

IAEA Safety Standards

for protecting people and the environment

Safety of Nuclear Fuel Cycle Research and Development Facilities

Specific Safety Guide

No. SSG-43



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

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

IAEA SAFETY STANDARDS SERIES No. SSG-43

SAFETY OF NUCLEAR FUEL CYCLE RESEARCH AND DEVELOPMENT FACILITIES

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2017

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email: sales.publications@iaea.org
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Printed by the IAEA in Austria
September 2017
STI/PUB/1745

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.

Title: Safety of nuclear fuel cycle research and development facilities / International Atomic Energy Agency.

Description: Vienna : International Atomic Energy Agency, 2017. | Series: IAEA safety standards series, ISSN 1020-525X ; no. SSG-43 | Includes bibliographical references.

Identifiers: IAEAL 17-01104 | ISBN 978-92-0-103116-7 (paperback : alk. paper)

Subjects: LCSH: Nuclear facilities — Safety measures. | Hazardous substances — Safety measures. | Research and development projects.

Classification: UDC 621.039 | STI/PUB/1745

FOREWORD

by Yukiya Amano
Director General

The IAEA's Statute authorizes the Agency to “establish or adopt... standards of safety for protection of health and minimization of danger to life and property” — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

Safety measures and security measures¹ have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

Safety Fundamentals

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

Safety Requirements

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

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

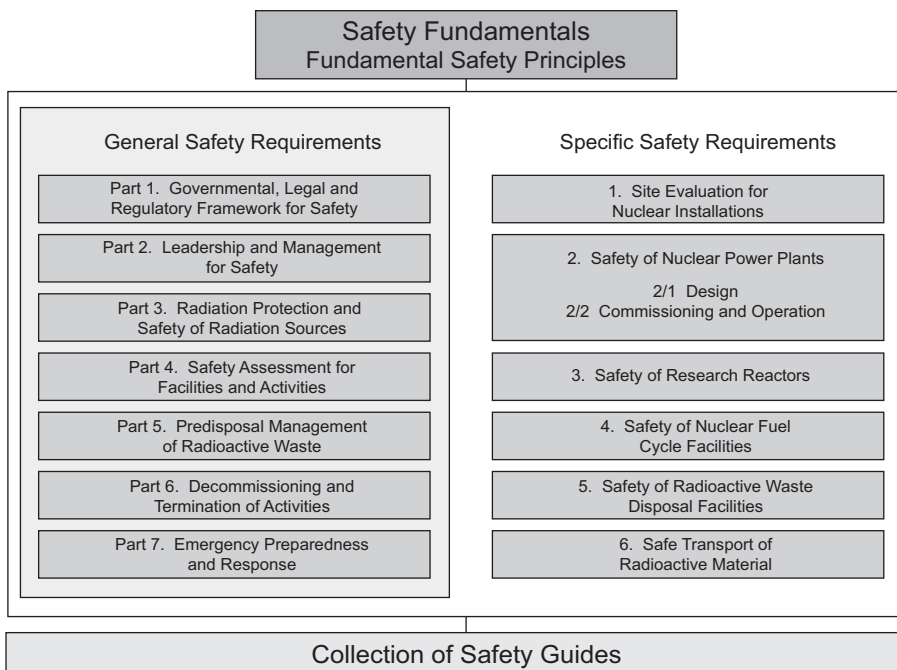


FIG. 1. The long term structure of the IAEA Safety Standards Series.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

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

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

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of

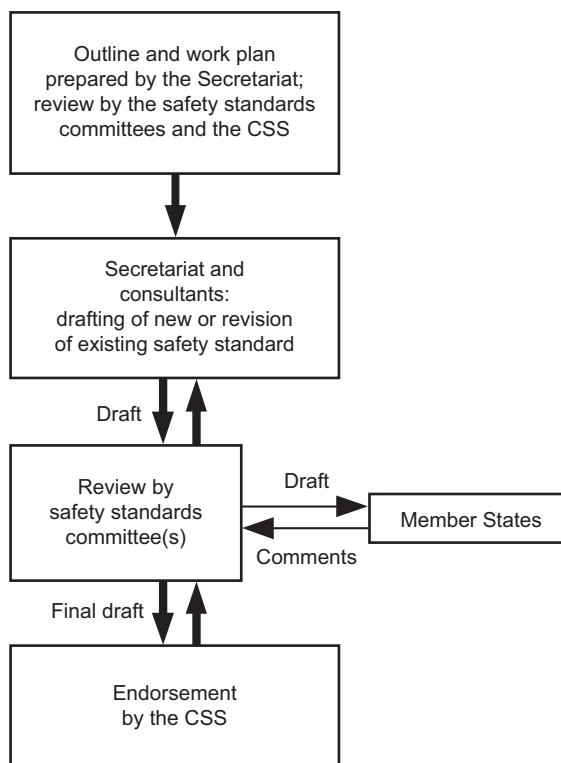


FIG. 2. The process for developing a new safety standard or revising an existing standard.

the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international

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

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see <http://www-ns.iaea.org/standards/safety-glossary.htm>). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

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

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

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

CONTENTS

1.	INTRODUCTION	1
	Background (1.1–1.6)	1
	Objective (1.7–1.8)	2
	Scope (1.9–1.14)	3
	Structure (1.15–1.16)	4
2.	GENERAL SAFETY CONSIDERATIONS FOR R&D FACILITIES	5
	General (2.1–2.2)	5
	Licensing of an R&D facility (2.3–2.15)	6
	Management system (2.16–2.24)	9
3.	SITE EVALUATION (3.1–3.4)	10
4.	DESIGN	11
	General (4.1–4.10)	11
	Safety functions (4.11–4.48)	14
	Postulated initiating events (4.49–4.99)	23
	Instrumentation and control (4.100–4.107)	36
	Human factor considerations (4.108–4.111)	39
	Safety analysis (4.112–4.125)	41
	Emergency preparedness and response (4.126–4.128)	44
	Management of radioactive waste (4.129–4.132)	45
	Other design considerations (4.133–4.139)	47
5.	CONSTRUCTION (5.1–5.8)	49
6.	COMMISSIONING (6.1–6.10)	50
7.	OPERATION	53
	Characteristics of R&D facilities (7.1–7.4)	53
	Qualification and training of personnel (7.5–7.9)	54
	Facility operation (7.10–7.20)	54
	Maintenance and periodic testing (7.21–7.26)	57

Control of modifications (7.27–7.32)	58
Criticality safety (7.33–7.36)	59
Radiation protection (7.37–7.55)	61
Industrial and chemical safety (7.56–7.59).	65
Management of radioactive waste (7.60–7.67).	66
Emergency preparedness and response (7.68–7.71).	68
 8. PREPARATION FOR DECOMMISSIONING (8.1–8.4)	 68
Preparatory steps (8.5–8.6).	70
Decommissioning process (8.7–8.9).	70
 REFERENCES	 73
 ANNEX I: PROCESS ROUTE IN AN R&D FACILITY: LABORATORY SCALE (CASE 1)	 77
 ANNEX II: PROCESS ROUTE IN AN R&D FACILITY: PILOT SCALE (CASE 2).	 78
 ANNEX III: STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS	 79
 ANNEX IV: EXAMPLES OF OPERATIONAL LIMITS AND CONDITIONS	 91
 CONTRIBUTORS TO DRAFTING AND REVIEW	 93

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide on the safety of nuclear fuel cycle research and development (R&D) facilities supplements the Safety Requirements publication on the Safety of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. NS-R-5 (Rev. 1) [1], including appendix V, which specifically covers R&D facilities. It addresses all the stages in the lifetime of R&D facilities, whether they are at the laboratory, pilot or demonstration scale, from design through to preparations for decommissioning.

1.2. R&D facilities receive, handle, process and store various nuclear materials including uranium, other actinides and fission products, and activated materials in multiple physical forms such as powders, liquids and gases. These can present diverse hazards such as:

- (a) Nuclear and radiological hazards;
- (b) Toxic hazards from bioactive or chemical materials (e.g. hydrofluoric acid, uranium hexafluoride or ammonia);
- (c) Explosive or flammable hazards from reactive materials (e.g. hydrogen, nitric acid, metallic powders).

1.3. Another feature of many R&D facilities is the diversity of research and operating personnel, organized in different teams with potentially different training, expertise, experience, expectations and goals. This may lead to situations where hazards are not properly recognized and controlled. This Safety Guide applies to the two classes of R&D facility described below and illustrated in Annexes I and II. It also applies to the experiments (activities) undertaken within these facilities, using a graded approach:

- Case 1: Small scale experiments, analyses and fundamental research studies conducted on the chemical, physical, mechanical and radiological properties of specific materials such as prototype nuclear fuels (before and after reactor irradiation) and investigations of nuclear materials and wastes arising from new processes;
- Case 2: R&D on processes and equipment envisaged for use on an industrial scale (e.g. pilot facilities for waste treatment).

1.4. R&D facilities can operate over extended periods of time to provide analytical services, materials and testing services, and the inventories of radioactive and toxic materials in such facilities can be significant. Consequently, all the relevant safety requirements for the management of nuclear fuel cycle facilities and activities, such as learning from experience, inspection and maintenance, apply to R&D facilities. The relevant safety requirements for specific types of facility also apply to Case 2 R&D facilities where similar operations are carried out.

1.5. R&D facilities may support all stages of the nuclear fuel cycle, from fundamental research to applied research, fuel processing, material examination and fuel safety, chemical analysis and the development of instrumentation. A variety of physicochemical processes may be employed to study different types of fuels or materials that may also be hazardous. Particular care should be taken when researching new or novel processes and when establishing the safety of developing processes, to ensure that the safety assessment and safety measures are appropriate to the state of knowledge. The normal practice of eliminating unknown factors relating to safety is not always possible in some R&D activities. In such cases the approach taken should involve additional margins of safety and a more cautious application of the graded approach.

1.6. R&D facilities are subject to the same international agreements and national laws as other types of nuclear facility.

OBJECTIVE

1.7. The objective of this Safety Guide is to provide up to date guidance on engineering actions, conditions and procedures to meet the requirements established in NS-R-5 (Rev. 1) [1] based on experience gained in Member States. This Safety Guide is intended to be of use to researchers, designers, operating organizations and regulatory bodies for ensuring the safety of R&D facilities.

1.8. In this Safety Guide, the operating personnel, researchers, contractors and subcontractors working at the R&D facility are collectively referred to as 'R&D facility personnel', or simply 'personnel'. More specific terms may be used where a distinction is necessary.

SCOPE

1.9. This Safety Guide provides guidance on meeting the safety requirements in NS-R-5 (Rev. 1) [1]. Sections 5–10 of NS-R-5 (Rev. 1) [1] establish requirements common to all nuclear fuel cycle facilities, i.e. engaged in milling, refining, conversion, enrichment, fabrication of fuel, reprocessing of spent fuel, waste treatment and storage and R&D facilities. Appendix V of NS-R-5 (Rev. 1) [1] establishes the requirements that are specific to R&D facilities.

1.10. This Safety Guide does not apply to irradiators, accelerators, research reactors, subcritical assemblies or radioisotope production facilities. It focuses specifically on the safe design, construction, commissioning, operation and decommissioning of R&D facilities. The scope of this Safety Guide is limited to the safety of the R&D facility, the protection of workers and the public and the management of any wastes generated. It does not address any subsequent impacts that the material produced by R&D facilities may have on end users.

1.11. Guidance on meeting the requirements for the management system established in Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2 [2], is provided in Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1 [3] and The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5 [4]. Safety requirements for the legal and governmental framework and regulatory supervision (e.g. requirements for the authorization process, regulatory inspection and regulatory enforcement) are established in Governmental, Legal and Regulatory Framework for Safety, IAEA Safety Standards Series No. GSR Part 1 (Rev. 1) [5].

1.12. Safety guidance relevant to Case 2 R&D facilities can also be found in the IAEA Safety Guides for the corresponding type of nuclear fuel cycle facilities, e.g. guidance applicable to fuel fabrication pilot facilities will also be found in the Safety Guide for fuel fabrication facilities, Safety of Uranium Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-6 [6].

1.13. This Safety Guide includes guidance on radiation protection measures to meet the safety requirements specified in Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3 [7]. GSR Part 3 [7] and the associated guidance in Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7 [8], also present measures for personnel dosimetry, optimization of protection, measures to control and limit the discharge of radioactive materials

to the environment and radiation monitoring of the workplace as well as contamination monitoring of workers.

1.14. This Safety Guide provides examples of the application of a graded approach to nuclear fuel cycle R&D facilities. The graded approach in itself is a requirement in many IAEA standards, e.g. Requirement 1 of Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1) [9], and Requirement 6 of GSR Part 3 [7]. Application of a graded approach ensures that safety measures and safety related activities are proportionate to the hazards of a facility.

STRUCTURE

1.15. This Safety Guide contains guidance specific to nuclear fuel cycle R&D facilities based on relevant IAEA safety requirements cited in this publication. The recommendations in this guide have been referenced to the corresponding requirements, where consistent with the readability of the text. This Safety Guide covers all stages in the lifetime of an R&D facility, including site evaluation, design, construction, commissioning, operation and decommissioning. It also provides specific guidance on modifications, maintenance, calibration, testing and inspection as well as emergency preparedness, where such guidance is appropriate.

1.16. General safety guidance for an R&D facility is provided in Section 2. The safety aspects to be considered during the process of evaluating the site for a facility are described in Section 3. Section 4 deals with safety in the design stage and Section 5 deals with safety aspects in the construction stage. Section 6 describes the safety considerations that arise during commissioning. Section 7 contains guidance on practices to ensure safety during facility operation. It also covers the management of facility operations and emergency preparedness and response. Section 8 provides guidance on meeting safety requirements in the decommissioning of an R&D facility. Annexes I and II show the typical process route for the two classes of R&D facilities covered by this guidance. Annex III gives examples of structures, systems and components (SSCs) important to safety in R&D facilities, grouped by process areas. Examples of operational limits and conditions for R&D facilities are provided in Annex IV.

2. GENERAL SAFETY CONSIDERATIONS FOR R&D FACILITIES

GENERAL

2.1. In R&D facilities, fissile material and other radioactive materials can be present in different forms with diverse physical and chemical characteristics. The main hazards are criticality, loss of confinement, radiation exposure (both internal and external), fire, chemical and explosive hazards, from which workers, the public and the environment need to be protected by adequate design, construction and safe operation, as required by NS-R-5 (Rev. 1) [1].

2.2. The factors affecting the safety of R&D facilities include the following:

- (a) The radiological consequences caused by the release of radioactive materials under accident conditions can be significant.
- (b) Fissile material (if present) has the potential to achieve criticality under certain conditions. The subcriticality of a system depends on many parameters, including the fissile mass, concentration, volume, density, geometry and isotopic composition. Subcriticality is also affected by the presence of other materials, such as neutron absorbers, moderators and reflectors; see Criticality Safety in the Handling of Fissile Material, IAEA Safety Standards Series No. SSG-27 [10].
- (c) When irradiated fuel is used, the radiation levels and the risk of internal and external radiation exposures are significantly increased.
- (d) The chemical toxicity of material used in R&D facilities has to be considered (e.g. uranium hexafluoride, which if released, reacts with the moisture in the air to form hydrogen fluoride and soluble uranyl fluoride). Therefore, the safety analysis of an R&D facility should also address impacts resulting from these chemicals and their potential mixing (e.g. in waste or liquid releases).
- (e) The presence of products, sub-products or waste arising from R&D programmes on exotic nuclear materials, such as those listed below, which should be included in safety assessments:
 - (i) Non-standard mixed oxide (MOX) or uranium dioxide fuel fabrication, or new fuel matrices, e.g. carbides, nitrides, metallic forms;
 - (ii) Isotopes with particular constraints for disposal, e.g. long half-life transuranics (such as curium), fission products (such as ⁹⁹Tc) and activated materials such as trace materials in cladding;

- (iii) Materials without an agreed national disposal route, e.g. graphite and aluminium in waste;
- (iv) Uranium with enrichment levels greater than 5%;
- (v) Materials in the thorium fuel cycle that contain high-energy gamma emitters such as ^{232}U .

LICENSING OF AN R&D FACILITY

2.3. A complete set of national safety regulations should be developed and implemented to ensure that the safety of an R&D facility is maintained for its full lifetime; see Section 3 of NS-R-5 (Rev. 1) [1]. The regulatory body should establish the basic requirements for protection of workers and members of the public against the hazards of the R&D facility (e.g. based on assessments of the doses arising from normal operations and postulated accidents). These requirements should be consistent with internationally agreed approaches.

2.4. The licensing of an R&D facility should be based on a complete and adequate safety case produced by suitably qualified personnel. This safety case should include the safety analysis report, any operational limits and conditions and a listing of the safety procedures to be followed. The safety analysis report should consider safety during normal operations and in the event of accidents. Postulated initiating events should be analysed to ensure that accidents are adequately prevented and detected and that their consequences are mitigated. Detailed requirements for the licensing documentation¹ are established in Sections 2 and 9 of NS-R-5 (Rev. 1) [1].

2.5. Requirement 23 of GSR Part 4 (Rev. 1) [9] states that:

“The results of the safety assessment shall be used to specify the programme for maintenance, surveillance and inspection; to specify the procedures to be put in place for all operational activities significant to safety and for responding to anticipated operational occurrences and accidents; to specify the necessary competences for the staff involved in the facility or activity and to make decisions in an integrated, risk informed approach.”

¹ In the context of fuel cycle facilities, the licensing documentation (or safety case) is a collection of arguments and evidence in support of the safety of a facility or activity. This will normally include the findings of a safety assessment and a statement of confidence in these findings. ‘Safety case’ is the same as ‘licensing documentation’ and these titles are used interchangeably in this Safety Guide.

Licensed operations are required to be conducted as defined in the safety case, including the operational limits and conditions. The management team of the R&D facility should be trained on the content and use of the safety analysis report and operational limits and conditions, in accordance with GS-G-3.5 [4].

2.6. Through the licensing process, the operating organization is required to involve the regulatory body in the case of new research programmes that are outside the scope of the existing safety case for the R&D facility, in accordance with national practices for the authorization of modifications.

2.7. The licensing documentation should be sufficiently broad in scope to capture the anticipated development of R&D programmes and take account of the additions and changes to safety requirements that could be expected. Nevertheless, the definition of licensing scope should be sufficiently detailed to ensure clarity of the controls necessary for protection and safety.

2.8. The safety approach (as documented in the safety analysis report) for an R&D facility should provide the same level of safety assurance, irrespective of whether small scale academic research is conducted at the R&D facility or the R&D facility is a large nuclear pilot plant. This equivalence of level is achieved with the application of a graded approach.

2.9. When shutting down or restarting parts of an existing R&D facility, the safety assessment of the facility should be reviewed and updated, addressing any ageing or obsolescence issues, and should cover potential legacy waste and decommissioning needs as far as is achievable. Radioactive material or hazardous materials, including any registered radioactive sources, should be relocated to safe storage before parts of an R&D facility are closed down.

2.10. In accordance with para. 3.9(e) of GSR Part 3 [7], an environmental impact assessment is required to be carried out by the operating organization as part of the licensing process for the R&D facility. The prospective assessment for radiological environmental impacts is required to be commensurate with the magnitude of the possible radiation risks arising from the R&D facility.

2.11. Paragraph 9.35 of NS-R-5 (Rev. 1) [1] states that “The operating organization shall establish a process whereby its proposals for changes ... are subject to a degree of assessment and scrutiny appropriate to the safety significance of the change...” and an R&D facility should be subject to a change management process in the same way as other nuclear facilities are. When there is a change in the use of an R&D facility (or part of it), an appropriate modification

programme should be implemented, with peer review by suitably qualified personnel. Where the increase in scale is large, the operating organization should plan the increase in stages where possible, in order to permit the gathering of feedback and the validation of each stage. Guidance on the configuration and audit of such changes is provided later in this Safety Guide.

2.12. The licensing documentation should also take into account the arrangements for radioactive waste management during operation and for decommissioning.

2.13. The licensing documentation should demonstrate that arrangements for emergency preparedness and response are in place and are commensurate with the hazards associated with the facility in accordance with Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7 [11], and Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1 [12].

2.14. Paragraph 4.26 of NS-R-5 (Rev. 1) [1] states that:

“In accordance with the national regulatory requirements, the operating organization shall carry out periodic safety reviews to confirm that the licensing documentation remains valid and that modifications made to the facility, as well as changes in its operating arrangements or utilization, have been accurately reflected in the licensing documentation. In conducting these reviews, the operating organization shall expressly consider the cumulative effects of changes to procedures, modifications to the facility and the operating organization, technical developments, operating experience and ageing.”

This requirement applies to R&D facilities because these facilities can operate for a long time and may also be subject to many modifications and changes of use.

2.15. The interfaces between security, safeguards and safety should be taken into account in the regulation of an R&D facility during all phases of its lifetime, not only during the siting phase.

MANAGEMENT SYSTEM

2.16. In accordance with the requirements of para. 4.5 of NS-R-5 (Rev. 1) [1], the overall responsibility for the safety of the R&D facility rests with the operating organization. Paragraph 4.7 of NS-R-5 (Rev. 1) [1] also states that:

“The operating organization shall clearly specify the responsibilities and accountabilities of all staff involved in conducting or controlling operations that affect safety. The person with the responsibility for direct supervision shall be clearly identified at all times.”

These management processes and organizational provisions should also reflect the requirements of GSR Part 2 [2].

2.17. These processes and provisions apply throughout the lifetime of the facility, from its siting to its decommissioning, and to operations, maintenance and experiments.

2.18. Leadership in the facility should encourage and reinforce a learning and questioning attitude at all levels of the organization, while maintaining a conservative approach to decision making. This is an important contribution to safety culture that should be maintained by adequate training and by example. Requirements relating to leadership for safety and safety culture are established in GSR Part 2 [2].

2.19. Operating organizations of R&D facilities and the regulatory body should promote the sharing of feedback on operating experience on safety with other R&D facilities worldwide. Whether a full scale plant or individual experiments, the operating organization should make use of such feedback as far as practicable.

2.20. The operating organization should develop and promote the attributes of a strong safety culture among all workers and researchers. These attributes should include a questioning attitude and challenging assumptions with the goal of maintaining and improving safety performance.

2.21. R&D facilities should take advantage of any existing infrastructural support at the site. In emergency planning and preparedness, account should be taken of all other facilities at the site, their interactions and their ability to support the R&D facility.

2.22. Due consideration should be given to the minimization and processing (i.e. pretreatment, treatment and conditioning) of radioactive waste that will be generated during the operation and decommissioning of the R&D facility, as well as any legacy material.

2.23. The safety of any existing R&D facility should be reassessed and, if necessary, the facility should be modified to meet current (or updated) safety standards as far as reasonably achievable. As an alternative, equivalent compensatory measures should be provided.

2.24. In an R&D facility, the use of remote handling operations, adequate shielding and confinement of contaminated atmospheres should be considered to reduce occupational exposures and to ensure safe operations, especially in experiments using highly toxic materials or highly radioactive materials.

3. SITE EVALUATION

3.1. Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3 (Rev. 1) [13], establishes requirements for the evaluation of sites for most land based nuclear installations including nuclear fuel cycle facilities. The site evaluation process for an R&D facility may involve a large number of criteria, some of which are specific to the site and others that are related to the facility. At the earliest stage of planning for an R&D facility, a list of these criteria should be prepared, considered in accordance with their safety significance and agreed with the regulatory body. In most cases, it is unlikely that all the criteria can be met, and the risks posed by certain externally generated initiating events (e.g. earthquake, aircraft crash, fire, extreme weather conditions and floods) and the resulting consequences will dominate the choice of a site. Guidance on the safety criteria used in this process is provided in: Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-18 [14]; Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 [15]; Volcanic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-21 [16]; and External Human Induced Events in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.1 [17].

3.2. An R&D facility may be a stand-alone facility, in which case the site should be capable of supporting the necessary infrastructure (e.g. for off-site

emergency response). However, many R&D facilities are a part of another site for which criteria for site evaluation have already been determined. Interactions with facilities nearby should be considered, as follows:

- In the case of an existing nuclear facility, the criteria will normally be encompassed by the evaluation studies of the existing facility.
- In the case of a non-nuclear site (e.g. a hospital, university or research centre), the main siting issue can be the feasibility of the necessary emergency arrangements, such as the arrangements for evacuation. This may require specific design provisions or other emergency provisions in order to meet the requirements of GSR Part 7 [11] and GS-G-2.1 [12].

3.3. Requirements for the evaluation of a site for an R&D facility are provided in NS-R-3 (Rev. 1) [13]. Where the facility is a pilot for a nuclear fuel cycle facility of another type, reference should also be made to the relevant specific safety guides, e.g. SSG-6 [6]; Safety of Conversion Facilities and Uranium Enrichment Facilities, IAEA Safety Standards Series No. SSG-5 [18]; and Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-7 [19].

3.4. The siting of an R&D facility should take into account any nuclear security threats and allow the implementation of physical security measures in accordance with the recommendations and guidance provided in the IAEA Nuclear Security Series publications [20, 21].

4. DESIGN

GENERAL

4.1. The SSCs, management system and procedures for an R&D facility should be designed in an integrated manner that ensures safe operation, prevents events that could compromise safety and mitigates the consequences of such events were they to occur. This design process usually begins with an analysis of potential internal initiating events (or faults) and external initiating events. It should proceed to the identification of safety functions that provide defence in depth, usually within boundaries defined by operational limits and conditions or limits of the safety case.

4.2. For the implementation of the defence in depth requirements (section 2 of NS-R-5 (Rev. 1) [1]), the first two levels are the most significant, as the risks are mainly eliminated by design and appropriate operating procedures (see sections 4, 6 and 7 of NS-R-5 (Rev. 1) [1]). However, all levels of defence in depth should be considered during the design and safety analysis process.

Main safety functions for R&D facilities

4.3. The main safety functions (see paras 6.37–6.53 and V.1–V.10 of NS-R-5 (Rev. 1) [1]) are functions, the loss of which may lead to criticality, radioactive or chemical releases or exposures with possible radiological consequences for workers, the public or the environment, namely:

- (1) Prevention of criticality;
- (2) Confinement of potentially harmful material and the removal of decay heat;
- (3) Protection against external radiation exposure.

4.4. Releases of radioactive, toxic or biologically active materials are all potentially harmful. The safety measures identified in the design of the R&D facility should comprise those individual items important to safety and operational limits and conditions which, when taken as a whole, provide the main safety functions above. The remainder of this section describes those accidents, events and particularly those safety functions that may be especially relevant to an R&D facility.

Specific engineering design requirements

4.5. The following specific engineering design requirements in NS-R-5 (Rev. 1) [1] apply for each of the main safety functions:

- (a) The requirements on prevention of criticality as established in paras 6.43–6.51 and V.4–V.6;
- (b) The requirements on confinement of radioactive materials as established in paras 6.37–6.39, 6.52 and V.7;
- (c) The requirements on protection against exposure, as established in paras 6.40–6.42 and V.8.

4.6. The design should give consideration to the handling of various types of radioactive materials. Owing to the nature of the work done in R&D facilities, there are often design and engineering provisions for flexibility and adaptation to anticipate future requirements or dismantling. These provisions should:

- (a) Be designed to enhance safety;
- (b) Give particular consideration to the potential for ageing and degradation of items important to safety;
- (c) Be operated to ensure safety is maintained over the lifetime of the facility;
- (d) Not be used for unassessed materials without a modification proposal or safety assessment.

Design basis accidents and safety analysis

4.7. In the context of nuclear fuel cycle facilities, anticipated operational occurrences and design basis accidents and their equivalents present challenges against which a facility is designed according to established design criteria such that the consequences are kept within defined limits. The specific safety requirements relating to anticipated operational occurrences and design basis accidents (or equivalent) are established² to ensure that the design keeps radiation exposures from normal operation and accident conditions as low as reasonably achievable. SSG-18 [14], SSG-9 [15] and SSG-21 [16] provide guidance on specific hazards of potential relevance.

4.8. In addition to the radiological hazards outlined above, particular consideration should be given to the following hazards:

- (a) Internal and external human induced phenomena such as fire, chemical explosion and aircraft crashes;
- (b) Natural phenomena such as earthquakes, tsunamis, flooding and tornadoes;
- (c) Human errors and organizational failings;
- (d) Chemical and toxic releases [22].

4.9. The analysis should take account of events that might be consequences of other events, such as a flood following an earthquake, or multiple events initiated by one external event, such as fire or multiple leaks within the facility caused by an earthquake.

Structures, systems and components important to safety

4.10. The design measures identified by the safety analysis are intended to prevent any abnormal situation where the safety margin has been reduced, to detect this situation and to mitigate its consequences should it progress further.

² See paras 6.4–6.9, V.1 and III–10 of NS-R-5 (Rev. 1) [1].

These measures are often implemented by means of SSCs important to safety, which are also known as items important to safety; see paras 6.6 and 6.8–6.12 of NS-R-5 (Rev. 1) [1]. Annex III presents examples of representative safety functions and their associated SSCs.

SAFETY FUNCTIONS

Prevention of criticality

General

4.11. For R&D facilities, criticality prevention should be addressed through strict compliance with paras 6.45 and 6.49 of NS-R-5 (Rev. 1) [1]. In addition, Case 2 R&D facilities should meet the requirements in appendices I, II, III or IV of NS-R-5 (Rev. 1) [1], which establish requirements applicable to specific types of pilot facility (e.g. for a pilot MOX facility handling fissile material, the requirements in appendix II of NS-R-5 (Rev. 1) [1] apply). In many R&D facilities handling fissile materials, prevention of criticality by means of mass control is used as a deterministic safety measure that is not usually available in full scale facilities. As far as possible, the control by mass in an area should be independent of all other factors. A number of such areas may coexist independently in a single facility with suitable interface controls. The rest of this section describes the basis for control by mass and other factors in more detail and concludes with guidance regarding the detection of criticality incidents.

Design for criticality prevention

4.12. Paragraph 6.45 of NS-R-5 (Rev. 1) [1] establishes requirements for all types of nuclear fuel cycle facilities in which criticality is considered: “For the prevention of criticality by means of design, the double contingency principle ... shall be the preferred approach” and para. 6.47 states that “Criticality evaluations and calculations shall be performed on the basis of making conservative assumptions.” When the requirements for a specific pilot facility type are not applicable, the requirements for the prevention of criticality in paras V.1, V.4 and V.5 of NS-R-5 (Rev. 1) [1] should be used. Some examples of the parameters that should be controlled to prevent criticality include the following:

- (a) Mass: In R&D facilities, mass margins³ should be based on a representative material with the lowest critical mass. The margin should be not less than 100% of the normal value in operation (unless the likelihood of double batching is demonstrated to be sufficiently remote), or a mass margin equal to the physical mass that can be accumulated.
- (b) Geometry or shape: The analysis should give consideration to the layout of the facility, the dimensions and locations of pipes, vessels and other laboratory equipment. For example, control by geometry could be used in the design of furnaces and dissolvers.
- (c) Density and forms of materials: The analysis should consider the range of densities for different forms of materials (e.g. powder, pellets or rods) used in an R&D facility.
- (d) Concentration and density in analytical laboratories and in liquid effluent units: The analysis should consider the range of fissile material in solution as well as any potential precipitates (e.g. recovery of Pu in waste streams).
- (e) Moderation: The analysis should consider a range of moderation to determine the most reactive conditions that could occur. Water, oil and similar hydrogenous substances are common moderators that are present in R&D facilities, or may be present under accident conditions (e.g. water from firefighting; see para. V.6 in NS-R-5 (Rev. 1) [1]). The possibility of non-homogenous distributions of moderators with fissile material should be considered (e.g. organic binders and porosity enhancing agents used in the pelletizing process).
- (f) Moisture content in powder material: The analysis should consider the range of moisture content for powder material used in an R&D facility.
- (g) Reflection: The most conservative margin of those resulting from different assumptions should be retained, such as: (i) a hypothetical thickness of water around the processing unit; and (ii) consideration of the actual neutron reflection effect due to, for example, the presence of personnel, organic materials, shielding materials, concrete or steel of the containment in or around the processing unit.
- (h) Neutron absorbers: If claims are made for neutron absorbers in the safety analysis, their effectiveness should be verified depending on the relevant operating conditions identified in appendices I–IV in NS-R-5 (Rev. 1) [1]. Neutron absorbers such as cadmium and boron may be used in R&D facilities and the safety analysis should address their effect as neutron absorbers; however, ignoring their effects would still yield conservative

³ The mass margin is: the difference between the safety limit (the maximum amount allowed within the operational limits and conditions) and the subcritical limit (effective neutron multiplication factor $k_{\text{eff}} < 1$, often taken as $k_{\text{eff}} < 0.95$).

results. The use of mobile or easily displaced or removed solid absorbers should be avoided.

- (i) Neutron interaction: Consideration should be given to neutron interaction between fissile material in all locations in the R&D facility and all potential locations that may be involved. Specific consideration should be given to the layout of the R&D facility and any possible changes. Physical locators are preferred to floor markings as a means of indicating or ensuring the placement of equipment with potential neutron interactions.
- (j) Fissile content: For any fissile material (e.g. fresh or irradiated fuel), the maximum fissile content (e.g. level of enrichment) in any part of the facility should be used in all assessments unless the extreme improbability of having this isotopic composition in a particular area of the facility is demonstrated in accordance with the double contingency principle.

Criticality safety analysis

4.13. The criticality safety analysis should demonstrate that the design of equipment is such that the values of control parameters are always maintained in the subcritical range for all operational states (i.e. normal operation and anticipated operational occurrences) and during and after design basis accidents, or their equivalent. This should be achieved by determining the effective multiplication factor k_{eff} , which depends on the mass, the distribution and the nuclear properties of the fissile material and all other materials with which it is associated. The calculated value of k_{eff} should then be compared with the value specified by the design limit or national regulations, whichever is lower.

4.14. A number of methods can be used to perform criticality safety analysis, for example, the use of experimental data, reference books or recognized standards, hand calculations or calculation by means of deterministic or probabilistic computer codes. Any method used to carry out the analysis should use conservative data and assumptions and should be fully verified and validated for the application. For detailed guidance, see SSG-27 [10].

4.15. The method employed should be appropriate to the types of material being handled in the R&D facility. The general procedure to be followed in this analysis should include the use of the following:

- (a) A conservative approach that takes into account:
 - Uncertainties in physical parameters, optimum moderation conditions and possible non-homogenous distributions of moderators;
 - Anticipated operational occurrences and their combinations;

- Facility states that result from postulated external and internal initiating events.
- (b) Appropriate computer codes that are verified and validated (i.e. compared with benchmarks to determine the effects of code bias and code uncertainties on calculated k_{eff} values) within their applicable range and that use appropriate cross-section libraries. Detailed guidance is provided in paras 4.20–4.25 of SSG-27 [10].

4.16. For a process where fissile material is handled in a discontinuous manner (including batch processing or waste packaging), the process and its equipment should meet the safety requirements for criticality control at all times. The design of the R&D facility, including any support systems to account for and control nuclear material, should provide the necessary equipment for accounting and control and should have clear and easily identifiable boundaries. Care should be taken at the interface between two areas to ensure that transfers of fissile material meet criticality control requirements for both areas. The effect of potential delays in handover or associated checks should be considered in the safety analysis so that any negative consequences of accumulations of fissile material can be avoided.

Mitigation of criticality events

4.17. Information regarding the need to install criticality accident alarm systems can be found in Ref. [23]. Where such systems are installed, the R&D facility should be designed to include safe evacuation routes to personnel regrouping areas. These routes should be clearly marked and personnel should be trained in criticality evacuation procedures.

4.18. Consideration should be given to the provision of remote mitigation devices, for example, devices to empty a vessel containing the solution initiating the event or to absorb the neutron flux.

Protection of people against radiation exposure and protection of the environment

4.19. Protection against radiation exposure relies on an appropriate combination of controls on the magnitude of the source, the time of exposure and the shielding or distance between the subject and the source. These should be used separately or in combination.

4.20. Consideration should be given to maintenance, calibration, periodic testing and inspection, with the aim of minimizing the dose to workers. Requirements for the design of items important to safety to minimize exposure during maintenance are established in para. 6.19 of NS-R-5 (Rev. 1) [1]. Examples of such provisions include connection junctions at containment boundaries and easily cleanable surfaces.

4.21. The potential for accumulation of radioactive material in (a) process equipment; (b) fume hoods, gloveboxes and hot cells; and (c) secondary systems (e.g. ventilation ductwork) should be minimized and, where appropriate, provisions should be made for its removal or reduction.

4.22. Consideration should be given to the remote operation of services and experimental equipment where possible.

4.23. Requirements for the classification of areas for control of radiation and contamination are established in para. 6.41 of NS-R-5 (Rev. 1) [1]. This requirement may be graded to avoid excessive restriction on the movement of personnel. However, any grading should be justified as even small quantities of alpha active material can cause a significant contamination hazard.

4.24. Background radiation controls in R&D facilities often rely on analytical data from samples. If possible, an instrumental method of analysis that does not require sampling should be chosen. Where samples need to be taken, their number and sizes should be kept to a minimum consistent with providing sufficient, timely information for the optimization of protection and safety. The requirements for radiation protection during operation established in NS-R-5 (Rev. 1) [1], which include housekeeping, waste management and dose control, also apply to equipment and facilities used for sample analysis.

4.25. Paragraph 6.42 in NS-R-5 (Rev. 1) [1] states that “Radiation levels shall be monitored so that any abnormal conditions would be detected and workers may be evacuated. Areas of potential exposure for workers shall be appropriately identified and marked.” Radiation protection monitoring should be provided to ensure compliance with regulatory limits and international practices for exposure limitation, including the following:

- Fixed gamma/neutron monitors and stationary samplers for activity in air, (beta/gamma, alpha) for access and evacuation purposes;
- Mobile gamma/neutron area monitors and mobile samplers for activity in air, (beta/gamma, alpha), for evacuation purposes during maintenance;

- Personal monitoring consistent with the radiation type(s) present in the R&D facility.

4.26. All estimates of source terms should include allowance for the ingrowth of radioactive decay products (such as ^{241}Am) over the lifetime of the facility.

Confinement of radioactive materials

4.27. In accordance with paras 6.38 and V.7 of NS-R-5 (Rev. 1) [1], containment is required to be the primary method for protection against the escape of radioactive material. Static and dynamic confinements are both required as complementary containment systems:

- The static containment system should consist of at least two independent static barriers between radioactive material and the environment.
- A dynamic containment system can also be used to create airflow towards areas that are more contaminated.

4.28. Dynamic containment cannot be provided for some circumstances. Sealed containers and isolated equipment, for instance, cannot be directly connected to a ventilation system. Also, it is sometimes impossible to provide ventilation for maintenance operations in open areas. Task assessments should be performed to ensure the safety of workers and the public against an unexpected leakage or a release from a source in such circumstances. Closed or sealed items should be treated as contaminated, as indicated by their history, and appropriate precautions should be specified for their handling, opening or unsealing. Consideration should be given in the design to the provision of equipment capable of determining the levels of radioactivity inside such items. Waste containers and other possibly contaminated containers should be appropriately characterized and labelled at (and with) the time and place of origin to avoid unexpected contamination release. Labels and containers can be colour coded and the colours may be specified to match equipment and pipework. Labels and bar-codes can be etched onto the surface of containers. Materials used for labels, inks and glues should be compatible with the containers to which they are applied and should be long lasting, with any inks used being pigment based.

4.29. In R&D facilities, the control of decay heat should normally rely on limiting the inventory of radioactive material in locations such as hot cells and gloveboxes. Where there is a potential for overheating, engineered cooling systems should be provided, for example, in the interim storage of waste, and the possibility of chemical reaction at high temperature or high pressure in sealed