background effect that is subtracted from the gross measurement effect to yield the net effect and the current background during a particular measurement. The variation of the background effect therefore has to be determined and needs to be included in the analysis of type B uncertainties.

Treatment of uncertainty related to radionuclide vector

4.82. For situations where the activity of more than one radionuclide has to be taken into account in the decision for compliance with clearance levels, the summation rule described in para. 3.7 has to be applied. In treating uncertainties due to a mixture of radionuclides, the concept of the radionuclide vector is applied. This implies inclusion of uncertainties in determination of correlation factors (activity ratios) between the activities of difficult-to-measure (DTM) radionuclides and activities of key radionuclides that are easy-to-measure (ETM).

4.83. The uncertainty in the determination of a certain correlation factor is associated with variations of the activity ratios from which this correlation factor was derived (e.g. as a mean value together with a standard deviation). Usually, correlation factors for radiologically relevant radionuclides (e.g. of Sr-90, using Cs-137 as a key nuclide) will be derived on a conservative basis so that the activity of the DTM nuclides will not be underestimated.

4.84. The uncertainty in the determination of correlation factors or radionuclide vectors need to be taken into account in the analysis of type B uncertainties.

Treatment of the wiping efficiency for indirect surface activity measurement

4.85. When surface activities are determined by wipe tests rather than by direct measurements, assumptions on the wiping efficiency have to be made. Usually, a conservatively small efficiency is assumed (often 10 % of the removable surface activity) to account for the fact that the real wiping efficiency is hard to determine and will depend upon many factors. Even if the wiping efficiency is determined under certain well-defined conditions, the chemical and physical boundary conditions during taking of wipe tests in real measurement environments may deviate from the idealised conditions. Assumptions on the variation of the deviation between idealised and real wiping efficiency have to be made and need to be included in the analysis of type B uncertainties for measurements with wipe tests only.

Treatment of uncertainty related to the content of natural radionuclides and other radionuclides to be disregarded in the measured material

4.86. Natural radionuclides can be present in the measured material, in particular in building rubble, where radionuclides of the ^{238}U- and ^{232}Th- decay chains as well as ^{40}K contribute to some extent to the measurement effect, in particular for gross gamma measurements (bulk monitors) and for measurements with surface contamination monitors, while this effect is less important for measurements with in situ or laboratory gamma spectrometry. As the radionuclides of natural origin were not part of the practice giving rise to the material to be cleared, they can be disregarded, and therefore their contribution to the measurement effect can be subtracted from the gross measurement effect. The activity of natural radionuclides will have to be determined in advance from a reasonable set of samples. However, the activity of natural radionuclides in real measurements may deviate from this previously determined value. Hence, this difference has to be determined and needs to be included in the analysis of type B uncertainties for gross gamma measurements and surface contamination monitor measurements on building rubble or on building surfaces.

4.87. The same consideration applies to other radionuclides in the material that are to be disregarded, e.g. ^{137}Cs from the "fallout".

Treatment of uncertainty during measurements as part of the calibration

4.88. Finally, the measurements performed as part of the calibration itself will also be associated with uncertainties. Examples are uncertainties in the real activity content of calibration standards (even after adjustment for radioactive decay), readout of instruments or determination of distances. However, these uncertainties can mostly be neglected in comparison to those described in the previous sections.

Treatment of other uncertainties

4.89. While the list of uncertainties in the previous sections is comprehensive with regard to clearance measurements, there may be other uncertainties that need to be taken into account in specific situations. Safety Report Series 67 [12] provides practical guidance and examples related to treatment of other uncertainties for decisions on clearance, such as those related to sampling.

ASPECTS RELATED TO USE OF MIXING AND DILUTION AS PART OF THE MATERIAL MANAGEMENT PROCESS

4.90. Deliberate dilution of material to meet the clearance levels, as opposed to the dilution that takes place in normal operations when radioactivity is not a consideration, should not be performed without the prior approval of the regulatory body.

4.91. The regulatory body should ensure that dilution is not used to clear relatively high specific activity materials by deliberately diluting them in order to meet clearance levels. Records should be kept of the dismantling operations in order to demonstrate that such materials are kept separate. Clearance should be carried out as the material arises.

4.92. Unavoidable mixing may occur, and is acceptable, where the extent of mixing is consequent on the operation or decommissioning technique employed. For example, the use of an excavator to dig out a volume of contaminated soil may result in some unavoidable mixing of soil with differing levels of contamination. In this case this is considered to be mixing as part of the material management process.

4.93. 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.

4.94. If it is necessary to reduce the uncertainty of the measurement result, it is acceptable to bulk two or more similar measurement units (e.g. drums) to produce a larger measurement unit. This has to be done after the initial measurement. This is not dilution as the purpose is solely to reduce the measurement uncertainty, not to alter the apparent characteristics of the waste or material. Similarly, two small samples of material could be put together.

4.95. In the case of conditional clearance, mixing with clean material can be part of the condition (e.g. melting of metals in a non-nuclear industrial melting facility). In this case, the average mixing ratio should be documented and approved by the regulatory body prior to implementation in the clearance process, as part of the traceability of the clearance process for this material.

5: CLEARANCE OF LIQUID MATERIAL

APPLICATION OF THE CONCEPTS OF DISCHARGE AND CLEARANCE OF LIQUIDS

5.1. Liquid effluents from nuclear facilities or from the use of radionuclides in medicine, industry and research are usually treated as discharges according to Requirement 31: Radioactive waste and discharges of GSR Part 3 [1]. Discharges require a licence or authorization. The dose criterion applying to liquid

discharges is generally chosen in the range between 0.1 mSv/a to 0.3 mSv/a [28], which is a fraction of the dose limit to members of the general public. Using radiological models like those recommended in Safety Report 19 [29], this dose criterion is converted into limits for annual discharge of single radionuclides or radionuclide groups, usually expressed in Bq/a. These limits are specific for a certain facility or a certain type of facility. More details are given in Safety Guide WS-G-2.3 [30].

5.2. A dose criterion of 300 µSv/year for any member of the public is usually considered as a basis for the values calculated for discharges of (non-hazardous) aqueous liquids or effluents. Such liquids are subject to direct discharge to sewer systems or various water bodies and not to clearance.

5.3. There are situations where discharge of liquids contaminated with radionuclides is not a relevant concept and therefore these liquids have to be released from radiological regulatory control in a different way. Examples are situations where the facility in which the liquids arise does not possess a licence or authorisation for discharging liquids or where the liquids are not suitable for discharge into the environment. There may also be cases where liquids constitute an asset and where there is commercial interest in reuse or recycling, e.g. in the case of lubrication oils used in pumps, cooling liquids in transformers in nuclear power plants or acids from the manufacturing process of nuclear fuel. Likewise, it may be beneficial to incinerate certain liquid chemicals used in industry, medicine or research in a conventional waste incineration plant because of hazardous chemical properties. In all such cases, it is not possible to treat the release of the materials as discharges, but instead the concept of clearance can be applied. For clearance of liquids, the basic principles given in Sections 2 and 3 of this Guide apply as for solid materials.

5.4. Clearance of liquids needs to be treated on the basis of the same dose criterion as clearance of solid material, i.e. individual effective doses on the order of 10 µSv per year. The fundamental difference between clearing and discharging liquids is that discharges, once released to the environment, in most of the cases remain dispersed, (i.e. the activity cannot be concentrated again by any process), while cleared liquids may remain together, so that after clearance the activity concentration in them may be significantly increased by filtration or by concentration processes, such as evaporation, distillation or fractionation. The activity concentration present at the time of clearance may therefore be much smaller than at any later time. This needs to be taken into account appropriately in the derivation of clearance levels.

ASPECTS OF LIQUID MATERIALS DETERMINING THE CLEARANCE OPTION

5.5. Liquid materials, in particular aqueous liquids, have some properties that distinguish them significantly from solid materials with respect to application of the principle of clearance. These are:

a. Aqueous liquids can be easily concentrated (by evaporation or distillation) so that the initial concentration of radionuclides in the liquid can change. Concentration processes will increase the radionuclide concentration in the liquid, if the radionuclides in question stay in the liquid phase during such a process.

b. Radionuclides can evaporate from aqueous liquids and can thus become a source of contamination.

c. Radionuclides can be accumulated on filters during filtration processes.

5.6. Non-aqueous liquids like oils, lubricants, antifreeze agents or other organic substances do not show such properties or only to a much lesser extent. This means that a given concentration in such liquids will remain constant or will decrease, but not increase, through subsequent steps of treatment. In this way, such liquids show properties that are similar to those of solid materials.

5.7. Furthermore, liquids containing contamination in the form of radionuclides bound to suspended particles can be purified by filtration processes, whereby, however, the activity in the filtrate accumulates. This may be the case for lubricants in which abraded particles that possibly bear some contamination accumulate.

5.8. Several IAEA Member States have therefore chosen to limit regulations for clearance of liquid materials to those types of liquids for which the likelihood of any processes leading to an increase of activity concentration is very small or negligible, and which have been filtered prior to clearance. In this case it is avoided to deal with concentration processes in the derivation of generic clearance levels.

NATURE AND SCOPE OF CLEARANCE REGULATIONS FOR LIQUIDS

5.9. If the application of the concept of clearance for liquids is limited to non-aqueous liquids, for which no inadvertent concentration processes have to be expected, this clearance is of the specific clearance type, as it is limited to certain types of materials, and cannot be termed “unconditional”. In such a case, additional limitations, e.g. with regard to the destination of the liquid material, can apply. The following options can be distinguished:

a. The liquids are cleared for any purpose, i.e. they can be directly reused, recycled or disposed of (e.g. by incineration). This may be the case for oil or lubricants after filtration, which can be directly reused, recycled (by converting it into fuel or used for energy recovery) or disposed of (by incineration in a waste incineration plant).

b. The liquids are cleared for a specific process only, e.g. for disposal by incineration in a conventional waste incineration plant.

5.10. In any case, the clearance is a type of specific clearance. In addition, case-by-case decisions are of considerable importance for the release of liquids, in particular when aqueous liquids like dilute acids that have been used in certain processes in nuclear facilities (like hydrogen fluoride HF in fuel element manufacturing) are to be cleared for further use in the chemical industry. In this case, the first use and any possibilities for concentration processes (e.g. when instead of a dilute acid a strong acid will be required) have to be taken into account. Radiological models describing such types of specific clearance need to include possible processes of concentration, filtration and in general all changes of the activity concentration in the liquid that are conceivable in the process, including those in water purification plants where many chemical elements are extracted from the water and concentrated in sewage sludge.

5.11. Like for solid materials, specific or conditional clearance of liquid materials requires that the conditions attached to the clearance process are being fulfilled, e.g. that the liquids are filtered before release, that they are brought to a specified first use or a specified recipient or that limitations of total or annual quantities are respected.

5.12. Finally, cleared aqueous liquids can also be discharged into a receiving water (lake, river, sea). As the liquid has been cleared, no authorization for the discharge from the nuclear regulatory body would be needed (while the approval of the water authorities would still be necessary). In such a case, the model used for describing the radiological consequences of this type of clearance needs to take into account all relevant pathways in the environment, i.e. migration of radionuclides in the water body, sedimentation oruse of water for radioecological pathways, as described in IAEA Safety Report 19 [29]. Special consideration should be given to ^{3}H as the concentration of this radionuclide cannot be increased by natural processes in liquids, sediments, plants or animals.

PRACTICAL APPLICATION OF THE CLEARANCE CONCEPT TO LIQUID MATERIALS

5.13. Clearance of liquid materials may give rise to similar exposure pathways as clearance of solid materials, i.e. external irradiation, inhalation, direct ingestion and secondary ingestion. The details of a radiological model that is specifically designed for liquids from medicine, industry and research and that covers all relevant exposure pathways and exposure scenarios has been given in TECDOC-1000 [23], which includes guidance on the practical application of the concept of clearance to liquids for release into the environment. The values in Table IV of TECDOC-1000 [23] were derived with the intention of assuring that if complied with, annual doses to individual members of the public arising from any single cleared practice will not exceed 10 µSv. These values are expressed in Bq/a and can be converted into limits for volume-related concentrations (Bq/m³ or Bq/l) if the annual amount of effluents is known. Compliance with these levels (or with similar levels derived on the basis of the clearance criteria in GSR

Part 3 [1]) will not require further monitoring or institutional control of the release as would be the case with discharges.

5.14. In addition, the clearance levels provided in Table I.2 of GSR Part 3 [1] may also serve as the basis for clearance of liquids, provided that concentration or filtration processes may not occur with the cleared liquids. The reason is that the scenarios of the radiological model underlying these clearance levels [3] cover various exposure situations for solid materials that would also be bounding for reuse, recycling or disposal of liquid materials, e.g. storage in a large tank giving rise to external gamma irradiation, evaporation of the liquid leading to inhalation, groundwater pathways. The only option that is not covered by the radiological model underlying these clearance levels [3] is release of large quantities of liquids into the environment. This means that the clearance levels provided in Table I.2 of GSR Part 3 [1] could be applied for clearance of non-aqueous liquids, such as oils and lubricants, for reuse, recycling or disposal by incineration.

5.15. A further example for practical regulations on clearance of liquids is the report [31] (BEIS-2018), which introduces the notion of "relevant liquid" covering non-aqueous liquids, and certain types of aqueous liquid with specified hazardous properties to the water environment. 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 for disposal as waste (usually via disposal in a waste incineration plant) is provided in Annex III to this document.

5.16. Characterization of liquids for clearance is based on the general principles and requirements described under the section on "Characterisation of the material to be cleared" (para 3.14-3.32). Special attention should be given to the homogeneity of the liquid and the possibility of deposition of sediments. In case of measurement by sampling, the samples should be taken in compliance with monitoring and sampling standards to assure their representativeness.

CLEARANCE LEVELS IN TERMS OF SPECIFIC ACTIVITY OR TOTAL ACTIVITIY

5.17. Like for solid materials, clearance levels for liquids will usually be expressed in terms of activity concentration for each radionuclide. While for solids the activity is related to the mass (Bq/g), in the case of liquids the activity is related to the volume (e.g. Bq/l). For example, the activity concentration defines the dose rates for gamma emitting radionuclides and the activity ingested with a given quantity of liquid material.

5.18. For certain clearance options, mainly for cases where liquids or their residues can accumulate in certain places, like in some case-by-case decisions, it may be necessary to limit the total activity for each radionuclide or of some radionuclide groups over time (e.g. in Bq/a), in addition to or instead of providing volume related clearance levels. Examples for such an approach are TECDOC-1000 [23], where clearance levels in Bq/a were provided for liquid releases treated with the concept of clearance (not as discharges), as well as the report [32] (HPA CRCE005).

DILUTION

5.19. Concerning the aspect of dilution in the context of clearance of liquid materials, it needs to be carefully distinguished whether dilution would take place before or after clearance. As for solid materials, deliberate dilution of the liquid material with clean material (e.g. uncontaminated water) to reach the clearance levels prior to release of material from regulatory control is not an acceptable practice, unless a permission is obtained from the regulatory body for such an action. Dilution of the cleared liquids after the act of release will occur at many subsequent stages and may be taken into account in the radiological models. However, there is a possibility of concentration in sediment downstream and in some industrial uses, and these situations need consideration.

NATURAL BACKGROUND

5.20. Like for clearance of solid materials, clearance of liquid materials will not be concerned with natural radionuclides in the liquid that do not originate from the practice in question. Examples for such background contamination may be radionuclides of the natural decay chains of Uranium and Thorium as well as potassium in water in appropriate chemical form (e.g. as U or Th oxides and complexes, potassium iodide or iodate). The activity values of such radionuclides that can be attributed to the background level may be disregarded in the clearance process, i.e. their contribution to the measurement effect may be neglected during measurements in the clearance process.

6: CLEARANCE OF GASEOUS MATERIAL

APPLICATION OF THE CONCEPTS OF DISCHARGE AND CLEARANCE OF GASES

6.1. Gases originating from nuclear facilities are usually treated as discharges according to Requirement 31: “Radioactive waste and discharges” of GSR Part 3 [1]. Clearance levels for gases that are to be discharged can be calculated on the basis of individual effective doses on the order of 10 µSv per year. Gases that meet such clearance levels can be discharged without any regulatory authorization.

6.2. Unlike for liquids, it is highly unlikely that gases once used in a nuclear facilities or a facility applying radionuclides in medicine, industry or research will constitute an asset for which reuse or recycling could be envisaged.

6.3. If, for some reason, it is nevertheless needed to apply the concept of clearance to reuse, recycling or disposal of gases, then the radiological analysis needs to take into account the possibility that the concentration of radionuclides in the gas is highly dependent on the volume in which the gas is present. This may change over orders of magnitude. Exposure scenarios relevant to a compressed gas may therefore be fundamentally different to those for a gas under standard conditions.

PRACTICAL APPLICATION OF CLEARANCE CONCEPT TO GASEOUS MATERIALS

6.4. The application of the clearance levels provided in Table I.2 of GSR Part 3 [1] or any other clearance levels derived for solid or liquid materials to the clearance of gases is not permissible for the reasons given above.

6.5. Guidance on the practical application of the concept of clearance to gases for release to the environment has been given in TECDOC-1000 [23]. The values in Table III of TECDOC-1000 [23] were derived with the intention of assuring that if complied with, annual doses to individual members of the public arising from any single cleared practice will not exceed 10 µSv. These values are expressed in Bq/a and can be converted into limits for volume-related concentrations (Bq/m^3) if the annual amount of effluents is known. Compliance with these levels (or with similar levels derived on the basis of the clearance criteria in GSR Part 3) will not require further monitoring or institutional control of the release as would be the case with discharges.

6.6. Sampling and characterization of gases for clearance purposes should be in compliance with monitoring and sampling standards.

7. CONCEPT OF CONDITIONAL CLEARANCE

RADIOLOGICAL BASIS

7.1. The concept of conditional clearance[8] is introduced in GSR part 3 Schedule 1 section I.13. The radiological basis for conditional clearance is the same as for clearance, namely those specified in Schedule 1, sections I.10 and I.11 and these are reproduced here:

I.10. 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.

I.11. Material may be cleared without further consideration under the terms of para. I.10(a) provided that in reasonably foreseeable circumstances the effective dose expected to be incurred by any individual owing to the cleared material is of the order of 10 μSv or less in a year. To take into account low probability scenarios, a different criterion can be used, namely that the effective dose expected to be incurred by any individual for such low probability scenarios does not exceed 1 mSv in a year.

CONDITIONAL CLEARANCE AS ADDITIONAL OPTION FOR MANAGEMENT OF MATERIAL

7.2. Conditional clearance is applied to a **particular material, for a specified amount** and to **a particular fate and destination** of that material: the type of material and **the amount**, fate and destination are therefore specified as the conditions attached to the conditional clearance. Examples of conditional clearance that have been considered in Member States are scrap metal for recycling (melting), buildings for demolition, and waste for disposal in landfill sites.

7.3. Conditional clearance is an additional option for management of material and waste, which provides for more flexibility in management of material/waste from authorized facilities and activities. It is essentially a form of application of the graded approach to regulatory control of materials and waste,

[8] The term used in the GSR Part 3 is "specific clearance". However, in this Safety Guide the well-established term "conditional clearance" is used with the same meaning, in order to avoid a need for constructions such as "radionuclide specific surface specific specific clearance levels".

and supports the application of "the waste hierarchy"[9], enabling reduction of amounts of waste to be managed as radioactive waste and increase of amounts to be reused/recycled or disposed of as non-radioactive waste. Conditional clearance can also be related to the criterion for clearance specified in Schedule I paragraph I.10 (b), whereby the regulatory body may decide (in exceptional cases, where the national regulatory framework so allows) that the optimum regulatory option is to remove a particular material from regulatory control.

CONDITIONAL CLEARANCE LEVELS

7.4. Following on from the concept of conditional clearance, introduced in paras 7.1-7.2, it is possible to derive conditional clearance levels in terms of activity concentration per unit mass, or activity concentration per unit surface area, using an appropriate set of scenarios. These conditional clearance levels ensure that the dose criteria for clearance are met for the specified material and the specified fate and destination. These conditional clearance levels would be expected to be higher or the same as the clearance levels specified in Schedule I Table I.2 and Table I.3 in GSR Part 3 [1] because they consider a particular tailored set of scenarios rather than the general scenarios that were considered for the clearance levels. The conditional clearance levels calculated for a specific set of materials and/or destinations would not be applicable to other materials or destinations.

7.5. Problems could occur if the conditional clearance levels are such that the conditionally cleared material (e.g. metals for melting) would require notification or authorisation upon receipt at the specified destination (e.g. smelter). In order to avoid such legal and regulatory problems it is recommended that the mass specific conditional clearance level does not exceed the corresponding exemption level for moderate quantities specified in Schedule I Table I.1 in GSR Part 3. In this way the conditionally cleared material will be below the mass specific exemption level for moderate quantities and therefore can be exempt from the requirement for notification. If the specified destination is an authorised practice, e.g. a licensed smelter, then these considerations are not relevant, conditional clearance is not an appropriate concept to be applied, and such material should be treated as radioactive.

7.6. It should be noted that during the metal melting process certain nuclides concentrate in the dusts and slags so that the activity concentration in these by-products may exceed the activity concentration in the metals, and hence exceed the exemption levels for moderate quantities [1]. The radiological

[9] The concept of "the waste hierarchy" is widely accepted to be fundamental to the sustainable management of all types of wastes, including radioactive wastes. The concept of the waste hierarchy has been widely adopted in national policies and has also been taken up internationally (e.g., EU, UNEP, OECD).

assessment used to derive the conditional clearance levels will include scenarios that account for this phenomenon, and therefore this will ensure that the doses from exposure to such dusts and slags do not exceed a value on the order of 10 μSv per year. The GSR Part 3 automatically exempts such material, so that reporting and authorization in such cases would not be necessary.

SURFACE SPECIFIC CONDITIONAL CLEARANCE LEVELS

7.7. Surface-specific clearance levels for unconditional clearance need to be carefully distinguished from surface-specific clearance levels for conditional clearance options. Surface specific levels for conditional clearance may be derived for the following options:

- clearance of metals for melting;
- clearance of buildings for reuse;
- clearance of buildings for demolition.

7.8. In these options for conditional clearance, surface-specific clearance levels fulfil different purposes. For example, limitation of the surface activity on metallic items protects people handling the material prior to melting. Limitation of surface activity on building surfaces will protect people using the room as a new workplace from high concentrations of activity on the surfaces, leading to increased levels of direct irradiation. Likewise, this will protect people refurbishing the room from high concentrations of resuspended activity in the breathing air. Examples for surface-specific clearance levels for metal scrap for melting can be found in recommendation RP 89 [21] and for buildings for demolition in recommendation RP 113 [33] of the European Commission.

7.9. Clearance on the basis of surface specific clearance levels generally only applies to surfaces where the contaminant can be detected by the surface measurement technique, and the depth of the contaminant is such that the measurement technique sees, to a reasonable degree, all the contamination. Surface specific clearance levels are not relevant to excavated soil or building rubble.

7.10. When surface-specific clearance levels are given for surfaces where the activity can penetrate into the volume, like for building surfaces and unsealed ground, it needs to be specified whether the clearance levels apply only to the top layer (i.e. the actual surface) or to the surface and a part of the volume beneath this surface. Usually it is a prudent approach to relate the surface-specific clearance levels to the sum of contamination present directly on the surface and inside the volume beneath the same surface area. Surface-specific clearance levels can then be understood as a limitation for the activity beneath the surface projected onto the surface area. Details are given in recommendation RP 113 [33] of the European Commission.

7.11. In addition, careful distinction needs to be made between surface-specific clearance levels for unconditional clearance on the one hand and other limitations for the surface-specific activity, such as activity limits in transport. The Regulations for the Safe Transport of Radioactive Material SSR-6 [22] provide surface-related values of 0.4 Bq/cm² for beta and gamma emitters and low toxicity alpha emitters and 0.04 Bq/cm² for all other alpha emitters in the definition of contamination. Furthermore, surface contaminated objects (SCO-I) are defined in SSR-6 as items where the removable surface contamination does not exceed 4 Bq/cm² for beta and gamma emitters and low toxicity alpha emitters and 0.4 Bq/cm² for all other alpha emitters, with additional criteria applying for the fixed surface contamination. It should be noted that these values have been derived from a radiological model [34], which was not relevant for clearance. A more recent endeavour to update those values in a Coordinated Research Project of IAEA [35] provides up-to-date radiological modelling for exposure from surface contamination on various types of packages for radioactive waste and spent fuel. However, this model is also unsuited for application to clearance.

7.12. Radiological models for the derivation of surface-specific clearance levels need to take account of all exposure pathways that can be caused by presence of surface contamination. In particular, these include:

- external irradiation from the contaminated surface,
- ingestion as a consequence of hand-to-mouth pathways when handling such objects and transferring part of the contamination to the hands,
- inhalation as a consequence of resuspension of the contamination into the breathing air when handling or machining such objects,
- skin contamination as a consequence of transfer of part of the contamination onto uncovered parts of the skin.

7.13. The dose criteria for clearance, specified in paragraphs 1.10-1.12 of GSR Part 3, should be applied when using these radiological models for derivation of surface specific clearance levels.

7.14. If material has been activated by particles (e.g., concrete structure of the reactor's biological shield), and there is no surface contamination present, surface specific clearance levels are not applicable.

MEETING THE CONDITIONS

7.15. It is clear that under conditional clearance, the condition(s) attached to the fate or destination of the material need to be achieved in order to consider the clearance process to be complete. Metal scrap that

was cleared on the condition that it was melted needs to actually reach a furnace and be melted there, and not be reused before that point. In case mixing is required with non-radiological metal as part of the condition, the mixing ratio used in the derivation of the conditional clearance levels should be respected. Likewise, a building that was cleared on the condition that it would be demolished must not be used in the meantime for new workplaces (e.g. as an office building or a workshop) but must be demolished without prior reuse.

7.16. Hence, it is necessary to establish a form of contract or arrangement between the site operator (consignor) and the operator of the final destination to ensure that the conditions are met. The practicalities of this will need to be agreed with the regulatory body. This could include overseeing the transport to the specified destination or requiring receipts to be sent to the consignor that can be reviewed by the regulatory body.

7.17. Conditional clearance can therefore be considered as a two-stage process. Stage 1 is the act of clearance when it is confirmed that a) the material meets the conditional clearance levels, b) the fate or destination is agreed, and c) a specific contract is in place for that material. Stage 2, confirmation, occurs when evidence is provided that the conditions attached to the conditional clearance have been met.

7.18. In the case of conditional clearance of scrap metal for melting, the process of dealing with scrap metal in the Member State 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 is not appropriate for conditionally cleared scrap metal. Therefore, conditions should ensure that conditionally cleared metal for melting is sent directly to the specified melting facility.

7.19. Similarly, in the case of conditional clearance of material sent to a landfill, the specificities of the landfill have to be understood and included in the scenarios taken into considerations for derivation of the corresponding levels. The conditions that are specified should take account of the capacity of the receiving landfill as well as the mass specific activity levels. Also the post closure period should be considered in the scenarios. Possible intrusion scenarios after the end of the institutional control period should be treated as low probability scenarios, whereby exposure does not exceed the 1 mSv/a criterion.

7.20. Any facility that receives conditionally cleared material does not require a license during operation nor after closure since the material it receives is not radioactive in a legal sense (compare with para 2.11).

7.21. In the concept of conditional clearance it is important to specify when the process of clearance may be considered to be finished, so that the residual activity of the material can be disregarded in a legal sense (compare with para. 2.11). While in the case of unconditional clearance the process of clearance may be

considered complete once compliance with clearance levels for unconditional clearance has been established, conditional clearance requires the material to reach a certain destination or end state (e.g. metal cleared for melting must reach the smelter, waste cleared for disposal must reach the landfill, buildings cleared for demolition without prior reuse must be demolished). In such cases the question often arises whether transport of the material to its destination, which is necessary to complete the clearance process, will require a license according to the Transport Regulations, and whether handling the material during this time will require an authorisation, permit or license (if so, it would be necessary that the transport is performed by a licensed shipping company).

7.22. The radiological models used for derivation of clearance levels for conditional clearance options explicitly take into account transport processes. Typical clearance levels for conditional clearance options are orders of magnitude lower than the Package Contents Limits Q_A – Q_E that is one of the technical bases of the Regulations for the Safe Transport of Radioactive Material SSG-26 [36]. Therefore, the regulatory body should make provisions that any steps that lie between the dispatch of the material from the facility where clearance measurements have taken place and its final destination do not require any kind of authorisation. During these steps, the material should be regarded from a radiation protection point of view as if the clearance process was already finished. This also implies that any containers used for transport of such material or the material itself should not be placarded or bear any label or sign designating it as radioactive.

7.23. Furthermore, concern is often raised that e.g. in a traffic accident involving a truck with such material or in the case of loss or theft of part of the load, material with activities above exemption values of Table I.1 of GSR Part 3 might be found by members of the public.

7.24. Traffic accidents, as well as theft or loss of part of the conditionally cleared material are generally not associated with radiological consequences of any concern, as this is also covered by the Transport Q System mentioned above. Furthermore, traffic accidents, theft or loss can be considered as examples of low-probability events in radiological models for conditional clearance options. Therefore, the regulatory body should not require any authorization for transport of conditionally cleared material.

7.25. If it is not possible to put in place an arrangement to ensure that the conditions associated with the conditional clearance of the particular material are met, to the satisfaction of the regulatory body, then it should be concluded that conditional clearance is not suitable for that particular material and should not be used. This is to ensure that GSR Part 3 section 3.12 is met, so *"that sources that have been cleared from regulatory control do not again become subject to the requirements for notification, registration or licensing…"*.

7.26. Hence, conditional clearance should be considered to be finished when the material has reached the final destination specified in the conditions. However, the regulator may decide that the conditional clearance is already finished when the material leaves the facility if they are confident that all the conditions will be met.

8. INVOLVEMENT OF INTERESTED PARTIES AND ENHANCING PUBLIC UNDERSTANDING

8.1. Clearance is a regulated process that is safe and in accordance with the IAEA GSR Part 3 [1]. It is defined as the release from radiological regulatory control of material that poses a trivial level of risk to people and the environment, irrespective of its future use. Hence, clearance involves the release of material arising within a radiation regulated industry, e.g. nuclear industry, to a destination that is not part of a radiation regulated industry. Cleared material will most likely be processed or used by people who are not familiar with radiation protection and who do not necessarily understand the concept of radiation risk, nor equate the dose criterion of 10 microSv per year with a trivial level of risk to people and the environment. Also, people who use the cleared material without taking any particular radiation protection measures may not understand that they are implicitly protected by the application of the clearance levels (because the scenarios used to derive the clearance levels assume that the material is used by people who are unaware of the origin of the cleared material and therefore do not apply any particular radiation protection measures).

8.2. Therefore, authorized parties and regulatory bodies should engage with interested parties including identified potential stakeholders to discuss the various aspects of clearance, including the social, economic and environmental benefits of clearance by increasing recycling, the derivation of the clearance levels, application of the concepts of clearance, the national framework for clearance and the approach to demonstrating compliance with the clearance levels. To build up confidence in the clearance process, this engagement should be carried out using clear terminology to avoid ambiguities, it should be carried out in a transparent manner, and it should take different forms depending on the interested parties. Examples of different forms of communication are a formal consultation or communication on the national framework; discussions between regulators, authorized parties and waste management organisations; seminars and workshops with interested parties; printed material including leaflets; and the use of electronic media such as web pages and social media.

8.3. The aim of the engagement is not only to understand the concerns of the interested parties and to address them with respect and in a proportionate manner, but also to share the social, economic and environmental benefit obtained from the cleared materials through recycling and a more sustainable use

of resources. Communication should be maintained in order to develop a common understanding, based on trust, of the concept of clearance with stakeholders. There is no need for regulators to require the authorized party to apply an excessively conservative clearance process simply in order to gain public acceptance. The regulators and authorized parties should cooperate with each other in order to pursue the social, economic and environmental benefits of clearance.

8.4. Demonstration of the clearance procedures undertaken and the measurements that are made as part of the clearance process can be effective in enhancing understanding by stakeholders, and, in some cases, may be sufficient to build confidence in the application of clearance.

8.5. One approach that is useful in enhancing public understanding of the trivial radiation risk from cleared materials is to compare the radiation risk from the cleared material with the average lifetime background cancer risks in the member state, and with the variation in these background cancer risks in the different regions in the member state. This comparison of risks should use the radiation risk coefficient of 5% per Sv defined by ICRP2007 [37]. Comparisons of the trivial risk from cleared material with commonly accepted radiation risks, e.g., intercontinental flights, natural radionuclides in foodstuffs, are also useful communication tools. Relevant information for these comparisons can be found in IAEA posters and leaflets about radiation protection [38].

8.6. Experiences in dialogue forums held in affected areas just after Fukushima Dai-ichi nuclear accident, found that an effective way to enhance public understanding of radiation risk was to compare the radiation risk with the variation of lifetime background cancer risks in 47 prefectures. Using the LNT (Linear Non-Threshold) model and a risk coefficient of 5% per Sv, radiation exposures of 20 mSv and 1 mSv will lead to an increase in risk of 0.1% and 0.005%, respectively. This means that the national average lifetime cancer risk will increase from 25.4% to 25.5% and 25.405%, respectively. On the other hand, the lifetime cancer risk in daily life without additional exposure varies between 23.7% and 28.3% among the prefectures of Japan due to differences in lifestyle such as diet [39] (Ogino and Hattori). An example of the material used for public communication is given in Fig. 8.1.

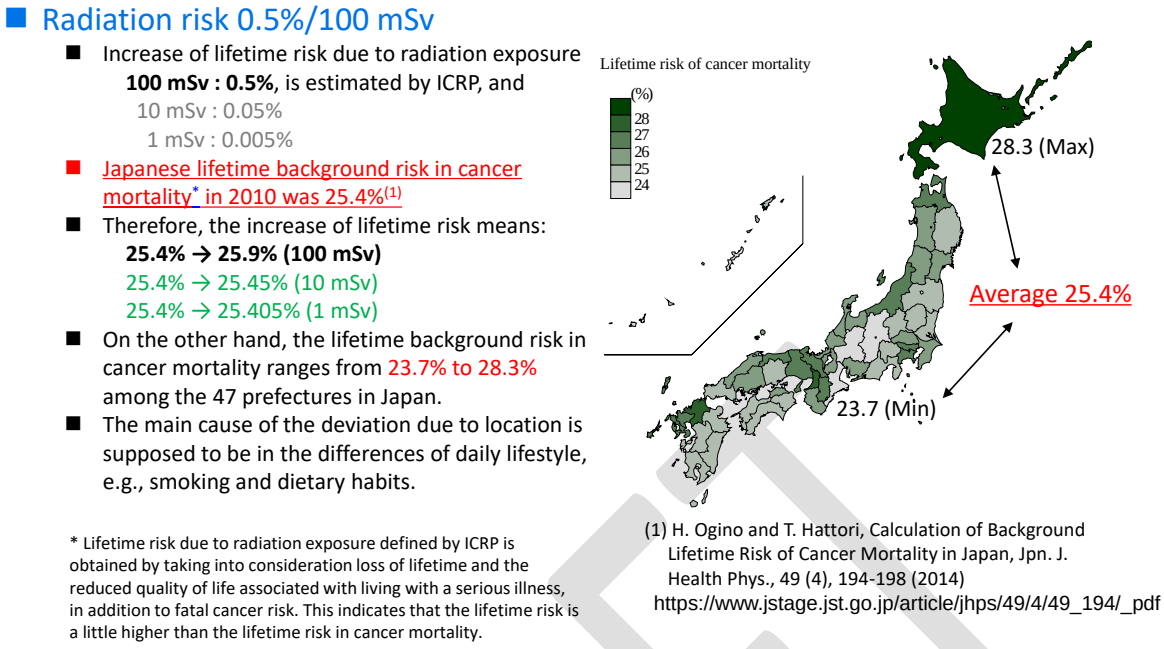

Fig. 8.1 Comparison between radiation risk and variation of lifetime cancer risk [40].

8.7. In the case of conditional clearance, where the fate and destination of the material is specified in the conditions, the authorized party proposing the conditional clearance should engage with the operator of the final destination so that the conditional clearance option is founded on an agreement and understanding between the authorized party clearing the material and the final destination. Other interested parties should also be consulted, e.g. transport operators, and the regulatory bodies. Since conditional clearance levels are normally higher than generic clearance levels (unconditional clearance levels), the regulators and authorized parties should carefully explain the difference between them to the interested parties in an easy-to-understand manner.

8.8. An important point to communicate to interested parties is that clearance is a safe and regulated process, it is allowed by law and the cleared material is no longer required to be regulated from a radiation or radioactivity point of view. Therefore waste management organisations that send cleared material to other destinations with no radiation marking or no reference to the radiation regulatory regime are not breaking the law. In fact, they are abiding by the law.

8.9. The last decade has seen an increased focus on the importance of involvement of interested parties, including the public, in a number of policy areas, particularly those concerned with environmental issues or technology evaluation. There are a number of arguments for involving interested parties in setting up the clearance process, as described in [41] (NW-T-2.5).

APPENDIX

SCREENING LEVELS FOR RECYCLING OR DISPOSAL ON LANDFILLS OF MATERIAL AND WASTE AFTER AN EMERGENCY

GENERAL

A.1. Nuclear or radiological emergency may continue for a long time. After the early and intermediate phases of the emergency, a next phase will come to manage recovery phase of the affected people and area under a regulatory system in association with the radiological protection in the existing exposure situation.

A.2. In the post-emergency period, under an existing exposure situation, the reference level for the optimisation of protection of people living in the affected areas is selected from the 1–20 mSv/year band.

A.3. Using the reference level, the regulatory body may need to set up a new regulatory system for the material and waste management in the affected area, e.g. for disaster waste, rubbish after cleaning homes up, paddy straw, and soil and waste generated from decontamination work. According to the regulatory system, some highly contaminated material and waste may be put under the regulatory control.

A.4. Due to radioactive decay, there is a possibility that the activity concentration of the material or waste that has been designated for need of regulatory control may become lower than the regulatory value. In this case, if necessary, recycle or disposal on landfills of the material and waste could be allowed, without further regulatory control in some cases, in the post-accident existing exposure situation, which is similar to a concept of conditional clearance in a planned exposure situation.

RELATIONSHIP TO CLEARANCE IN THE PLANNED EXPOSURE SITUATION

A.5. In the planned exposure situation, the term of “clearance” is usually used for the release of material and waste from regulatory control. The primary radiological basis for establishing values of activity concentration for clearance is that the effective doses to individuals should be of the order of 10 μSv or less in a year. To take account of the occurrence of low probability events leading to higher radiation exposures, an additional criterion is used, namely, the effective doses due to such low probability events should not exceed 1 mSv in a year. In this case, consideration was also given to doses to the skin; an equivalent dose criterion of 50 mSv in a year to the skin was used for this purpose.

A.6. In existing exposure situations, the concept of reference levels should be used for a protection strategy in conjunction with the implementation of the optimisation process for exposures. They should

be used as tools for optimization in defining, selecting, analysing or benchmarking a certain protection strategy.

A.7. Recycling of material or disposal of waste on landfills in post-emergency existing exposure situation often cannot be done using the same dose criteria as for clearance in planned exposure situation, due to the increased background radiation. In such cases different dose criteria may be selected, which is more appropriate, and which takes into account specificities of the existing exposure situation. This criterion is not the same as the reference level for the remediation actions. Considering such a difference in the radiological basis used for clearance in planned exposure situation and for recycling or disposal on landfills under an existing exposure situation, the term "clearance level" for limiting concentrations of radionuclides in materials should not be used for the later, as it may create confusion in the public. This Appendix instead uses the term "screening level" for the operational values used in measurements.

A.8. If a clearance-like process in such situations is necessary, any derived screening level should be based on an underlying, individual effective dose criterion whose numerical value is smaller than or equal to the selected reference level for the existing exposure situation under consideration. In such cases a value of the order of 1 mSv/y or less is recommended for such dose criterion, considering the band of reference levels for existing exposure situations and adhering to the general criteria for clearance as specified in Schedule I, para I.10 (a) and (b), and in para I.11-I.13 of GSR Part 3 [1], below which no further optimisation or protective actions may be necessary. The basis for selecting this value of annual dose is considering the coincidence with the dose criteria applied for clearance on the basis of low-probability scenarios for artificial radionuclides and the dose criteria for clearance of residues with radionuclides of natural origin, where no further protective actions may be necessary as they would yield no net benefit. Hence, for practical application to support decision making in a clearance-like process, an approach of using "screening levels" of measurable quantities (in Bq/g), derived from above mentioned dose criterion, is recommended.

CATEGORIZATION OF RECYCLE OR DISPOSAL ON LANDFILLS

A.9. When a nuclear or radiological emergency happens, in the affected area there is usually a facility containing nuclear or radioactive materials. It may even be the cause of the emergency. That facility has been already regulated by the relevant radiological regulatory system before the emergency. Hereinafter, the facility in the affected area is referred to as on-site. On the other hand, there may also be contamination in the broad area outside the facility due to the emergency. Hereinafter, the affected area except for the facility is referred to as off-site. In the off-site, it may be necessary for the regulatory body to set up a new

regulatory system for the material contaminated as a result of an emergency and waste clean-up management because there has been no radiological regulatory system since before the accident.

A.10. When we consider the recycle or disposal on landfills of the material and waste in the post-accident existing exposure situation, a distinction should be made between on-site and off-site in locations of the origin of material or waste and the target location of recycle or disposal on landfills, because the applicable regulatory systems are different. In this sense, according to the locations of the origin and the target of the material and waste, the possible ways of recycle or disposal on landfills would be categorized into three types; category 1: from off-site to off-site; category 2: from on-site to on-site and category 3: from on-site to off-site.

A.11. After the Fukushima Dai-ichi Nuclear Power Station (1F plant) accident, as for the category 1, the Ministry of the Environment (MOE) of Japan, which is responsible for regulating off-site contamination, developed activity concentration for recycling of the removed soil in 2016. This is a good example for recycling of the material and the waste generated from off-site, but is not the case of a clearance, because this is a recycle under the Act on Special Measures concerning the Handling of Environment Pollution by Radioactive Materials Discharged by the NPS Accident Associated with the Tohoku District - Off the Pacific Ocean Earthquake That Occurred on March 11, 2011 (Act on Special Measures, issued by MOE).

A.12. The Minister of the Environment (MOE) defines waste contaminated by radioactive material over 8,000 Bq/kg of ^{134}Cs and ^{137}Cs as designated waste, for which the national government is responsible for the treatment under the Act on Special Measures. The MOE established the procedure of cancelling the designation of the designated waste, which is applicable when the radioactivity concentration of the designated waste is reduced to 8,000 Bq/kg or less due to the radioactive decay. This is also an example for disposal on landfills of the material and waste, but the cancellation is not the case of a free release, since after the cancellation the waste is disposed on landfills under the standard in the Waste Management and Public Cleansing Act.

A.13. As for the category 2, in 2017 the Japan Atomic Energy Agency (JAEA) examined activity concentration of the waste generated in the 1F plant for recycling on-site, which is under regulatory oversight by the Nuclear Regulatory Authority (NRA) of Japan. This is also not an example of clearance of the waste because the recycled material is still under the Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors (the Nuclear Reactor Regulation Law).

A.14. As for the category 3, it has not been applied yet in Japan.

A.15. The above-mentioned two examples of the categories 1 and 2 after the 1F plant accident are given bellow.

Category 1: From off-site to off-site

- **Example of recycle of removed soil off-site generated from decontamination work in off-site**

A.16. The MOE established the Technology Development Strategy for Volume Reduction & Recycling of Removed Soil in April 2016 towards the final disposal of removed soil outside Fukushima Prefecture. In this strategy, it was clarified that the MOE would make a basic concept on safe use of recycled removed soil keeping the safety of radiation for the recovery workers who handle the soil and for the public. According to the strategy, the MOE subsequently established the basic concept on safe use of recycled removed soil in June 2016.

A.17. In the basic concept, it is clarified that the use of the recycled removed soil is limited such as for basic structure material of banking for coastal levee, disaster prevention forest on the beach, or roads, which is assumed not to change the form artificially for a long time period and is constructed in the public projects managed by the public authority. The recycled removed soil has to be used by the appropriate management according to the criteria based on the Act on Special Measures under the condition that activity mass concentration level of recycled removed soil is restricted by shielding of radiation with soil-covering in order to confine additional exposure dose below 1 mSv/y for workers and the public. The safety assessment was carried out by setting some exposure scenarios by the MOE in order to ensure that the dose exposed to the removed soil for workers and public was 1 mSv/y or less. Later, the concept was extended to management of other materials, when facilities were constructed using the recycled materials. In such cases the appropriate thickness of the shielding for radiation was ensured in order to make the exposure dose for workers and public below the level that needs no measures for prevention of radiation hazards.

A.18. The activity mass concentration level is below 8,000 Bq/kg in principle and is set to7,000, 6,000, 5,000 and 4,000 Bq/kg according to the purpose of the recycle, the shielding condition and annual working time for the use of the recycled removed soil. The value of 8,000 Bq/kg is the same as the concentration level that is given in the Act on Special Measures as a kind of a screening level for exemption from radiological regulatory requirements in the existing exposure situation.

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

A.19. The MOE designates waste contaminated by radioactive material over 8,000 Bq/kg (designated waste). If the activity mass concentration is over 8,000 Bq/kg, the exposure dose for workers and public would be above 1 mSv/y, according to the safety assessment provided by the MOE setting some exposure

scenarios. The national government is responsible for treating the designated waste since special control is necessary. On the other hand, when the radioactivity of the waste is 8,000 Bq/kg or less, it can be treated technically and safely using normal treatment method as additional exposure dose by the treatment is 1mSv/y or less for both workers and the public.

A.20. Taking into consideration such a situation, the MOE established the procedure to cancel the "designated waste" as follows:

- When the radioactivity of the designated waste reaches 8,000 Bq/kg or less due to radioactive decay, the Minister of the Environment can cancel the designation after consultation with person or entity who stores the designated waste temporarily and person or entity who has obligation for the material management, including transport and disposal in landfills after cancelling the designation (local municipalities or business operators). The cancellation of the designation is not carried out without their acceptance.
- After cancelling the designation of the designated waste, the waste is treated by local municipalities or business operators according to the treatment standards in the Waste Management and Public Cleaning Act. The MOE provides technical and financial support for the treatment as necessary such as explanation for safety of the treatment of the waste whose radioactivity is 8,000 Bq/kg or less, in order to facilitate the disposal of the waste after cancelling the designation.

Category 2: From on-site to on-site

- **Example of recycle on-site of waste generated on-site -**

A.21. Tokyo Electric Power Company (TEPCO) has planned that contaminated rubbles with less than 5μSv/h of surface dose rate, which are stored outdoor in the Fukushima Daiichi Nuclear Power Station (1F) on-site, will be recycled and applied in a restricted use only within 1F on-site. If the rubbles can be appropriately recycled to construction materials with suppressing additional effective doses for workers in 1F on-site and for the public outside 1F on-site, it will contribute to reduce the amount of radioactive wastes in the future because clean materials are not brought from outside of the site.

A.22. Activity concentration for recycling to restricted use within 1F on-site is estimated using a case-by-case approach in consideration of decommissioning activities of 1F on-site under existing exposure situation. This approach is based on the following basic concepts:

- The recycle within 1F on-site should not lead to increasing the effective dose unduly in the work environment and restrain a series of future decommissioning activities.

- The policy of radiation protection to regulate the implementation for 1F Nuclear Power Station should apply to the radiation risk to the hypothetical member of the public immediately outside the 1F site boundary due to additional sources and activities associated with the recycling on-site.

A.23. Figure A.1 shows the procedure for estimation of activity concentration for the recycle within 1F on-site. At the first step, activity concentration in material to be recycled is determined to meet the criteria of 1μSv/h of additional exposure due to the recycling. The 1μSv/h corresponds to the minimal value of the dose rates in the air at the 1m height from ground surface observed in 1F site. To justify estimated activity concentration, additional effective doses for workers closest to recycling material do not exceed 10% of dose limit for worker, 20mSv/y, in consideration of small margin to it at the 1F decommissioning activities. Additionally, two kinds of criteria are set up to assure the justification of radiation protection for the public outside the 1F site boundary. One of the criteria is effective dose rate along the boundary of less than 1mSv/y brought from all radiation sources of 1F after the recycle. Secondly, activity concentrations in groundwater for radionuclides migrated from recycling material do not exceed the concentration for operational target at 1F boundary to the ocean.

A.24. According to the above procedure, Japan Atomic Energy Agency (JAEA) calculated activity concentrations for recycling to restricted use; road materials and base of concrete building, within 1F on-site [42,43]. As shown in Table A.1, for example of radioactive cesium (^{134}Cs and ^{137}Cs), the results of estimated concentrations were 1.3×104 Bq/kg for the roadbed of asphalt road, 7.4×103 Bq/kg for the pavement of asphalt road, 1.0×105 Bq/kg for the roadbed of concrete road, 8.1×103 Bq/kg for the pavement of concrete road, and 1.6×104 Bq/kg for the base of the concrete building, under the ratio of ^{134}Cs to ^{137}Cs of 0.209 as of March 2016.

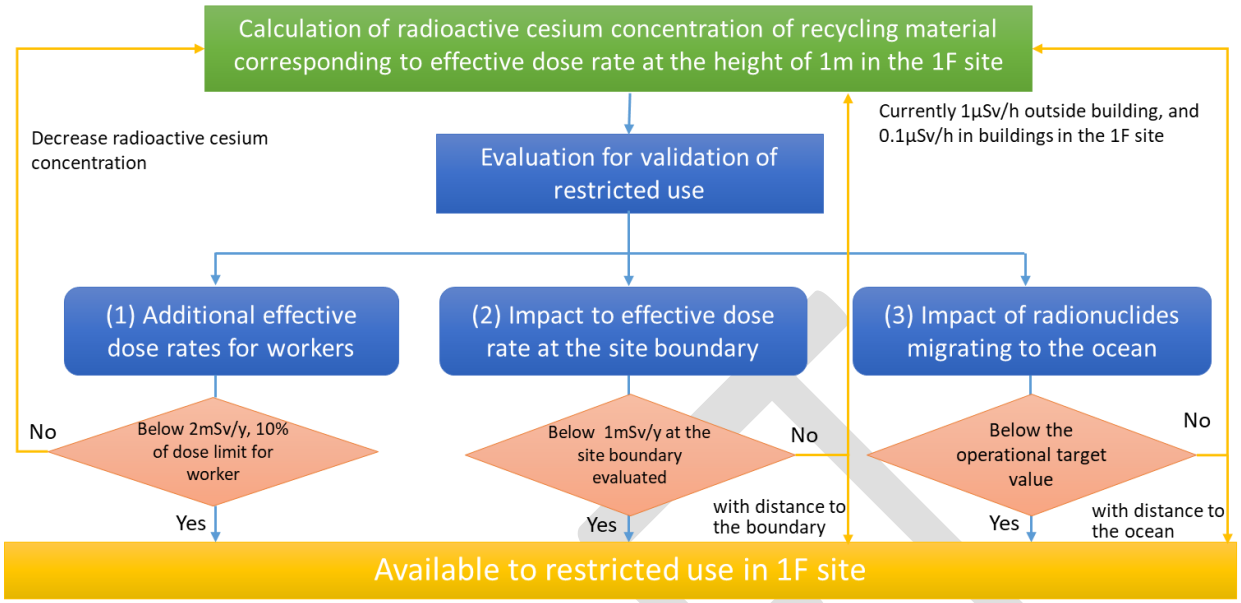

Figure A.1 Procedure for estimation of activity concentration for the recycle within 1F on-site

Table A.1 Estimated activity concentrations for recycling to restricted use; road materials and base of concrete building, within 1F on-site

Material	Application		Activity concentration (Bq/kg)	Shielding condition
Concrete	Asphalt road	Roadbed	13,000	Pavement thickness 5cm
		Pavement	7,400	No shielding
	Concrete road	Roadbed	100,000	Pavement thickness 15cm
		Pavement	8,100	No shielding
	Building concrete	Base	16,000*	Floor slab thickness 20cm

*Restricted use in the building based on the effective dose rate of 0.1μSv/h in the building

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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 endeavoured in various national and international studies in Europe, in the USA and by the IAEA. The following list is a non-exhaustive overview of available studies:

- European Commission: RP 89 0/ RP 101 [I-2]
- USA: NUREG 1640 [I-3] and ANSI/HPS N 13.12 [I-4]
- USA: Argonne National Laboratory "Surface Clearance Criteria for Workers" [I-5]
- JAPAN: JHPS Standardization Committee [I-6,I-7,I-8]
- UK Code of Practice [I-9]
- SUDOQU [I-10,I-11]

I-2. The radiological models underlying these studies are presented briefly in the following sub-sections.

RECOMMENDATIONS ON CLEARANCE BY THE EUROPEAN COMMISSION: RP 89 / RP 101

Overview of the approach

I-3. The recommendations RP 89 0 and RP 101 [I-2] on clearance issued by the European Commission contain radiological models for the derivation of surface specific clearance levels. RP 101 as the technical document contains the detailed description of this model, which is briefly summarised in this section. While the surface specific clearance levels were derived for metal scrap (steel, copper and aluminium), they can be regarded as fulfilling the requirements to be posed to clearance levels for unconditional clearance, as they cover scenarios for reuse and recycling. Because of the nature of the underlying scenarios, especially those for reuse, these clearance levels are applicable not only to metals but also to other items that are handled, treated and used, e.g. items made from plastics, wood, or glass.

I-4. The surface specific clearance levels recommended in RP 89 apply to the total surface activity concentration, fixed plus non-fixed, and are intended as an average over moderate areas, which is understood as "several hundred square centimetres up to 1 square meter", "depending on the type of material, contamination and homogeneity of the contamination". It is further argued in RP 89 that "surface contamination limits for metal scrap are largely independent of the metal type (steel, copper, aluminium), since the transport and handling are similar regardless of the metal."

I-5. The scenarios used for the derivation of surface specific clearance levels for metal scrap for recycling or reuse are based on a number of scenarios outlined in the following subsections. An overview of these scenarios is provided in Figure I-1. It can be seen that both sets of scenarios are very similar in structure. However, the significance of surface-specific clearance levels is deemed different in RP 89 and RP 101 in the two areas recycling and reuse of metal scrap: recycling is mainly governed by mass-specific 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."

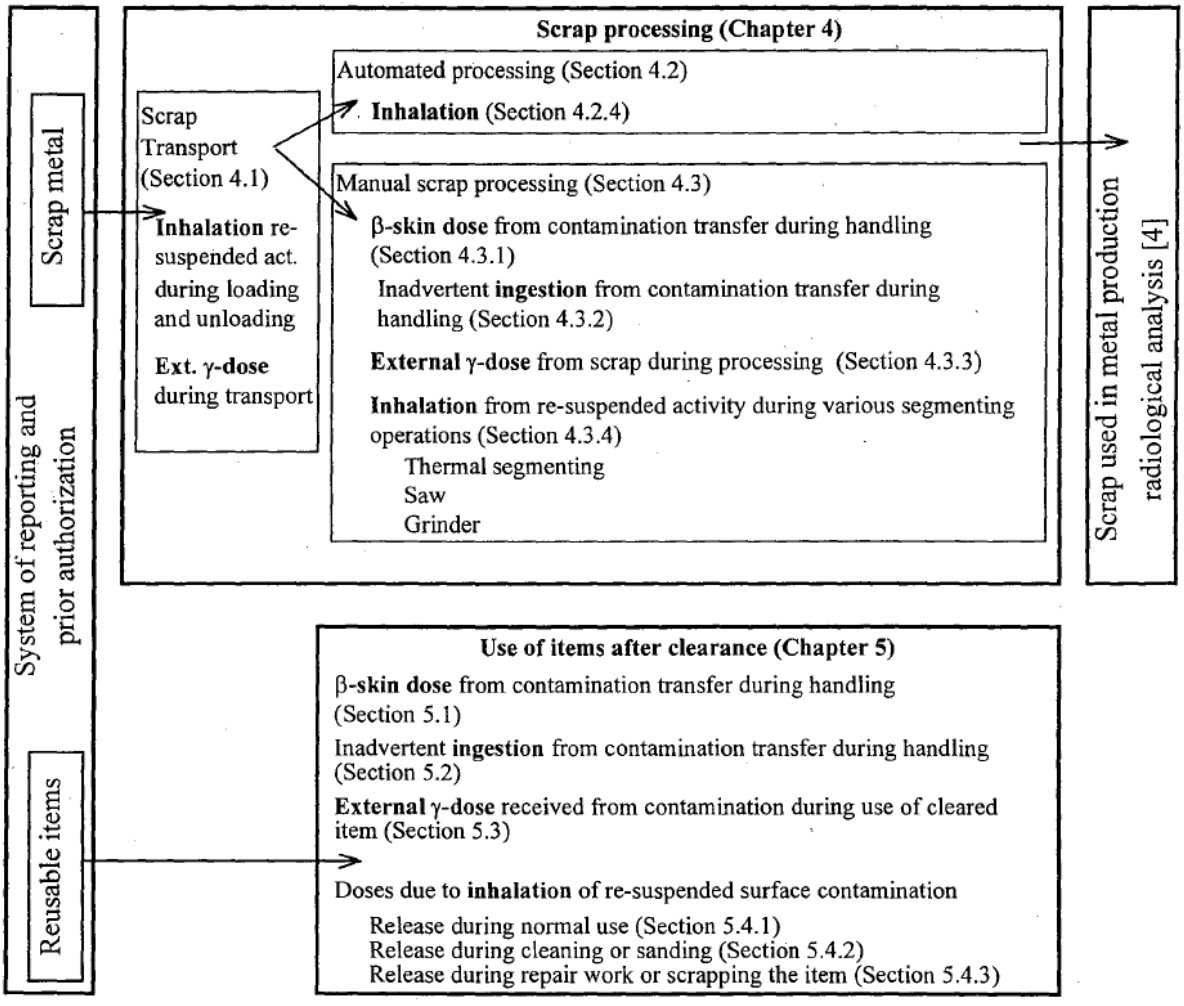

Figure I-1: *Overview of the scenarios for the derivation of clearance levels for metals forrecycling and reuse from RP 101 [I-2]*[10]

[10] The section references in the Figure I-1refer to sections of the publication RP 101

I-6. The scenarios developed in RP 101 are primarily of deterministic nature and represent normal situations during which contact and exposure to the cleared metal can occur. It should be noted that the radiological model for surface-specific clearance levels developed in RP 101 is totally independent from that developed for mass-specific clearance levels in RP 112 for metal scrap. This means that the surface-specific clearance levels are not derived using a conversion factor accounting for a mass to surface ratio.

I-7. In addition, the clearance levels for all radionuclides possessing radioactive decay products include the dose contributions from these decay products, which are accounted for by assuming they are in secular equilibrium with the parent nuclide and adding their doses to the dose calculated for the parent nuclide. A table in RP 89 and RP 101 lists these parent and daughter nuclides.

I-8. The following sub-sections describe those scenarios that have been used in RP 101 for the derivation of surface-specific clearance levels for unconditional clearance. Scenarios that explicitly pertain only to recycling are not discussed here.

General considerations in RP 101 to modelling reuse scenarios

I-9. RP 101 defines that "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." It is pointed out that "modelling reuse requires different scenarios than in the case of melting. 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." As this is not the case for reuse, no dilution or any modification of the material is assumed.

I-10. The surface-specific clearance levels for reuse rely primarily on surface contamination limits since the measurement of the mass specific activity would in many cases mean destroying the equipment's integrity.

I-11. All relevant exposure pathways are taken into account in the radiological model:

- external irradiation,
- direct ingestion of contamination via a hand-to-mouth pathway,
- inhalation of contamination from resuspension of activity,
- skin dose from transfer of contamination to parts of the human body.

External y-dose incurred during the reuse of cleared equipment

I-12. In RP 101 it is acknowledged that there may be a large variety of exposure conditions by which a person using for instance a cleared piece of equipment may be exposed. Therefore, an enveloping

approach has been taken where a worker is exposed by a large item, in this case a tool cabinet that has a comparatively large overall surface (2 panels (doors and back), 6 shelves, 2 sides; overall dimensions 2 m height, 1 m width, 0.4 m depth, leading to a total surface of 8 m²). It is assumed that the person using the cabinet is effectively exposed to 4 m² 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 incurred during the reuse of cleared equipment

I-13. An inadvertent ingestion dose 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 a sandwich or smoking a cigarette. This part of the model has been chosen similar to the one used for derivation of surface-specific clearance levels for recycling. It is conservatively assumed that ingestion takes place during 200 h/a with an ingestion rate of 1.2 cm²/h and a transfer of 1 % of the surface activity to the hand.

Inhalation dose incurred during the reuse of cleared equipment

I-14. RP 101 points out that basically four types of inhalation scenarios can occur:

- during normal use the surface activity can be shaken lose and re-suspended leading to inhalation of the activity;
- the item can be cleaned or sanded, for example in preparation for a new paint job, leading to re-suspension of the surface activity;
- repair work like welding or thermal cutting can be carried out; and
- at the end of the item's useful life it will almost certainly be scrapped, which means it could be thermally segmented,

of which the last two scenarios are very similar and are therefore treated together.

I-15. The normal-use scenario uses the following assumptions: exposure time 1,800 h/a, fraction of dust in the breathing air originating from the reused item 1 %, ambient dust concentration 0.2 mg/m³, normal breathing rate 1.2 m³/h.

Skin dose from 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 β-skin dose. This scenario uses the following parameter values: contaminated area of the skin 0.1 m², exposure time 1,800 h/a, factor describing transfer of contamination from the item to the skin: 1%. This scenario converts the skin dose to effective dose by using the weighting factor for the skin as 0.01.

Other scenarios for the derivation of surface-specific clearance levels

I-17. As seen in Figure I-1, RP 101 contains a number of other scenarios that are also used for the derivation of surface-specific clearance levels. These include scenarios for automated scrap processing (mainly for 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 above are included, yet with different parameter values. Manual scrap processing leads to the highest doses per unit activity, since the workers are in direct contact with the contaminated scrap.

I-18. A further analysis of the two most important exposure situations, inhalation from manual processing of scrap and external gamma irradiation from using cleared items, has been performed in RP 101 using dedicated stochastic models. These two exposure situations are deemed critical since they involve prolonged close contact with large quantities of scrap or large items and since the leading radionuclides in typical contamination vectors contain either high energy γ-emitters like ^{60}Co and ^{137}Cs or radionuclides with large inhalation dose coefficients like uranium and plutonium. It is shown in RP 101 that the choice of parameters in the deterministic scenarios has been chosen on the conservative side.

USA: REPORTS NUREG 1640 AND ANSI/HPS N 13.12

I-19. The radiological model chosen in the very comprehensive report NUREG 1640 [I-3] for the derivation of surface-specific clearance levels is different from the one described for RP 101, as NUREG 1640 primarily develops a complex radiological model aiming at deriving mass-specific clearance levels for the reuse, recycling and disposal of iron and steel scrap, scrap aluminium, scrap copper, and concrete rubble. These categories comprise the bulk of components that would be potentially cleared from nuclear or other licensed facilities.

I-20. Surface-specific clearance levels are derived from the mass-specific values by a conversion factor describing the mass-to-surface ratio of the material in question. The most probable values for these conversion factors have been set to 5.1 g/cm² for steel and to 280 g/cm² for concrete. This approximately 50-fold difference is why the clearance of steel or copper scrap yields the highest mean surficial effective dose equivalent or, correspondingly, the lowest surface-specific clearance levels.

I-21. A similar approach has been used in ANSI/HPS N 13.12 [I-4], where a similar conversion has been performed on the basis of the clearance levels provided in RS-G-1.7. The conversion factor in this case has simply been set to 1 g/cm², so that the values in Bq/cm² are numerically equal to those in Bq/g.

I-22. It should thus be noted that neither NUREG 1640 nor ANSI/HPS N 13.12 contain a genuine model for the derivation of surface-specific clearance levels. Nevertheless, the conversion of mass-specific to surface-specific clearance levels may be a viable approach for cases where a dedicated radiological model for derivation of surface-specific clearance levels would be too challenging.

USA: ARGONNE NATIONAL LABORATORY "SURFACE CLEARANCE CRITERIA FOR WORKERS"

I-23. The document "Potential Dose Distributions at Proposed Surface Radioactivity Clearance Levels Resulting from Occupational Scenarios" [I-5] contains an evaluation of the potential dose distribution resulting from surface radioactivity, using occupational radiation exposure scenarios. The aim was to test a set of surface-specific clearance levels for their compliance with dose limits or constraints for workers.

I-24. Two scenarios were considered in calculating dose distributions for thirteen selected radionuclides, most common in nuclear facilities:

I-25. 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 surfaces of interior floor and inside of the four surrounding walls, with equal levels on all surfaces. A worker inside such a building was assumed to incur radiation doses through (1) external radiation from the floor and interior walls; (2) inhalation of contaminated particles suspended from the contamination on the floor and interior walls; (3) ingestion of deposited dust particles; (4) external radiation from submersion in contaminated air; and (5) external radiation from deposited dust particles.

I-26. The second scenario assumes use of a contaminated desk in an office setting. It was assumed that the top of a writing desk of ordinary size was uniformly contaminated and that the receptor (worker) was sitting at a normal distance from the centre of the desk. A worker was assumed to incur radiation doses through (1) external radiation from the top of the desk; (2) inhalation of contaminated particles suspended from the contamination on the desk; (3) ingestion of deposited dust particles; (4) external radiation from submersion in contaminated air; and (5) external radiation from deposited dust particles.

I-27. The analysis was carried out assuming statistical distributions for each key parameter value, with a distribution type appropriate for the parameter and limited by reasonable boundaries. The analysis establishes a link between a given contamination on the surfaces and the distribution of resulting dose values, using the mean for the final assessment. As in this case scenarios for workers were analysed, the

dose values against which the results were assessed were in the range between 50 and 100 μSv/a. On this basis, the surface-specific clearance levels from which the analysis started were judged to be applicable.

JAPAN: JHPS STANDARDIZATION COMMITTEE - GUIDELINE FOR MOVING OUT COMMODITIES CONTAMINATED WITH RADIOACTIVE MATERIALS IN PLANNED EXPOSURE SITUATION[11]

I-28. With respect to the surface contamination control of commodities, guidelines have been developed by the Standardization Committee on Radiation Protection of Japan Health Physics Society for planned, emergency and existing exposure situations [I-6]. Table I-1 summarizes the main points of the guideline for planned exposure situation. In the Guidelines document the "commodities" are defined as solid-state valuable goods justified for the reuse or recycle when moving out from controlled area (e.g., vehicles, equipment and the other items). In this Annex, the word is replaced by the more pertinent term "object".

[11] The term "commodities" is used in the official translation of the document. However, in the context of this Annex the meaning "object" is more appropriate.

Table I-1: Summary of a guideline for moving out objects contaminated with radioactive materials in planned exposure situation by Standardization Committee on Radiation Protection of Japan Health Physics Society (extracted from Ref [I-6] with permission).

	Planned Exposure Situation
Dose criteria (effective dose)	Order of 10 μSv/y or less
Referred concept	Clearance
Basic point of view	• Moving out from controlled area to general • Application of the concept of clearance of many relatively small objects moved out
Exposure Scenarios	Handling of small packages [I-7] Handling of general objects [I-8]
Examples of readings of typical GM survey meter widely used in Japan	• 1,000 cpm (10 Bq/cm^2 of ^{60}Co) • 2,300 cpm (10 Bq/cm^2 of ^{137}Cs)

I-29. This guideline is assumed to be applied for moving out objects from radiation controlled area. In this case, applicability of the concept of clearance to moving out was examined. In general, there is no control of usage after moving out, and therefore it is similar to the concept of clearance. Clearance concept is based on an assumption of handling large amount of materials such as dismantling waste from a nuclear facility, while moving out objects in planned exposure situation assumes handling of many relatively small objects.

I-30. The surface contamination level of small objects moved out from radiation controlled area were calculated that correspond to the clearance criteria of 10 μSv or less of annual effective dose, on the basis of realistic exposure scenarios. It was also concluded that continuous control of objects moved out is not justified in terms of radiation protection. The applicability of clearance concept for objects moved out in planned exposure situation is demonstrated and included in the guideline.

I-31. Two exposure scenarios are used for derivation of surface contamination levels equivalent to the dose criterion for moving out objects in planned exposure situation. These are handling of small packages [I-7] and handling of general objects [I-8] (includes manually handled objects with 0.1 m^2 area, closely handled objects with 1 m^2 area and remotely handled objects with 10 m2 area). Considering both scenarios, ^{60}Co and ^{137}Cs surface contamination levels corresponding to the dose criterion of 10 μSv of annual effective dose are calculated to be 10 Bq/cm^2 for both nuclides. These surface contamination levels for ^{60}Co and ^{137}Cs correspond to readings of 1,000 cpm and 2,300 cpm respectively, using a typical GM surface contamination survey meter with 20 cm2 window based on JIS Z4504 (2008).

I-32. It should be noted that in the case of especially high energy gamma emitters, such as ^{60}Co and ^{137}Cs, surface contamination levels equivalent to the dose criterion of 10 µSv of annual effective dose significantly depends on the assumption of the size of the contaminated surface. Such high energy gamma emitters sometimes become key nuclides for surface contamination measurements of beta radiation. The contribution of such radionuclides is sufficiently high for surface measurements to be carried out for demonstrating compliance with clearance levels.

I-33. When the daily radiation control at nuclear or radiological facilities involves mostly objects of small size (much smaller than vehicles), applying surface contamination levels derived for clearance of large objects would be too strict and conservative. Separate numerical radiological criteria for surface contamination, used for daily radiation control and clearance, are recommended to be applied according to the dimensions of the surface likely to be contaminated.

UK: NUCLEAR INDUSTRY GUIDE TO CLEARANCE AND RADIOLOGICAL SENTENCING

I-34. The UK Nuclear Industry Guide to Clearance and Radiological Sentencing [I-9] contains a derivation of surface clearance levels for contaminated items in its Appendix F. This appendix illustrates the calculation of maximum alpha, beta and total activity levels for reuse of metallic equipment from dismantling of nuclear installations. The model given there is a direct reproduction from RP 101 [I-2], leading to the same derived surface-specific clearance levels.

SURFACE DOSE QUANTIFICATION – THE SUDOQU MODEL

I-35. A model for "SUrface DOse QUantification" or SUDOQU [I-9] is currently under development. This model is intended to evaluate the annual effective dose for members of the public resulting from exposure to surface-contaminated objects. Its initial objective thus was to calculate exposure from objects contaminated e.g. from nuclear accidents, taking into account all relevant exposure pathways (external-gamma radiation exposure, inhalation, indirect ingestion and skin contamination through wipe-off).

I-36. Recently, the applicability of this model to calculation of surface-specific clearance levels was 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 nuclides. The scenarios were then used to calculate surface-specific activities that would correspond to an annual effective dose of 10 µSv/a.

I-37. The results of these calculations were then compared with the results of RP 101 [I-2]. Differences were traced to different assumptions in parameters and exposure situations. One of the main

differences of both models is the fact that the SUDOQU approach considers reduction of the surface activity with time not only by the radioactive decay, but also by resuspension and wipe-off (transfer of activity to the hands). Likewise, the resuspended activity contributes to the increase in airborne activity concentration and can, in turn, partly re-deposit onto the object surface.

I-38. It was concluded in [I-11] that the suitability of the SUDOQU model for dose assessments related to clearance of objects from nuclear facilities could be demonstrated, but that further development will be needed to develop this model to such a level suitable for calculating surface-specific clearance levels.

REFERENCES TO ANNEX I

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ANNEX II

EXAMPLES OF SURFACE SPECIFIC VALUES FOR UNCONDITIONAL CLEARANCE

GENERAL

II-1. While no surface-specific clearance levels are provided in GSR Part 3 [II-1], there are a number of international and national recommendations and guidelines containing surface-specific clearance levels. This Annex is limited to examples on surface-specific clearance levels for unconditional clearance, as a comparison of values is only meaningful for radiological models for unconditional clearance. In all cases where the models aim at providing clearance levels for a specific clearance option, too many differences in model assumptions, exposure scenarios and parameter values will exist that preclude a direct comparison of the models.

OVERVIEW OF EXISTING RECOMMENDATIONS AND STUDIES

II-2. This Annex provides a short overview of the following recommendations and studies in which surface-specific clearance levels have been derived. In addition, the IAEA Transport Regulations are addressed, which do not contain clearance levels, but from which surface-related activity values have been frequently misused as clearance levels, in order to point the fundamental differences in the radiological models underlying calculation of exposure from surface contaminations in the case of clearance and in the case of transport.

II-3. A synopsis of the derived surface-specific clearance levels for unconditional clearance is given below.

European Commission: RP 101 and RP 89

II-4. The most comprehensive and pertinent study on surface-specific clearance levels for unconditional clearance is RP 101 [II-2]. This document is a technical support document for the European Commission's recommendation Radiation Protection 89: "Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations" [II-3] and contains a compilation of the methods and parameters used to derive the clearance levels for surface contamination published in RP 89 (while a second technical document deals with the methodology and models used to derive the clearance levels for mass specific activity). The surface-specific clearance levels have been derived for metallic material originating from nuclear installations and intended for recycling and reuse outside the nuclear regime. All calculations are based on an individual effective dose of 10 µSv per year.

II-5. The scenarios have been divided into two major categories: scrap processing and the reuse of cleared items. The scrap processing scenarios have been divided up further into the categories: transport, automated processing and manual processing. The results of the deterministic radionuclide specific dose calculations are presented in tables as μSv/a for a unit surface activity of 1 Bq/cm². To derive the surface contamination clearance levels, the set of deterministic scenarios is used to calculate the nuclide specific contamination level for each scenario which would lead to a dose of 10 μSv/a. The smallest derived value has then been used as the clearance level for the radionuclide in question. In most cases, this value is based on the reuse scenario.

Germany

II-6. The study [II-4] was conducted in the course of the preparation of a new version of the German Radiation Protection Ordinance in 1998 and 1999. Much of this work was carried out in parallel to the study RP 101 [II-2] presented in section 0 above, so that many of the scenarios are similar. The main aim of the study [II-4] was to derive surface-specific clearance levels for unconditional clearance, so that only one set of clearance levels has been provided, so that reuse and recycling are both fully covered. This makes the surface-specific clearance levels applicable also for unconditional clearance.

United Kingdom

II-7. A Guide on good practice concerning clearance and radiological sentencing [II-5] provides guidance on the principles, processes and practices that should be used when determining whether an article or material may be released from any further controls on the basis of radiological protection considerations. It identifies approaches to segregate radioactive or potentially radioactive substances and articles from substances and articles that are 'out of scope' of radiation protection considerations. This document does not derive surface-specific clearance levels itself, but makes reference to document RP 101 [II-2]. The surface-specific clearance levels provided in [II-2] are discussed in the Guide [II-5] for application in clearance of objects with surface contamination.

USA

II-8. A very comprehensive study on clearance of metal scrap carried out in the USA has been published as report NUREG-1640 [II-6]. In this report mass-specific and surface-specific clearance levels have been derived for metal scrap. This report provides a description of calculations and their results estimating potential annual doses, normalized to a unit concentration, to an individual following the clearance of specific materials. These materials are scrap iron and steel, copper, aluminium, and concrete rubble from licensed nuclear facilities. The estimated potential doses are calculated probabilistically to

account for a large number of possible variations in each of the 86 scenarios. These scenarios encompass the full range of realistic situations likely to yield the greatest normalized doses. Each scenario was analysed with the 115 radionuclides considered most likely to be associated with materials from licensed nuclear facilities. The design basis of the analyses is to realistically model current processes, to identify critical groups on a nuclide-by-nuclide basis, and to enable the conversion of a dose criterion to a concentration.

II-9. Normalized surficial effective doses to critical groups are provided for the various recycling and reuse scenario for each material stream, given in µSv/a per Bq/cm². Surface-specific clearance levels can be calculated by dividing the dose constraint 10 µSv/a by these values.

Sweden

II-10. The safety guide of the Swedish Radiation Safety Authority [II-7] covers guidance on clearance of materials, rooms, buildings and land for Swedish nuclear installations and facilities using radioactive materials. It contains surface-specific clearance levels only for clearance of buildings for reuse or demolition, taken directly from the Guide RP 113 [II-8], but does not contain surface-specific clearance levels for unconditional clearance.

IAEA Transport Regulations: SSR 6

II-11. As has been pointed out in section 7 of this Safety Guide, the IAEA transport regulations SSR-6 [II-9] contain surface-related values of 0.4 Bq/cm² for beta and gamma emitters and low toxicity alpha emitters and 0.04 Bq/cm² for all other alpha emitters in the definition of contamination (for fixed and non-fixed contamination) as well as values of 4 Bq/cm² and 0.4 Bq/cm², respectively, for the definition of surface contaminated objects (SCO-I), relating to the non-fixed contamination only. These limits are applicable when averaged over any area of 300 cm² of any part of the surface.

II-12. These values have been derived in 1961 using a very simple model [II-12], i.e. almost half a century ago, with adjustments in the transition from Ci to Bq and appropriate rounding applied. A review of this model together with proposal of new modelling approaches for limiting the surface contamination on packages and conveyances in transport has been given in [II-11]. Concerning the Fairbairn model [II-12], the following assessment is given there:

> *"The Fairbairn model limits its consideration to inhalation of airborne contamination and transfer of contamination to the hands under a specified set of exposure scenarios. The permissible levels of contamination are constrained so as not to result in an airborne concentration greater than the maximum permissible concentration in air (MPCa) specified in the 1959 recommendations of the*

International Commission on Radiological Protection (ICRP). These levels should also constrain the dose to contaminated hands to what was considered to be good-practice in the 1960s. These constraints were applied to the alpha- and beta-emitting radionuclides then considered the most hazardous, ^{239}Pu and ^{90}Sr respectively, giving the limits noted above.

Since these limits were derived there have been a number of changes in radiation protection philosophy and dosimetry, mainly as a result of the recommendations of the International Commission on Radiological Protection. These include changes in the dose coefficients for inhalation of radionuclides and a change in the specification of the annual dose limit for workers. Also, during the period since the contamination limits were derived much experience has been gained in their use, and in contemporary transport operations. These developments have created the conditions in which a review of these limits is required.

Some inherent limitations of the Fairbairn model have also been recognised. For example, the limited range of radionuclides and exposure pathways considered, the high occupancy times assumed, uncertainties in the resuspension mechanism, out-dated dose coefficients and dose criteria, and the fact that the possible exposure of members of the public were not considered."

II-13. The limit values derived in 1961 [II-12] are therefore not applicable to clearance.

SYNOPSIS OF SURFACE-SPECIFIC CLEARANCE LEVELS

II-14. The studies and recommendations discussed above contain the surface-specific clearance levels given in Table II-1. This table contains only those clearance levels that refer to a clearance option which can be reasonably identified with unconditional clearance, i.e. no clearance levels for buildings or land. All surface-specific clearance levels given in this table have been derived on the basis of an effective individual dose of 10 µSv per year.

II-15. The comparison shows that for strong gamma emitters like ^{60}Co, ^{137}Cs or ^{154}Eu as well as for alpha emitters like ^{242}Pu and ^{241}Am, there is generally good agreement, indicating that the model assumptions for external irradiation (gamma emitters) as well as those for inhalation of resuspended surface contamination (alpha emitters) are based on similar assumptions.

II-16. Agreement of the values for strong beta emitters like ^{90}Sr can also be considered to be fairly good, indicating that ingestion pathways (direct and secondary ingestion) are generally based on similar assumptions.

II-17. The values for weak beta emitters and electron capture emitters like ^{3}H, ^{14}C, ^{36}Cl and ^{55}Fe, differ more significantly, indicating that the scenarios in the assessment underlying the derivation of these

surface-specific clearance levels are significantly different with respect to assumptions on secondary ingestion pathways, skin contamination and other scenarios.

Table II-1: Overview of surface-specific clearance levels for unconditional clearance from the studies discussed in the Annex II

Radionuclide	RP 101 MIN [Bq/cm²]		German StrlSchV [Bq/cm²]	NUREG 1640 – [Bq/cm²] based on the:	
	unrounded	rounded	rounded	Mean	95th percentile
H-3	25,000	10,000	100	1,500	700
C-14	770	1,000	100	1,600	1,100
Cl-36	130	100	100	29	7
Fe-55	1,500	1,000	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

REFERENCES TO ANNEX II

[II-1] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

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ANNEX III

EXAMPLES OF MASS SPECIFIC VALUES FOR CONDITIONAL CLEARANCE

Example from United Kingdom

III-1. Examples of conditional clearance levels that have been derived in Member States are: scrap metal for melting (EC RP 89 [III-1] for activity concentration per unit mass and EC RP 101 [III-2] for activity concentration per unit surface area), and conditional clearance levels for buildings for demolition or reuse (EC RP 113 [III-3] and EC RP 114 [III-4]). Conditional clearance levels for disposal of wastes in landfill sites have been adopted in Germany (reference) and in the UK (for small volumes of very low level waste VLLW). In the UK, this conditional clearance is called conditional exemption (for reasons of continuity with old UK terminology) and the conditions relating to conditional clearance of VLLW for disposal are given Table III.1.

Table III-1. UK conditional clearance of VLLW for disposal in landfill sites [III-5]

Radioactive waste	**Maximum concentration of radionuclides**	**Maximum quantity of waste to be disposed of per calendar year**
Solid radioactive waste, with no single item >$4x10^4$ Bq	$4x10^5$ Bq for the sum of all radionuclides per 0.1 m^3	2 x 10^8 Bq/a
Solid radioactive waste containing tritium and C-14 only, with no single item >$4x10^5$ Bq	$4x10^6$ Bq of tritium and C-14 per 0.1 m^3	2 x 10^9 Bq/a

Example from Belgium

Introduction

III-2. FBFC International is a nuclear facility situated in Dessel, Belgium. From 1960 until 2012, it produced fuel assemblies of uranium for NPPs. In October 2011, it was decided to shut down the installation for economic reasons.

III-3. 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 to be collected in the water treatment building, where it was treated before discharge in the environment. During

operation a reduction of the discharge limit happened due to regulatory change (the discharge limits were never exceeded).

III-4. An initial decommissioning survey identified several leaks in the underground network (mostly at the level of joints between pipes) resulting in deposition of small amounts of uranium in the soil (mainly sand). In addition, slightly contaminated sand was found in canals outside the facility site due to sedimentation. This sand was brought on-site and was part of the contaminated soil to be evacuated. The total volume was estimated at 8300 m^3 and about 12000 ton.

III-5. For unconditional clearance of soil, a level of 1Bq/g for U_{tot} is accepted by the Belgian nuclear regulator (Federal Agency for Nuclear Control - FANC). The soil samples showed levels slightly above this concentration.

III-6. According to the Belgian regulation, conditional clearance is possible based on a licence from the regulator. In his license application, the authorized party has to propose a conditional clearance level below the exemption level and include a radiological impact study demonstrating that the individual dose criterion of 10 µSv/a is not exceeded and that the collective dose stays below 1man.Sv/a.

III-7. The authorized party decided to apply for a conditional clearance license and to radiologically sort the sand in 3 categories, with final destination based on the activity concentration:

- < 1Bq/g : unconditional clearance
- 1Bq/g to 10 Bq/g : disposal in conventional landfill
- > 10 Bq/g : radioactive waste to NIRAS/ONDRAF[12]

Impact study for disposal in conventional landfill

III-8. The selected landfill site for disposal of the uranium contaminated soil is situated in the province of Antwerp and is destined for hazardous waste. It also accepts NORM-waste. Specific zones of the disposal will be used for the low-level contaminated soil of FBFC.

III-9. The impact study [III-6] 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}. The expected volume is lower.

III-10. The results are summarized in the table below for non-radiological workers who might be affected by the conditional clearance process.

[12] Belgian Waste Management Agency

Table III-2. Results of the impact study for disposal in conventional landfill [III-6]

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

III-11. The table shows that the most exposed worker will be the worker involved in the unloading of the sand on the landfill site. It is assumed that the considered workers perform only one of the listed tasks and that the job of unloading would be shared by 2 workers. Therefore, it was concluded that an activity concentration level of 10 Bq U_{tot}/g of soil will not give an individual annual effective dose in excess of 10 µSv to any non-radiological worker.

III-12. A similar analysis was performed for members of the public for all age categories, living in the vicinity of the landfill, cultivating a garden and walking on the landfill, leading to a similar conclusion.

III-13. In addition, a dose calculation was performed to estimate the impact of on-site sorting of sand on involved workers. The result was also found to be below 10 µSv per year, if workers use protective equipment typical for such works. Nevertheless, these workers are considered as occupationally exposed workers by the authorized party.

III-14. On the basis of this study, a licence for conditional clearance up to activity concentration levels of 10 Bq U_{tot} /g of sand was granted to FBFC by the FANC for removal of a maximum of 12450 ton of waste to a conventional landfill for hazardous waste.

III-15. Since the activity concentration stays below 10 Bq U_{tot} /g, no transport license for evacuation to the landfill site is required.

Traceability of the conditionally cleared soil

III-16. The information about the amount and the location of the cleared soil will be preserved by means of 2 sets of documents per transported container:

- Departure document by the nuclear operator containing: Container ID, Type and amount of packages, Radionuclide and Activity content, Total mass, Date of pick-up.
- Reception document by the landfill operator containing: Container ID, time of delivery, total mass, location on site.

The documents have to be kept by the operator during 30 years. At license termination, the documents are transferred to the FANC.

Example from new IAEA publication on disposal of waste in landfill sites

III-17. IAEA is preparing a publication on disposal of wastes in landfill sites [III-7]. The information provided in this document will enable Member States to derive conditional clearance levels for disposal in a landfill site that are relevant to their situation (e.g. climate and rainfall).

III-18. The study was triggered by the Fukushima accident, where large amounts of solid materials with a low level of radioactivity had to be disposed of in the remediation phase. Large amounts of solid materials with a low level of radioactivity are also encountered in decommissioning projects. The study started therefore with a focus on specific clearance of waste into landfills.

III-19. For the purpose of the study, a new tool, called "Clearance Tool" was developed for derivation of specific Clearance Levels (CLs) for different types of landfills and ultimately also for reuse and recycling of materials from decommissioning projects. The dose criteria and scenarios for derivation of these clearance levels are based on IAEA SR44 [III-8] and GSR Part 3 [III-9], namely 10 µSv/a for realistic scenarios and 1mSv/a for low probability scenarios.

III-20. The derivation of the conditional clearance levels focuses on radionuclides that are potentially relevant for accidental releases from nuclear power plants and takes into account the following 10 radionuclides: ^{90}Sr, ^{99}Tc, ^{106}Ru, ^{131}I, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{239}Pu, ^{241}Pu, ^{241}Am.

III-21. A basic set of exposure scenarios covers the situation where the materials are disposed on an ordinary landfill without any special radiation protection arrangements. They take into consideration exposure of workers that may arise from transportation of the material to the site, the handling of the material at the landfill and releases of radionuclides to the atmosphere in case of a landfill fire. Also, exposures of residents living close to the landfill are considered.

III-22. For the post-operational phase, a scenario with recreational use of the previous landfill, including the possibility of small excavations being performed in the landfill. In addition, an intrusion scenario is considered for estimating doses arising in case residential houses were to be built on the landfill. For this scenario, only the exposure of people living in this house are considered because these will be because of longer exposure duration and additional exposure pathways (ingestion of contaminated garden products) substantially higher than exposures of construction workers building the house.

III-23. Furthermore, the consequences of the controlled and uncontrolled release of leachates to groundwater and surface water are considered.

III-24. The model considers three generic landfill types:

- Landfill for inert waste (IWL)
- Landfill for municipal non-hazardous waste (MWL)
- Landfill for hazardous waste (HWL)

III-25. The different landfills types are assumed to have different properties concerning bottom liner, leachate collection system and top cover.

III-26. The life time of the landfill is divided into two phases, namely the operational phase and the post-operational phase, where a distinction is made between the period during and after institutional control.

III-27. In addition, the developed calculation tool [III-10] may be used to calculate conditional clearance levels that take account of specific site features.

III-28. Applying the clearance dose criteria described in section 2, conditional clearance levels for disposal of solid radioactive waste to landfill were derived.

III-29. The scenario in which a resident is living on the closed landfill after closure and after end of institutional control was treated as an unlikely scenario. Therefore, for this scenario the 1 mSv/a criterion is used.

III-30. Both deterministic and probabilistic calculations have been performed. The deterministic results are shown in the table below for the 10 radionuclides and for the 3 types of landfill. The results are not yet finalized.

Table III-3. Results of deterministic calculations of activity levels allowing disposal of waste in conventional landfills [III-7]

Radionuclide	Clearance level (GSR part 3) [Bq/g]	Activity level allowing disposal on landfills [Bq/g] (Deterministic calculation)		
		IWL	MWL	HWL
Sr-90	1	1.1	2	8
Tc-99	1	0.5	0.5	0.9
Ru-106	0.1	6	6	6
I-131	10	20	20	20
Cs-134	0.1	0.4	0.4	0.4
Cs-137	0.1	1	1	0.8
Ce-144	10	20	20	20
Pu-239	0.1	1.2	1.2	1.2
Pu-241	10	30	30	30
Am-241	0.1	1.4	1.4	12

III-31. The parameter values used in this calculation for food ingestion have been updated according to the latest IAEA publication on this subject [III-11].

III-32. These results have not applied the order-of-magnitude rounding used in SR-44 whereby, for example, a clearance level of 0.5 Bq/g was rounded to a clearance level of 1 Bq/g.

Example on clearance of liquids

III-33. In the framework of a regulatory initiative, the Health Protection Agency of the UK conducted two studies on clearance levels for liquids. Distinction was made between aqueous and non-aqueous liquids.

III-34. The study on non-aqueous liquids [III-12] (HPA-CRCE-006, 2010) demonstrates that the clearance levels for solids as recommended in RP122 part 1 are suitable for use for unconditional clearance of non-aqueous liquids for a majority of 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 conditional clearance by e.g. restricting the activity concentration or applying disposal constraints. Guidance is given in the referenced study.

III-35. A similar study performed for aqueous liquids [III-13] (HPA-CRCE-005, 2010) and based on a dose criterion of 10µSv/a gives clearance values ranging from 10^{-4} Bq/l to 10^{3}Bq/l, 80% of them laying

between 0.01 Bq/l and 1Bq/l. HPA recommends that the volume of these liquids at these levels that can be disposed of to a sewer is restricted to 3000 m^3/a. At the same time, HPA illustrates in this report the difficulty of measurement of these clearance levels for some radionuclides under laboratory conditions.

REFERENCES TO ANNEX III

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[III-13] HPA-CRCE-005: Derivation of Liquid Exclusion or Exemption Levels to Support the RSA93 Exemption Order Review, Public Health England, 2010.

ANNEX IV

EXAMPLE OF THE APPLICATION OF THE CLEARANCE CONCEPT IN SMALL MEDICAL FACILITIES[13]

INTRODUCTION

IV-1. Certain facilities conducting practices with unsealed or sealed radiation sources use different amounts of radionuclides with short and very short half-lives (less than 100 days). Examples of such facilities are small research laboratories, medical departments and facilities for industrial applications where radiation sources with those characteristics are used, processed or stored. These facilities may be identified with the term 'small medical, industrial and research facilities'[14]. The amount of activity of the radionuclides used in such small facilities varies according to the practice under development. For example, for medical purposes, the activity used can vary from less than 1 MBq up to 100 GBq depending if it is for medical research, clinical therapy or diagnostic. Information on unsealed sources and sealed sources and their range of activity per practice can be found in several IAEA's documents [IV–1, IV–2].

IV-2. In such small facilities and activities, moderate amounts of radioactive waste[15] are generated, requiring an adequate management in order to guarantee the radiological protection of people[16] and the environment. With the proper methodology, a significant volume of these radioactive wastes could be cleared from regulatory control, being this the best option to contribute to waste minimization and to decrease the costs associated to waste management.

SCOPE

IV-3. The small facilities considered in this Annex are those which, due to the simplicity of the practices from the radiological perspective, have standardized procedures for the safe use of the radioactive sources. A standardized methodology for the clearance of radioactive material within those

[13] This Annex is based on a Practical Guide of the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO), developed through a FORO project on Implementation of the Clearance Concept and Criteria for Small Nuclear Installations Handling Radioactive Waste. The project was implemented under the IAEA's extrabudgetary programme on nuclear and radiation safety and security in Ibero-America. The Practical Guide entitled "Guía práctica para la implementación de la dispensa en instalaciones radiactivas" is available at www.foroiberam.org.

[14] See, for example, Decommissioning of Small Medical, Industrial and Research Facilities, Technical Reports Series No. 414 and Decommissioning of Research Reactors and Other Small Facilities by Making Optimal Use of Available Resources Technical Reports Series No. 463

[15] By moderate amounts of wastes, moderate quantity means less than 3 tons per year and per facility [IV–5].

[16] The term 'people' may refer to occupationally exposed workers, workers in general and members of the public.

practices would be a useful recommendation to facilitate the safe management of the radioactive waste by the authorized parties and, at the same time, to smooth the regulatory control process, including the records, regulatory inspections and verification of compliance with the relevant standards and regulations.

IV-4. The present Annex describe, as an example, a methodology applicable for the solid radioactive waste in a nuclear medicine department. This example of a methodology could assist authorized parties and regulatory bodies to protect people and the environment effectively and efficiently by using the concept of clearance in a practical way. Solid radioactive waste in a medical department is generated in the form of paper and plastic, 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 or contaminated equipment [IV–1].

IV-5. Liquid radioactive waste generated in a nuclear medicine department, that includes contaminated water and effluent, waste arising from chemical processing and decontamination solutions, blood or body fluids, discarded liquid radiopharmaceuticals, wound or oral discharges, and urine [IV–1], needs a special consideration by the treatment systems in the facility, making difficult to provide a general example. For this reason, it is not discussed in detail in this Annex, but some considerations are presented at the end.

CASE STUDY

IV-6. Considering a nuclear medicine department that has authorization of practices using the following 2 techniques:

- Gammagraphy studies for diagnostic and follow-up with Technetium-99m;
- Thyroid function tests and treating thyroid cancer with Iodine-131.

IV-7. The maximum activity of each radionuclide as well as the number of patients per week authorized in the facility for these practices is shown in Table IV–1[17].

IV-8. In a nuclear medicine department that provides the above-mentioned treatment and diagnosis techniques, radioactive solid waste is generated, needing adequate management to guarantee radiological protection of the workers, public and environment.

IV-9. The following is a non-exhaustive list of the types of solid radioactive waste that may occur as a result from the use of radionuclides, such as Technetium-99, and Iodine-131 [IV–2]:

- Solid compactable waste (papers, cottons, chiffon gloves);

[17] This example is taken from a real nuclear medicine department. All information presented in the tables are based on real case.

- Metals (syringe needles);
- Glass (vials).

TABLE IV–1. Maximum activities and number of patients for the practices within the example

Practice	Radionuclide	Half-life	Type of emitter	Patient per week	Maximum activity per week
Diagnostic	Mo–Tc-99m Generator	6.03 hours	gamma	70	40 GBq
Diagnostic and Therapy	I-131	8.04 days	gamma	45	74 GBq

Methodology for the clearance of waste in small facilities and activities

IV-10. The practical methodology presented in this Annex for the clearance of waste arising in small facilities and activities consists on the following main steps:

1. Segregation and collection;
2. Measurement/estimation of the activity in the waste;
3. Management options (storage, decay, clearance, disposal);
4. Reports.

STEP 1: Collection and Segregation

IV-11. Appropriate collection and segregation of the remnant radioactive material is a very important step of the methodology, and it is required in order to minimize waste hazards and to facilitate subsequent management of waste. A recommended procedure is that the waste collection and segregation is performed at the time and place where it is generated. This process would be done according to the type of radionuclide, its half-life, physical and chemical form, and other properties of the wastes such as pathogenic or physical hazards (stabbing).

IV-12. In contrast to other nuclear applications, the use of radionuclides in small facilities and activities usually involves only one radionuclide being used per medical procedure. This makes segregation of waste by individual radionuclides feasible [IV–1].

IV-13. In some cases, it may be convenient to segregate wastes according to their half-life, e.g. 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-14. In other cases, the solid wastes can be segregated depending on their physical characteristic such as compactible, non-compactible, incinerable, non-incinerable. It is important to remark that the bags and the containers used to collect wastes would not be over-filled such that their integrity is compromised.

IV-15. To assure an adequate collection and segregation, the nuclear medicine department needs to be provided with containers and bags with the corresponding labelling. For further information on segregation and labelling of wastes in nuclear medicine department it is recommended to refer to Ref. IV-1.

STEP 2: Measurement or estimation of activity concentration of the wastes

IV-16. The proposed methodology in this Annex to measure or estimate the activity concentration of the solid wastes in a nuclear medicine department is practical and simple. However, it is considered appropriate for the purposes of clearance in this type of small facilities, due to the low activities and the short lives involved[18]. By using this methodology, it is valid to assume that the risks of exposure to workers, public and environment is very low, particularly if the simple practice is conducted systematically and with adequate precaution.

IV-17. Once the waste is adequately segregated and collected as explained in Step 1, it is important to carry out its radiological characterization to determine the initial activity concentration or total activity for each waste stream generated in the practice under consideration.

IV-18. A standardized process for the measurement of initial activity concentration or initial total activity in the wastes is difficult to define, due to existence of a wide variety of containers with different geometries and materials properties in different medical departments. Consequently, each facility or activity needs to establish a measurement procedure according to their recipients or containers used for the practice and considering the technical properties of the devices employed to perform the measurement.

IV-19. It is important to emphasize that, due to the characteristics of medical applications, the activity and radionuclides involved in each medical practice, as well as the total activity authorized, is known with precision. Hence, the remnant activity in the waste could be estimated by mean of a simple balance of activity and the corrections for decay, in correspondence with the characteristics of the practice and the time frame involved.

IV-20. It is a responsibility of each nuclear medicine department to establish a simple and practical method to estimate the activity of the wastes, and to consider the existence of shielding factors that could affect the result of the measurements and make the appropriate corrections.

[18] Measurements or estimations of the activity within this type of facilities and activities is more a matter of confirmation of the inexistence of risk and a reassurance.

IV-21. The methods for initial activity measurement or estimation in the wastes differ depending if the wastes were originated from liquid or solid radioactive material. For example, for solid wastes which will result from the remaining liquids, after they dry, the measurement of activity is done directly on a radionuclide calibrator. For solids wastes generated as a consequence of patients' treatment (papers, cottons, chiffon gloves), the method to estimate the activity on the waste bags could be done by simple measuring the dose rate or counts rate at a certain distance.

IV-22. The methodology proposed to perform a simple and gross estimation of the activity concentration is described for solid wastes in the next paragraphs. In addition, the activity concentration of the wastes originated in the nuclear medicine department, are also calculated to show how to apply the proposed methodology.

IV-23. In case of solid waste disposed in containers, it is recommended to perform the measurements related to each container at the time in which the container/recipient/bag is shut, usually when the remnant activity is the highest. This would allow an adequate labeling of each container indicating activity, date and its transfer to the storage room. In order to ensure representativeness, it is recommended to take several measurements in connection with each container/bag and use the most conservative (i.e. the maximum measured value) for the subsequent calculations, described below.

IV-24. Depending on the involved radionuclides, the measurement of the activity in the wastes can be done by counting on a detector for beta emitters or by measuring the gamma dose rate at a certain distance.

IV-25. For practical reasons, it can be assumed that the container/recipient behaves like a point source, and the measurement is done as if there was no shielding, as shown in Figure IV–1:

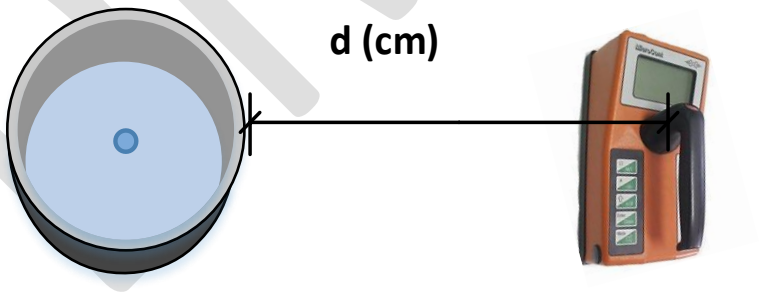

Figure IV–1. Geometry of the measurement. Reproduced courtesy of FORO [IV–3].

Estimation of the activity concentration from the measurement of the counts rate in the waste bag/containers[19]

IV-26. Taking into account the distance between the source and the detector, the activity concentration of the wastes could be estimated using the expression (V–1) [IV–4]:

$$C_A = \frac{N \cdot 4 \cdot \pi \cdot d^2 \cdot Fc}{A_D \cdot \varepsilon \cdot M_B} \quad \text{(IV–1)}$$

where C_A is activity concentration on the waste (Bq/g), N is measurement of the counts on the detector minus the background (cps), ε is efficiency of the detector (0 – 1), A_D is area of the detector (m^2), d is distance between the surface of the waste bag and the detector (m), M_B is waste bag weight (g) and Fc is correction factor[20] (equal to 2).

IV-27. For beta emitters radionuclides, the bag used for the wastes would be of a lower thickness of the one used for collecting gamma emitters so as to decrease the absorption of the beta particles at the time of the measurement. It is recommended to take several measurements near the surface of the bag, at a distance of ~5cm, and use the most conservative one (i.e. the highest measured value).

Estimation of the activity concentration from the measurement of the dose rate in the waste bag or container[21]

IV-28. The activity concentration could be estimated by measuring the dose rate of the waste bag or container, using the expression (IV–2).

$$C_A = \frac{\dot{D} \cdot d^2 \cdot Fc}{\Gamma \cdot M_B} \quad \text{(IV–2)}$$

where C_A is Activity Concentration of the waste (Bq/g), $\dot{D}$ is Dose rate at the distance d, minus the background dose rate (mSv/h), d is distance between the surface of the bag and the detector (m), Γ is specific gamma constant for the radionuclide (mSv m^2/h Bq), M_B is waste bag weight (g) and Fc is correction factor (equal to 2).

[19] This expression applies for beta and gamma emitters.

[20] The empirical correction factor is to considers geometry and autoabsorption effects

[21] This expression only applies for gamma emitters.

IV-29. The distance between the bag or the container and the detector recommended for gamma emitters is ~30 cm.

IV-30. In both cases, Fc would be applied in order to compensate the fact that, for practical reasons, the source is assumed to be a punctual source. Empirical studies [IV-3] show that if the measured activity is multiplied by a factor of 2, the correction factor can be assumed as adequate.

IV-31. For illustration purposes, examples of the measured dose rate in each radioactive solid waste stream in a nuclear medicine department is presented in Table IV–2.

TABLE IV–2. Example of results of measurement of dose rates of the solid wastes in a nuclear medicine department

Radionuclide	Waste bag	Waste type	T ½ (days)	Weight (g)	Dose rate to 30 cm (mSv/h)
Tc-99m	1223	Gloves, paper, cotton	0.25	3000	0.05
	1224	Vials, syringes (without needles)	0.25	2500	0.09
I-131	3220	Gloves, paper, cotton	8.04	2500	0.045
	3221	Vials, syringes (without needles)	8.04	3250	0.062

IV-32. Using the expression (IV–2), the activity concentration for each waste stream is estimated and shown in Table IV–3.

TABLE IV–3. Calculation of activity concentration of solid waste

Waste bag	Radionuclide	$\mathbb{\Gamma}$ (mSv m²/h Bq)	Activity concentration (KBq/g)
1223	Tc-99m	3.317E-11	90.4
1224	Tc-99m	3.317E-11	195.3
3220	I-131	7.64E-11	42.4
3221	I-131	7.64E-11	44.9

IV-33. Taking into consideration the activity concentration values obtained by using the expression (IV–2), and the clearance levels for the radionuclides involved, the most adequate waste management option for these wastes can be defined according to the Fig. IV–2: Management Options.

STEP 3: Management options and storage decay time

IV-34. As mentioned before, once the measurement or estimation of the activity concentration of the waste has been done, the management options needs to be chosen taking into consideration the Figure IV-2.

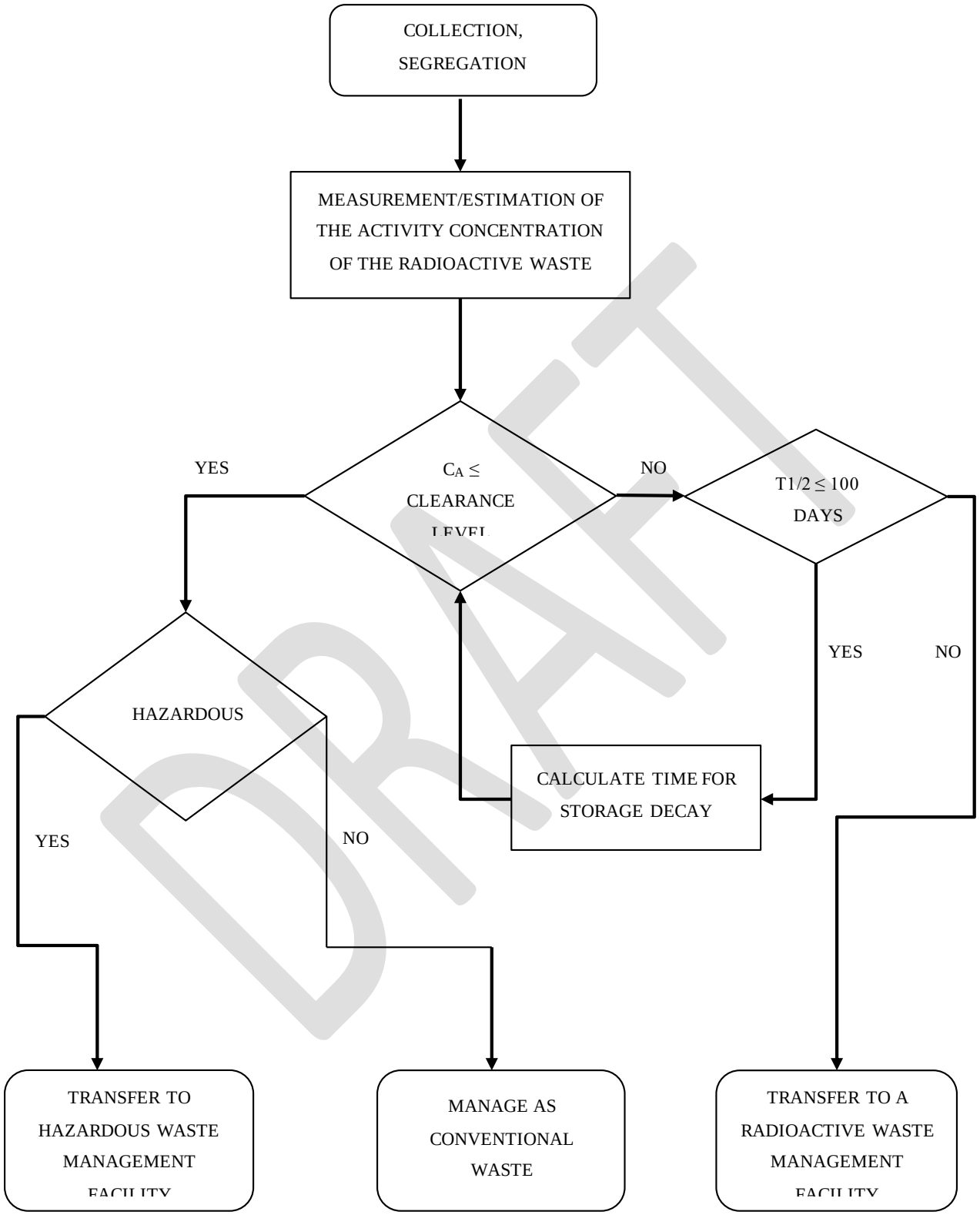

Fig. IV–2. Management Options

IV-35. As mentioned above, once the wastes resulting from the practices have been properly collected and segregated, the measurement or estimation of the activity concentration of the waste would be performed to determine the management option. The result of this estimation of the activity concentration from the measurement should be compared to the relevant clearance level for the radionuclide involved as established in GRS Part 3.

IV-36. Wastes from a nuclear medicine department with an activity concentration below the clearance level could be managed as conventional waste because, from the radiological perspective, if they do not contain other hazardous material. If the wastes do not comply with the clearance level, they should be transferred to a conventional or hazardous waste treatment facility as appropriate for their management.

IV-37. If the activity concentration is above the clearance level and the half-life of the radionuclide of the radioactive waste is below 100 days, the wastes could be stored in the small facility for a period of time (t) in order to allow radioactive decay until the clearance levels authorized are met. For radioactive material which is not in conditions to be cleared with the described methodology because the activity concentration is so high that they would require long term storage in the small facility, it is recommended to transfer them to a radioactive waste management facility for adequate treatment or disposal, according to the applicable regulations in the country.

IV-38. In order to calculate the period of time (t) for radioactive decay the following IV–3 expression could be used [IV–6]:

$$t = \frac{T_{1/2} \cdot \ln\left|\frac{C_A}{N_D}\right|}{\ln 2} \qquad \text{(IV–3)}$$

where $T_{1/2}$ is half-life of the radionuclide, N_D is clearance level of the radionuclide (Bq/g o Bq/l) and C_A is activity concentration of the radionuclide (Bq/g o Bq/l).

IV-39. In the example of the nuclear medicine department the storage decay time for each solid waste stream is shown in Table IV–4.

IV-40. Once the calculation of the storage decay time until the clearance level is met has been performed, the wastes should to be transferred to the radioactive waste storage room for temporary storage. In addition, the labelling of the wastes should be carried out including radionuclide, activity concentration, date and probable clearance date.

TABLE IV–4. Storage decay time for solid wastes

Waste bag	Radionuclide	T ½ [days]	Initial activity concentration [kBq/g]	Clearance level* [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

* For the purposes of the example of the methodology for the application of the clearance concept, the clearance levels where taken from GSR Part 3 table I.1.

IV-41. For the example under development, it can be observed that the wastes should be stored in the facility only for a few days before performing the actual release to the environment. The waste stream that requires more storage decay time is the one corresponding to the wastes arising from clinical therapy, where usually higher doses are used.

IV-42. It is recommended that after the storage decay time is reached and before proceeding to clearance measurements of the wastes, a quick and simple check is performed to indicate whether the applied methodology provided the expected outcomes or not. Such quick check could be performed by gamma dose rate measurements. Normally, the result is expected to be close to the background radiation level. If the result of such measurement is double the background or above, that is a clear indication of a failure in the procedure. Therefore, the pertinent investigation should be conducted, and if necessary, the procedure repeated. If the result is below, one could proceed with the final activity concentration measurements to confirm compliance with clearance levels.

IV-43. Wastes that are in condition to be cleared from regulatory control could be treated as conventional or hazardous waste, as relevant. If conventional, they can be disposed in common landfills with domiciliary wastes without any further consideration on radiation protection. If hazardous, they are sent to a hazardous material landfill. This management option is the most convenient for the small facilities and activities given that minimizes the costs of waste treatment or disposal in special landfills that requires continuous surveillance.

IV-44. Table IV–5 shows the management option for each waste stream of example of the nuclear medicine department after the storage decay time has been reached.

TABLE IV–5: Solid waste management option

Waste bag	Initial activity concentration [kBq/g]	Estimated activity concentration after decay time [Bq/g]	Control-measurement	Management option
1223	90.4	100	Background	Conventional waste
1224	195.3	100	Background	Conventional waste
3220	42.4	100	Background	Conventional waste
3221	44.9	100	Background	Conventional waste

IV-45. It can be observed that after the decay storage time in the nuclear medicine facility, each of the waste streams fulfill with the clearance level [IV-7]. Therefore, they are in conditions of being treated as conventional wastes. It is important to remark that in some countries vials and containers are often recycled instead of being disposed in landfills.

STEP 4: Record keeping

IV-46. Once the compliance with the clearance levels is verified, it is important to remove any label with the radioactive material logo of the waste packages before proceeding the release dispose as conventional or hazardous wastes.

IV-47. It is necessary that small facilities like nuclear medicine departments implement an adequate record keeping system that ensure that the clearance procedure has been performed in the framework of a management quality system, and that wastes are traceable from the cradle to the grave. These records are important to guarantee authorized parties and regulatory bodies control, and to allow tracing of every step in the waste management procedure. In addition, the records should be kept at least during the whole life of the nuclear medicine facility.

IV-48. For the purpose of record keeping, the following information could be registered for solid wastes:

- Identification of container;
- Radionuclide;
- Waste weight;
- Result of measurement and date;
- Activity or activity concentration;
- Decay time;

- Clearance probable date;
- Result of control measurement;
- Release date.

IV-49. The record of each solid waste stream considered in the example of the nuclear medicine department is presented in Table IV–6.

TABLE IV–6. Records of each waste stream for the nuclear medicine department

[ID] Waste bag	[RN] Radio-nuclide	[$T_{1/2}$] Half-Life [days]	Net container weight (g)	Dose rate (mSv/h)	Measurement date	[CA] Activity concentration [kBq/g]	[CL] Clearance level [Bq/g]	[t] Decay time [d]	Clearance probable date	Check measurement	Release date
1223	Tc-99m	0.25	3000	0.05	12-03-2018	90.4	100	2.46	15-03.-2018	background	16-03-2018
1224	Tc-99m	0.25	2500	0.09	12-03-2018	195.3	100	2.73	15-03.-2018	background	16-03-2018
3220	I-131	8.04	2500	0.045	12-03-2018	42.4	100	70.17	22-05-2018	background	23-05-2018
3221	I-131	8.04	3250	0.062	12-03-2018	44.9	100	70.85	22-05-2018	background	23-05-2018

Considerations on liquid wastes

IV-50. Accumulation of liquids could arise in a nuclear medicine facility or in any other small facility or activity, if remnants of the vials are placed in a recipient for storage, pending proper management. Even though this Annex does not provide an example for clearing liquids some general recommendations are provided[7].

IV-51. For segregation of liquids wastes containing radioactive material, the following two categories could be used: aqueous or organic.

IV-52. The existence of a wide variety of shapes and volumes of containers used for liquid radioactive materials makes it impossible to develop a unique simple method for activity estimation of the liquid's waste. As consequence, every small facility or activity should have its own procedures for the activity estimation according to their ways of storage. For example, the measurement of liquid samples could be done in an ionization chamber, calibrated to a specific radionuclide, where the vial is introduced.

IV-53. For the management of radioactive waste, it is usually recommended to make available standardized containers, using certificate reference materials (CRM) from a laboratory to proceed to measure the activity in the waste. It is necessary to select a representative sample of the waste. In the case of aqueous solutions, this is usually achieved by mechanically homogenizing the liquid before taking the sample. In the case of two or more liquid phases, it is necessary to take a sample of each phase and use the most conservative value of the activity concentration, this means the highest activity concentration measured.

IV-54. The activity concentration of a liquid sample could be estimated using the expression (IV–4):

$$C_A = \left(\frac{N - N_0}{\varepsilon \cdot V_M}\right) . Fc \qquad \text{(IV–4)}$$

where C_A is activity concentration of the Radionuclide Bq/l, N is counts rate s^{-1}, N_0 is background (s^{-1}), ε is efficiency of detector, V_M is volume of sample (l) and Fc is a correction factor.

IV-55. Fc is usually applied after the activity concentration is estimated (for example, 1.2 or more) in case of gamma or beta emission [IV–5].

IV-56. For liquid wastes the following information would be registered:

- Identification of liquid;
- Radionuclide;

[7] Usually, in a nuclear medicine department a significant part of the radioactive wastes results from the urine of the patients that are treated with radioactive material for diagnosis or therapy, especially with Iodine-131. The management of these liquid wastes are not treated in the present example.

- Volume;
- Estimated Activity or activity concentration;
- Date of estimated activity;
- Decay time;
- Clearance probable date;
- Result of control measurement;
- Released activity and annual released limit;
- Release date.

REFERENCES TO ANNEX IV

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