IAEA Safety Standards

for protecting people and the environment

Safety of Nuclear Fuel Cycle Research and Development Facilities

Specific Safety Guide

No. SSG-43

IAEA

International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

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

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

Information on the IAEA's safety standards programme is available on the IAEA Internet site

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

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

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The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

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SAFETY OF NUCLEAR FUEL CYCLE RESEARCH AND DEVELOPMENT FACILITIES

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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

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Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
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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 measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

Safety Fundamentals

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

Safety Requirements

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

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

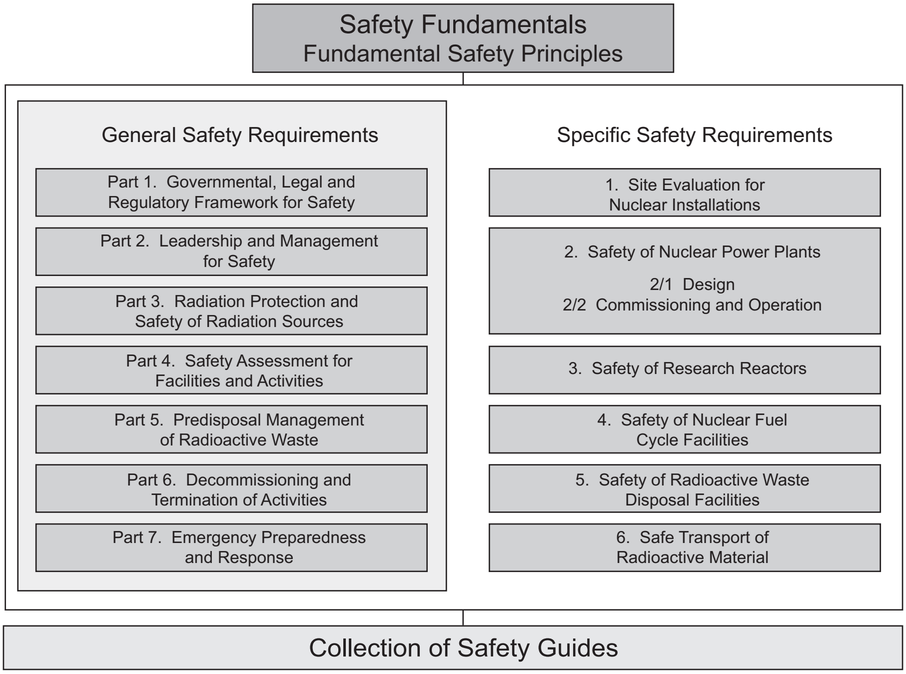

FIG. 1. The long term structure of the IAEA Safety Standards Series.

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

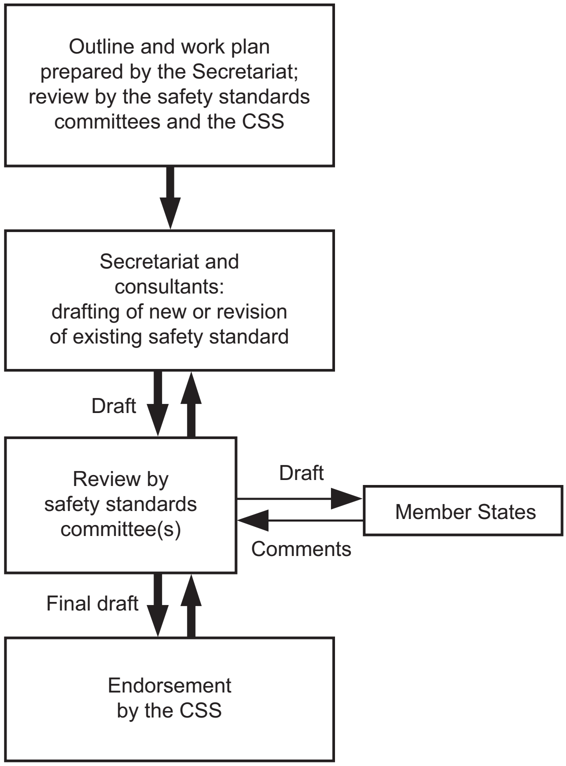

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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

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 ^{99}Tc) 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 documentation[1] 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."

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[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 established[2] 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, tsunami, 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 margins[3] 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 $k_{eff} < 1$, often taken as $k_{eff} < 0.95$).

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 containers should also be considered and provisions to manage this should be provided.

4.30. The first static barrier could include fume hoods, hot cells, gloveboxes, fuel cladding, vessels, pipework or other containers. The second static barrier should consist of rooms around the fume hoods, hot cells and gloveboxes, and/or the building walls. The design of the static containment should take into account typical openings between different confinement zones (e.g. doors, penetrations).

4.31. The dynamic containment should be used to create a pressure gradient (i.e. negative pressure) between the environment outside the building and the radioactive or hazardous material inside the fume hood, hot cell or glovebox. Backflow of gaseous or particulate contamination should be prevented. The exhaust air should be filtered (see para. 4.35).

4.32. Specific attention should be paid (particularly at the design stage) to maintaining containment during operations that involve the transfer of radioactive material through or out of the static containment. Where appropriate, equipment should be designed to withstand radiation damage and contamination by highly radiotoxic nuclides.

4.33. The design of confinement areas should include contamination monitoring devices covering all locations inside the R&D facility and outside the primary containment boundary provided by vessels, gloveboxes, fume hoods, pipework (and closures such as valves or blanking plates), ventilation ducting and the primary filters.

4.34. The design of the R&D facility should facilitate operations such as maintenance and decontamination. Consequently, the design should employ compartmentalization as one of the means available for the optimization of radiation protection.

4.35. Airborne contamination (from liquids or dispersible solids) should be prevented or minimized where possible. The ventilation system should include filters, in series, to protect workers, the public and the environment by filtering the air during normal operation and to ensure the integrity of the static barriers. Filters should also be used when airflow passes through confinement barriers, for example, at cooling inlets and where air exits the facility.

4.36. Where radioactive gases or particulates are generated, para. 6.38 in NS-R-5 (Rev. 1) [1] states that "the performance of air purification systems…

shall be commensurate with the degree of the potential hazards”. The materials of the ventilation system should be resistant to any corrosive gases present. The ventilation system should include a final monitoring stage and should be designed according to accepted standards, such as those of the International Organization for Standardization (ISO) and relevant national requirements.

4.37. The potential for the failure of a fully loaded filter should be considered. Additional standby fans and filters should be provided as specified in the safety analysis. These should be capable of maintaining ventilation during filter changing. Fans should be supplied with emergency power such that, in the case of a loss of electrical power, the standby ventilation system will begin operation within an acceptable period of time. The safety analysis should indicate what period of delay may exist between the loss of the primary ventilation system and initiation of the standby ventilation; this may define an operating limit or a condition. Local monitoring and alarm systems should be installed to alert operators to system malfunctions resulting in high or low flows or differential pressures. A detailed analysis should be undertaken for filters for which heavy use is planned.

4.38. To reduce risks relating to transfer operations involving radioactive material, the number of transfer operations should be minimized in the design of the facility. To reduce the complexity of transfer operations, R&D facilities should be designed to accommodate standardized means of transport of radioactive material, both on the site and off the site. Where possible, fixed equipment should be provided for the monitoring of such transfers.

Protection of workers from contamination and internal exposure

4.39. The first static barrier is normally the most important for protecting workers. Its design requirements should be specified to ensure and to control the efficiency of this barrier. Its design specifications should include specifications relating to: welding; choice of materials; effectiveness of confinement; ability to withstand seismic loads; design of equipment (including equipment for fume hoods, hot cells and gloveboxes); seals for electrical and mechanical penetrations; and the ability to perform inspections, maintenance and monitoring. For contained systems, leaktightness should achieve a high standard of confinement.

4.40. For fume hoods, gloveboxes and hot cells, the effectiveness of confinement is determined by the size of any openings and the air velocity at the face. The dynamic containment system should also be designed to minimize occupational exposure to hazardous material that may escape the first confinement barrier and be inhaled by workers.

4.41. At the design stage, provisions should be made for the installation of equipment to monitor airborne contamination. These should provide an immediate alarm on detection of airborne contamination with a low threshold. Monitoring points should be chosen that would best represent the normal and foreseeable accident exposures of personnel undertaking operations, experiments and other activities; see para. 6.39 in NS-R-5 (Rev. 1) [1] and paras 4.44–4.46 of this Safety Guide on protection against external radiation exposure.

4.42. Where radioactive powders or liquids are handled in the R&D facility or experiment, the installation of collection equipment (such as drip trays) should be considered to prevent the accidental spreading of radioactive material or hazardous material and control fissile geometry.

4.43. For normal operation, the need for use of respiratory protective equipment should be minimized through careful design of the static and dynamic containment systems.

Protection against external radiation exposure

4.44. The design of any radiation shielding should ensure compliance with targets relating to occupational exposure (see section 6 and para. V.1 of NS-R-5 (Rev. 1) [1]), on the basis of assumptions regarding the movement of material, occupancy time and sources to be handled. External radiation exposure can be controlled through a combination of source removal, reduction, distance, shielding and administrative controls. Provision of shielding should also be considered in storage areas. Application of the requirement for minimization of occupational exposure should also take maintenance workers into account.

4.45. In high radiation areas (such as those where spent fuel is handled), the protection of workers should rely primarily on shielding. In the design of the shielding, consideration should be given to both the inventory and the location of radioactive material, including deposited radionuclides. In medium or low activity areas (such as a teaching laboratory), a combination of shielding and administrative controls should be utilized for protection of workers from exposure to the whole body and to extremities. A general design guide is to shield as close to the source as is practical.

4.46. For the determination of radiological hazards, the potential for radiation from deposited radionuclides inside pipes, equipment, fume hoods, gloveboxes and hot cells should be taken into account. The interior surfaces of equipment such as gloveboxes should be made from non-absorbent material (such as stainless steel) or should be covered or coated to prevent accumulation of deposits from processed materials or their decay products. Shielding (or provisions to add shielding easily) should be considered where radioactivity may accumulate.

Environmental protection

4.47. R&D facilities should be designed so that effluent discharge limits can be met in normal operation and accidental releases to the environment are prevented. Paragraph V.7 in NS-R-5 (Rev. 1) [1] requires that a graded approach is taken to the provision of barriers for the confinement of radioactive materials, depending on the magnitude of the radiological hazard. Uncontrolled dispersion of radioactive substances to the environment from accidents can occur if a containment barrier is impaired. The barriers that provide environmental protection include rooms and the wider building structure. In addition, ventilation components that scrub or filter gases before discharge through a stack should be used to reduce all environmental discharges of radioactive material to acceptable levels.[4]

4.48. The design of an R&D facility should provide measures for continuous monitoring and control of the stack exhaust and for the monitoring of the environment around the facility. Further requirements on environmental protection that are also relevant to different pilot R&D facilities (Case 2) are established in paras I.9, II.14, III.9, IV.7 and IV.8 of NS-R-5 (Rev. 1) [1].

POSTULATED INITIATING EVENTS

4.49. Annex I of NS-R-5 (Rev. 1) [1] lists a number of postulated initiating events that could be applicable for an R&D facility, and further guidance on the related hazards is provided below. R&D facilities are often highly reliant on human operations; see paras 4.108–4.111. The systems that should be designed

[4] In this context, acceptability may include regulatory limits and considerations of the optimization of protection.

to continue operating in order to maintain the R&D facility and experiments in a safe state during and immediately after an event include the following:

(a) Heat removal systems in storage areas to remove decay heat from heat generating materials, and from heat producing experimental apparatus;
(b) Dynamic containment systems (i.e. ventilation), which should continue to operate to prevent leakage of radioactive material from the facility;
(c) Safety monitoring systems;
(d) Systems that provide chemical safety under high temperature conditions;
(e) Inert gas feed systems, for example, to hot cells or gloveboxes.

Internal hazards

Fire hazard analysis

4.50. R&D facilities should be designed to control fire hazards in order to protect R&D facility personnel, the public and the environment. Fire can lead to the dispersion of radioactive material and/or toxic materials by destroying the containment barriers (static and/or dynamic) or can cause a criticality accident by modifying the safe conditions of geometry, moderation or the control system. Fire hazards are often associated with the presence of flammable or combustible materials such as chemical reagents, electrical cabling and shielding. Fires affecting fume hoods, gloveboxes and hot cells can be particularly hazardous. A fire hazard analysis should be performed in order to identify appropriate measures that should be taken to ensure that fire is prevented and, if it occurs, that its consequences are mitigated while minimizing any resulting spread of radioactive material.

4.51. The fire hazards analysis should identify any areas that require special consideration. Locations subject to analysis should include the following:

(a) Areas where radioactive material is processed and stored;
(b) Facilities that process or produce radioactive material and/or other hazardous materials as a powder;
(c) Workshops, laboratories, and storage areas containing flammable and/or combustible liquids, solvents and resins and reactive chemicals, or involving mechanical treatment of pyrophoric metals or alloys (e.g. cuttings, shavings);
(d) Areas with high combustible loadings, for example, waste storage areas;
(e) Waste treatment areas, especially if incineration is used;

(f) Rooms housing safety related equipment, i.e. items such as air filtering systems and electrical switch rooms, whose degradation might have radiological or criticality consequences;
(g) Process control rooms and supplementary control rooms, where appropriate;
(h) Evacuation routes.

4.52. The fire hazards analysis should identify potential causes of fires, i.e. any fuels or oxidizing agents present. The potential consequences of fires should be assessed with, where appropriate, an estimation of the frequency or probability of the occurrence. The analysis should also assess the inventory of radioactive materials, ignition sources and combustible materials nearby, and should determine the adequacy of protective features.

4.53. Modelling may be used to support the fire analysis. Requirement 18 in GSR Part 4 (Rev. 1) [9] states "Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation." The results of modelling can provide valuable information on which to base decisions or to identify weaknesses that might otherwise have gone undetected. Even if the probability of a fire occurring may be low, a potential fire may have significant consequences with regard to nuclear safety and, as such, certain protective measures should be undertaken as described below.

4.54. Analysis of fire hazards should also include a review of the provisions made for preventing, detecting, mitigating and fighting fires.

Fire prevention, detection and mitigation

4.55. Prevention is the most important aspect of fire protection. R&D facilities should be designed to limit fire risks by taking measures to ensure that fires do not break out. Should a fire break out despite the precautions taken, measures should be in place to detect the fire and minimize its consequences.

4.56. For limiting the risks and consequences of a fire, a number of general and specific measures should be taken, including the following:

(a) The amount of flammable and combustible material in individual rooms, fume hoods, gloveboxes and hot cells should be minimized to the extent practicable.
(b) The storage of non-radioactive hazardous material should be separated from process areas.

(c) In gloveboxes and hot cells, where there is a high likelihood of fire (e.g. from cutting metal clad fuel elements), inert atmospheres with oxygen monitoring alarms should be used to minimize the risk of a fire spreading.
(d) Materials should be chosen according to functional criteria and fire resistance ratings.
(e) Buildings and ventilation ducts should be compartmentalized as far as possible in order to prevent spreading of fires. Buildings should be divided into fire areas. If a fire starts within a given fire area, its capability to spread beyond the area boundary should be eliminated or curtailed. The higher the fire risk, the greater the number of such fire areas a building should have. Utility lines penetrating fire compartment boundaries (e.g. electricity, gas or process lines) should be designed to ensure that fire does not spread.
(f) Ignition sources such as open flames or electrical sparks should be limited to the extent practicable (e.g. use of electrical earthing or grounding devices).
(g) Fire detection systems should be placed inside rooms where radioactive material is handled. Provision of detectors inside cells, gloveboxes and ventilation ducts should also be considered.
(h) Automatically or manually operated fire extinguishing devices using an appropriate extinguishing material should be installed in areas where a fire is possible and where the consequences of a fire could lead to the dispersion of contamination outside the first static barrier. Paragraph V.6 of NS-R-5 (Rev. 1) [1] states that "the choice of fire extinguishing media (e.g. water, inert gas or powder) and the safety of their use shall be addressed." The installation of automatic devices with water sprays should be carefully assessed for areas where fissile materials may be present, with account taken of the risk of criticality. Extinguishing gas may be preferable for fume hoods, gloveboxes and hot cells.
(i) Where extinguishing devices are installed inside fume hoods, gloveboxes or cells, the possible spread of contamination due to dynamic containment acting in reverse or due to uncontrolled water flows should be assessed.
(j) Where inert gas is used as a fire suppressant, account should be taken of the potential for operator asphyxiation and to the integrity of the gas supply by providing suitable alarms, backup or diversity.
(k) Where 'active' firefighting systems are used in radioactive environments, special consideration should be given during design to the requirements for their commissioning and subsequent examination, inspection, maintenance and testing.
(l) The design of ventilation systems should be given particular attention with regard to fire prevention. Dynamic containment comprises ventilation ducts and filter units, which may constitute weak points in the system unless they

are of suitable design. Fire dampers should be mounted in the ventilation system unless the frequency of occurrence of a fire spreading event is acceptably low. Such dampers should close automatically on receipt of a signal from the fire detection system, or by means of fusible links. Spark arrestors should be used to protect filters if necessary. The operational performance of the ventilation system should be specified so as to comply with fire protection requirements.

(m) Suitable monitoring equipment should be installed and the remote control of ventilation should be considered. Smoke particulates can lead to the rapid loading (blinding) of filters and consideration should be given to the need to provide dampers and other design means to reduce the challenge to filters in the event of a fire.

Explosions

4.57. A number of design requirements relating to chemical, toxic, flammable and explosive substances are established in para. 6.54 of NS-R-5 (Rev. 1) [1]. Examples of such materials in R&D facilities include: extraction solvents, hydrogen, hydrogen peroxide, nitric acid, degradation products and pyrophoric materials (e.g. metallic hydrides, dust or particles).

4.58. Consideration should also be given to the following:

(a) Fault scenarios such as leakage leading to contact between incompatible materials;
(b) The use of blow-out panels to mitigate the effects of explosions;
(c) Identification of parameters (e.g. concentration, temperature) to prevent situations leading to explosion;
(d) The use of inert atmospheres;
(e) Controlling levels of humidity.

4.59. In addition, effective airlocks should be provided between flammable atmospheres and other areas; see para. 6.55 in NS-R-5 (Rev. 1) [1].

Internal flooding

4.60. Flooding in R&D facilities can lead to dispersion of radioactive material and changes in the moderation of any fissile material present. Rainwater, groundwater, condensates and heating and cooling fluids are all capable of flooding a facility unexpectedly. Flooding, and even dew, can cause harm to equipment, including electrical damage and corrosion, and could infiltrate emergency supplies or fissile material. Recommendations relating to flooding by water in paras 4.61–4.63 are also applicable to any moderating fluid.

4.61. Where fissile material is present, a criticality assessment should be undertaken to determine the risk of condensation and flooding. The use of full disconnection from the water supply or limited water volumes should be considered and equipment should not have water supply connections during normal conditions unless the criticality assessment has taken into account the presence or leakage of water.

4.62. In R&D facilities where there are vessels and/or pipes with moderating fluids such as water, or where fissile material is stored, the criticality safety analyses should consider the presence of the maximum amount of fluid within the considered location, as well as in connected locations (e.g. via transfer tunnels).

4.63. The walls (and floors if necessary) of locations with the potential for flooding should be designed to withstand accidental flood loads and other items important to safety should not be affected by flooding (e.g. by means of installing sumps or floor drainage to remove water).

Leaks and spills

4.64. Leaks from equipment and components such as pumps, valves and pipes can lead to dispersion of radioactive material, fissile material, toxic chemicals and the creation of unnecessary waste. Leaks of hydrogenous fluids (water, oil, etc.) can change the neutron moderation in fissile material and reduce criticality safety. Leaks of flammable gases (H_2, natural gas, propane) or liquids can lead to explosions and/or fire. Leak detection systems should be used if such fluids are present.

4.65. Vessels containing significant quantities of fissile material in liquid form should be equipped with alarms to prevent overfilling and should be provided with drip trays configured to ensure criticality safety and of a capacity that equals or exceeds the volume of the vessel.

4.66. In-leakage of coolants should also be considered where there may be physical or chemical incompatibility with the materials or equipment present. The possibility of an unintended chemical reaction causing the precipitation of fissile material should be considered.

4.67. Spillage may occur from cans, drums and waste packages during transit within the R&D facility and in storage areas. Appropriate mechanical protection and containment should be provided during material movements.

Loss of support systems

4.68. To fulfil the requirement established in para. 6.28 of NS-R-5 (Rev. 1) [1], support systems of the R&D facility should be robust. Typical support systems include the electrical power supply, water supplies, compressed air supplies, ventilation and supplies of inert gases.

4.69. Electrical power supplies to R&D facilities should meet accepted industry codes and standards and the provision of diverse or remote electrical supplies should be considered. In the event of loss of normal power and depending on the status of the R&D facility, an emergency power supply should be available to certain SSCs important to safety, including the following:

(a) Ventilation fans and monitoring systems for the confinement of radioactive material;
(b) Heat removal systems;
(c) Emergency control systems;
(d) Fire detection and alarm systems;
(e) Monitoring systems for radiation protection;
(f) Alarm systems for criticality accidents.

4.70. The loss of general supplies such as gas for actuators of the instrumentation and control, water for process equipment and ventilation systems, heating, breathing air and compressed air may also have consequences for safety. In the design of an R&D facility, suitable measures to ensure safety should be provided. For example:

(a) Loss of gas supply to gas actuated safety valves and dampers: In accordance with the safety analysis, valves should be designed to fail to a safe position or an air reservoir should be provided.
(b) Loss of water or heating: Adequate backup capacity or a redundant supply should be provided for in the design.
(c) Loss of breathing air: Adequate backup capacity or a secondary supply should be provided to allow work in areas with airborne radioactive material to be terminated safely and workers to evacuate.

Loss or excess of process media

4.71. Consideration should be given to the loss and excess of process media or additives, which may have safety consequences. Examples include the following:

(a) The loss or excess of process gas supplies, for example, hydrogen, nitrogen, helium and argon;
(b) Overpressure in gloveboxes that may cause an increase of airborne contamination and/or concentration of hazardous materials;
(c) A release of large amounts of nitrogen, helium or argon in working areas that may result in a reduction of the oxygen concentration in breathing air.

Loss of heat removal

4.72. Consideration should be given to processes that generate heat and ventilation systems that require cooling. A loss of cooling can challenge the main safety functions by reducing the safety margin for confinement (and for criticality where fissile material is present). A large pilot plant can have significant heat loads and might be shut down quickly if there is a loss of a service such as power. The provision of an alternative means of cooling should be considered for heat generating materials and pilot plants with large heat sources.

4.73. Related functions of the ventilation system should be considered in the safety analysis, such as the maintenance of cooling to prevent heat stress to operating personnel or the control of humidity where materials are handled. These can have an indirect effect on the safety of operations.

Dropped loads

4.74. Requirement 10 of GSR Part 4 (Rev. 1) [9] requires an assessment that SSCs, including lifting equipment, are sufficiently robust. Potentially damaging dropped loads should be avoided by qualification of cranes, avoidance of clashes, provision of appropriate slings and grabs, handling at a low elevation and the training and qualification of relevant operators.

4.75. Mechanical or human failures during the handling of radioactive material may result in degradation of criticality control, confinement or shielding. Dropped loads are recognized as postulated initiating events and their possible consequences should be minimized (see para. IV.42 and annex I of NS-R-5 (Rev. 1) [1]). Mechanical or human failures during the handling of non-radioactive loads may cause a degradation of the safety functions of an R&D

facility. Safe travel paths should be provided and floors should be designed to withstand a dropped load. The design of hoisting devices should provide a high level of confidence that a load drop is extremely unlikely. Containers should be designed and qualified to maintain containment and to protect their contents wherever appropriate.

Mechanical failure

4.76. Measures for maintaining the integrity of commercially supplied equipment (e.g. mechanical guards) installed in the R&D facility should be retained. If there is a need to adapt such equipment to their nuclear environment, this should be justified.

4.77. Mechanical failures could result in damage (e.g. missiles, crushing, bending, breakage), which may result in degradation of criticality control, confinement or shielding. For complex or critical systems (e.g. rod handling systems designed to avoid the risk of breaking the rod), a systematic failure analysis should be carried out.

Radiolysis hazard

4.78. A number of chemical processes can be affected by radiolysis, potentially generating secondary hazards. Irradiation of organic or hydrated substances by radioactive material can lead to gas generation, especially of hydrogen. Radiolysis risks should be taken into account in the safety analysis for the following:

— Liquid effluents and organic solvents used in the laboratory;
— Contaminated oil and flammable waste;
— Process scraps enclosing hydrogenated additives.

Where necessary, the design should prevent or mitigate the effects of hazards associated with radiolysis.

External hazards

4.79. As stated in para. 6.21 of NS-R-5 (Rev. 1) [1],

> "SSCs important to safety shall be designed to withstand the effects of extreme loadings and environmental conditions (e.g. extremes of temperature, humidity, pressure, radiation levels) arising in operational states and in relevant design basis accident (or equivalent) conditions."

The R&D facility design should take account of operating experience regarding the effects of extreme loadings due to these events individually and in combination, for example, an earthquake and a tsunami.

Earthquake

4.80. The R&D facility should be designed for the design basis earthquake to ensure that an earthquake does not induce a failure that would result in a loss of confinement or a criticality accident. Seismically induced failures of containment or criticality safety parameters (such as geometry and moderation) could have significant consequences for other personnel on the site or members of the public.

4.81. To determine the design basis earthquake, the main characteristics of the disturbance (e.g. intensity, magnitude and focal distance), based on historical data and the distinctive geological features of the area close to the facility, should be determined. The approach should ideally evaluate the seismological factors on the basis of historical data for the site. Where historical data are inadequate or yield large uncertainties, an attempt should be made to gather palaeoseismic data to facilitate determination of the most intense earthquake for the R&D facility location. These different approaches can be combined since the regulatory body generally considers both in the approval of the design.

4.82. One means of specifying the design basis earthquake is to consider the historically most intense earthquake, but increased in intensity and magnitude, for the purpose of obtaining the design response spectrum (i.e. the relationship between frequencies and ground accelerations) used in designing the R&D facility. Another way of specifying the design basis earthquake is to perform a geological review, to determine the existence of capable faults and to estimate the ground motion that such faults might cause at the location of the R&D facility.

4.83. An adequately conservative spectrum should be used for calculating the structural response to guarantee the stability of buildings and to ensure the integrity of the ultimate means of confinement in the event of an earthquake. Certain SSCs important to safety will require seismic qualification. This will apply mainly to equipment used for storage and vessels that contain materials necessary for safety and hazardous chemical materials. Design calculations for the buildings and equipment should be made to verify that, in the event of an earthquake, no unacceptable release of radioactive material to the environment would occur and the risk of a criticality accident would be very low.

External fire and explosions

4.84. Hazards from external fires and explosions could arise from various sources near to R&D facilities, such as petrochemical installations, forests, pipelines, and road, rail or sea routes used for the transport of flammable material such as gas or oil.

4.85. To demonstrate that the risks associated with such external hazards are within acceptable levels, the operating organization should first identify all potential sources of hazards and then estimate the associated event sequences affecting the R&D facility. The radiological and associated chemical consequences of any damage should be evaluated and it should be verified that they are within acceptance criteria. The operating organization should carry out a survey of potentially hazardous installations and transport operations for hazardous material close to the R&D facility. In the case of explosions, risks should be assessed for compliance with overpressure criteria.

4.86. To evaluate the possible effects of flammable liquids, falling objects (such as chimneys) and missiles resulting from explosions, their possible distance from the R&D facility and hence their potential for causing physical damage should be assessed. Toxic hazards should be assessed to verify that specific gas concentrations meet the acceptance criteria and do not affect the controllability of the R&D facility.

Extreme weather conditions

4.87. Typically, the extreme weather conditions used to design and/or evaluate the response of an R&D facility are wind loading, tornadoes, rainfall, snowfall, ice storms and extreme temperatures.

4.88. The general approach is to use a deterministic design basis value for the extreme weather condition and to assess the effects of such an event on the safety of the R&D facility. The rules for obtaining the design basis values for use in the assessment may be specified by local or national regulations.

4.89. The design provisions will vary according to the type of hazard and its effects on the safety of the R&D facility. For example, extreme wind loading is associated with rapid structural loading and thus design provisions for this event should be the same as those for other potentially rapid loading events such as earthquakes. However, the effects of extreme precipitation or extreme temperatures would take time to develop and hence there is time for operational actions to be taken to limit the consequences of such events.

4.90. An R&D facility should be protected against extreme weather conditions by means of appropriate design provisions. These should generally include:

(a) The ability of structures important to safety to withstand extreme weather loads;
(b) Prevention of flooding of the R&D facility;
(c) The safe shutdown of experiments in the R&D facility in accordance with the operational limits and conditions.

Tornadoes

4.91. Measures for protection against tornadoes will depend on the meteorological conditions in the area where the R&D facility is located. The design of buildings and ventilation systems should comply with specific regulations relating to hazards from tornadoes.

4.92. High winds are capable of lifting and propelling objects such as automobiles or telegraph poles. The possibility of impacts by missiles such as these should be considered in the design stage for the R&D facility, taking account of their initial impact and possible secondary fragments arising from collisions with, and spallation from, concrete walls or by other momentum transfer mechanisms.

Extreme temperatures

4.93. The possible duration of extreme low or high temperatures should be taken into account in the design of support system equipment to prevent unacceptable effects such as the freezing of cooling circuits or adverse effects on ventilation and cooling systems.

4.94. If safety limits for humidity and/or temperature are specified in a building or a compartment, the air-conditioning system should also be designed to perform efficiently under extreme hot or wet weather conditions.

4.95. Human access may be essential for safety, and under such circumstances, the combined effects of low temperatures and ventilation on operators should be considered.

Snow and ice

4.96. The occurrence of snowfall and its effects should be taken into account in the design of the R&D facility and in its safety analysis. Snow is generally taken into account as an additional load on the roofs of buildings. Snow can also block the inlets of ventilation systems and the outlets of drains. The neutron reflecting effect or the interspersed moderation effect of the snow should be considered, if relevant. The effect of ice on wall loadings should also be considered where this is a possibility.

External floods

4.97. Floods should be taken into account in the design of an R&D facility. Two approaches to cope with flood hazard used in various States are as follows:

(a) The highest flood levels historically recorded are taken into account and the nuclear facilities are sited at specific locations above the flood level, or at sufficient elevation to avoid major damage from flooding.
(b) Where the use of dams is widespread and where a dam has been built upstream of a potential or existing site of a nuclear facility, the hazard posed by a breach of the dam is taken into account. The buildings of the facility are designed to withstand the water wave arising from the breach of the dam. In such cases, the equipment — especially that used for the storage of fissile material — should be designed to prevent any criticality accident.

Accidental aircraft crash hazards

4.98. The likelihood and possible consequences of impacts onto the R&D facility should be calculated by assessing the number of aircraft that come close to the R&D facility and their flight paths, and by evaluating the areas vulnerable to impacts, i.e. areas where hazardous material is processed or stored. If the risk is acceptably low, no further evaluations are necessary. Further guidance is provided in section 5 of NS-G-3.1 [17] and requirements are established in para. 5.5 of NS-R-5 (Rev. 1) [1].

4.99. For evaluating the consequences of impacts or the adequacy of the design to resist aircraft impacts, only credible crash scenarios should be considered, which may require the knowledge of such factors as the possible angle of impact, or the potential for fire and explosion due to the aviation fuel load. In general, fire cannot be ruled out following an aircraft crash, and so the establishment of specific requirements for fire protection and for emergency preparedness and response will be necessary.

INSTRUMENTATION AND CONTROL

Instrumentation

4.100. Instrumentation should be provided to monitor facility parameters and systems over their respective ranges for: (1) normal operation; (2) anticipated operational occurrences; (3) design basis accidents (or their equivalents); and (4) design extension conditions[5]. The information obtained on the status of the facility and experiments should allow any necessary actions to be undertaken in accordance with operating procedures or in support of automatic systems.

4.101. Instrumentation should be provided to measure all the main variables that may affect the processes and to monitor the general conditions of the R&D facility for safety purposes (such as radiation doses due to internal and external exposure, releases of effluents and ventilation conditions) and for obtaining any information on the facility necessary for its reliable and safe operation. Provision should be made for the automatic measurement and recording of parameters that are important to safety, including remote monitoring if necessary.

Control systems

4.102. Passive and active engineering controls are more reliable than administrative controls, and should be preferred for control in normal operational states and in accident conditions. When used, automatic systems should be designed to maintain process parameters of the R&D facility or experimental apparatus within operational limits and conditions or to bring the process to its safe stable state, which is generally the shutdown state.

4.103. Appropriate information for monitoring the effects of automatic actions should be made available to the R&D facility operators. The layout of the instrumentation and the mode of presentation of information should provide the operating personnel with an adequate overall picture of the status and

[5] Design extension conditions are postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.

performance of the R&D facility. Devices should be installed that efficiently provide visual and, as appropriate, audible indications of operational states that have deviated from normal conditions and that could affect safety. Control systems should be provided to ensure compliance with regulatory limits, for example, on discharges.

Control rooms

4.104. Control rooms should be provided to centralize the main (e.g. surveillance and overview monitoring) data displays, controls and alarms for general conditions at the R&D facility. For specific experiments, it may be useful to have local control areas where relevant information can be gathered together and monitored. Such controls should be located in parts of the R&D facility where risks to operators and occupational exposure can be minimized. Particular consideration should be given to identifying events, both internal and external to the control rooms, that may pose a direct threat to the operators and to the operation of control rooms. Ergonomic factors should be taken into account in the design of the control room.

Safety related instrumentation and control for normal operation

4.105. For normal operation, safety related instrumentation and control systems should be separated from experimental instrumentation and should include, where appropriate, systems for:

(a) Criticality control: Where there is a risk of criticality and depending on the method of criticality control, monitoring and control parameters should include mass, density, moisture content, isotopic content, fissile content, reflection and moderation by additives and the location of materials.
(b) Monitoring and control of equipment and supplies: For the safety of R&D equipment, it may be necessary to monitor and control a number of safety parameters, for example, temperature, gas flow, fluid compositions or flow rates and pressure. A means of confirming correct concentrations of reactive media in supplies to hot equipment should be provided.
(c) Glovebox control: For gloveboxes under inert atmosphere, the gas concentration should be monitored and controlled for safety and possibly for product quality purposes. Temperatures should also be monitored. Instrumentation and controls for fulfilling requirements for negative pressure and requirements for fire control should be in place, in accordance with paras 9.60 and II.25 of NS-R-5 (Rev. 1) [1].

(d) Monitoring of external occupational radiation doses: Sensitive dosimeters with real time displays and/or alarms should be used to monitor and control occupational radiation doses, especially in areas with inspection equipment using X rays and sealed radiation sources. Installed equipment should be used where possible to control gamma and neutron whole body exposures.

(e) Monitoring of internal occupational radiation doses: In R&D facilities with the potential for airborne contamination, the following provisions should be considered in order to ensure early detection of radioactive particulates:
   (i) Installation of continuous air monitors to detect contamination as close as possible to the working areas;
   (ii) Installation of detectors for surface contamination (alpha, beta or gamma) close to working areas and for self-monitoring at the exits of rooms.

(f) Monitoring and control of liquid discharges: The liquid discharges of R&D facilities should be appropriately monitored and controlled. This can be done by sampling and analysis, and by measuring the volume of discharge.

(g) Control of gaseous effluents: Generic requirements for control of atmospheres and pressures are established in paras 6.37–6.39 of NS-R-5 (Rev. 1) [1], which state that:

> "The nature and number of the barriers and their performance, as well as the performance of air purification systems, shall be commensurate with the degree of the potential hazards, with special attention paid to the potential dispersion of alpha emitters… Means of monitoring and appropriate alarm systems for atmospheric contamination shall be installed."

Such means should include measurements of, for example, differential pressure to confirm that the filtration systems are working effectively, and continuous monitoring of discharges. Monitoring and control is necessary to ensure that the airflows in all areas of the R&D facilities are flowing in the correct directions, i.e. from less contaminated to more contaminated areas. In work areas, the temperature, humidity and pollutants should be controlled to ensure worker comfort and hygiene. In some cases, local ventilation should be used, for example, in rooms housing backup batteries.

Safety related instrumentation and control systems for operational occurrences

4.106. In addition to the list in para. 4.105, safety related instrumentation and control systems for use in anticipated operational occurrences should include the following provisions:

(a) Fire detection and extinguishing systems and building evacuation systems;
(b) Radiation detection and alarm systems;
(c) Airborne activity detection and alarm systems;
(d) Gas detectors and alarm systems, where leakage of gases such as hydrogen could produce an explosive atmosphere;
(e) Diluting gas flows for vessels where hydrogen accumulation could be an issue.

Safety related instrumentation and control systems for design basis accidents

4.107. In addition to the lists in paras 4.105 and 4.106, the safety related instrumentation and control systems for design basis accident conditions (or equivalent) should include:

— Where there is a potential for criticality, criticality detection systems, alarm systems and building evacuation systems;
— Detection and alarm systems for abnormal releases of effluents.

HUMAN FACTOR CONSIDERATIONS

4.108. R&D facilities are often highly reliant on human operations but such reliance should not preclude the provision of design safety features that minimize the potential for accidents caused by significant human errors. Requirements relating to consideration of human factors are established in paras 6.15 and 6.16 of NS-R-5 (Rev. 1) [1].

4.109. Human factors in operation, inspection, periodic testing and maintenance should be considered at the design stage. Factors to be considered include:

— Possible effects on safety of human errors (with account taken of ease of intervention by the operator and tolerance of human error);
— The potential for occupational exposure.

4.110. The design of an R&D facility to take into account human factor considerations is a specialist area. Experts and experienced operators should be involved from the earliest stages of design. Areas that should be considered include:

(a) Design of working conditions to ergonomic requirements:
   - (i) The human–machine interface, for example, electronic control panels displaying all the necessary information and no superfluous information;
   - (ii) The working environment, for example, ensuring good access to and adequate space around equipment, and suitable finishes to surfaces for ease of cleaning;
   - (iii) Safety features of commercial equipment that has been adapted for nuclear use (e.g. in a glovebox).

(b) Choice of location and clear labelling of equipment so as to facilitate inspection, maintenance, testing, cleaning and replacement.

(c) Provision of fail-safe equipment and automatic control systems for accident sequences for which reliable and rapid protection is required.

(d) Good task design and job organization, particularly during maintenance work, when automated control systems may be disabled.

(e) Minimization of the need to use personal radiation protection (such as tabards).

4.111. In the design and operation of fume hoods, gloveboxes and (where appropriate) hot cells, the following specific considerations should be taken into account:

(a) The design of equipment to avoid conventional laboratory hazards that may result in injuries to workers, including internal radiation exposure through cuts in the gloves, wounds on the operator’s skin and/or the possible failure of confinement;

(b) Ease of physical access, adequate working space and good visibility;

(c) The potential for loss of confinement, including damage to gloves;

(d) Training of operators on procedures to be followed in normal and abnormal conditions.

SAFETY ANALYSIS

4.112. The safety analysis for an R&D facility should be performed in two main steps:

(1) The assessment of occupational exposure and public exposure for operational states of the R&D facility and comparison with authorized limits for operational states;
(2) The determination of the radiological and associated chemical consequences to the public from accidents and identification of design extension conditions, and verification that they can be controlled within the limits specified for accident conditions.

4.113. The results of these two steps should be reviewed to identify a possible need for engineered safety features and/or additional operational limits and conditions.

Safety analysis for operational states

Occupational radiation exposure and exposure of the public

4.114. At the design stage of a new R&D facility, an assessment should be made of the radiation exposure of workers in all workplaces within the facility, based on conservative assumptions for factors including the following:

(a) Licensed inventories of radioactive materials in each part of the R&D facility;
(b) Calculated radiation levels, which should use the enveloping R&D facility source term wherever it is located;
(c) The maximum cumulative annual working time at each workplace for both normal operation and anticipated maintenance work;
(d) Calculations of the efficiency of shielding during normal operation based on conservative assumptions regarding its performance.

4.115. The design of equipment and the layout of equipment and shielding in the R&D facility should be based on adequate interaction and feedback between process and mechanical designs, safety assessment and operating experience from similar facilities and/or facilities upstream in the process.

4.116. Cleaning operations (e.g. the elimination of dust from fume hoods, gloveboxes and hot cells) should be given special consideration in the design.

4.117. The calculated doses should be compared with actual doses during subsequent operation of the R&D facility. If considered necessary, maximum permissible annual working times for specific workplaces may be included in the operational limits and conditions.

4.118. Calculations of estimated public doses should be made on the basis of maximum estimated releases of radioactive material and maximum depositions to the ground. Conservative models and parameters should be used to calculate the estimated doses to the public.

Release of non-radioactive hazardous materials

4.119. This Safety Guide deals principally with those material hazards that can give rise to radiological hazards (see para. 2.2 of NS-R-5 (Rev. 1) [1]). Realistic and robust (i.e. conservative) estimations of material toxicity to personnel of the R&D facility should be made. Releases of hazardous radioactive chemicals or biological materials affecting the public or the environment should be evaluated using conservative models and parameters, to standards that are no lower than those used in equivalent non-nuclear industries; see Ref. [22].

Safety analysis for accident conditions

Methods and assumptions for safety analysis for accident conditions

4.120. For R&D facilities, the consequences of accidents are not necessarily limited to individuals located on the site and in close proximity to the location of the accident. Consequences will depend on various factors such as the release rate and quantity, distance between receptor and source of release, material transport to the receptor and exposure time.

4.121. The acceptance criteria associated with the accident analysis should be defined in accordance with para. 6.5 of NS-R-5 (Rev. 1) [1] and with respect to any national regulations and risk criteria. To estimate the on-site and off-site consequences of an accident, the wide range of physical processes that could lead to a release of radioactive material to the environment should be modelled in the accident analysis and the enveloping cases encompassing the worst consequences should be determined (see paras 2.6, 2.10–2.12 and 4.24 of NS-R-5 (Rev. 1) [1]).

4.122. The following approaches should be considered in the assessment:

(1) An approach using the bounding case (the worst case approach), with account taken only of those safety features that mitigate the consequences of accidents and/or that reduce their likelihood. If necessary, a more realistic case can be considered that includes the use of some safety features and some non-safety features beyond their originally intended range of functions to reduce the consequences of accidents (the best estimate approach). Mobile or easily displaced or removed equipment should not be credited in safety analysis.
(2) An approach using the bounding case (the worst case approach), with no account taken of any safety feature that may reduce the consequences or the likelihood of accidents. This assessment is followed by an assessment of the possible accident sequences, with account taken of the emergency procedures and the means planned for mitigating the consequences of the accident.

The second approach should only be used if safety features cannot be demonstrated to be effective.

Assessment of possible consequences of an accident

4.123. Safety assessments should address consequences associated with possible accidents. The main steps in the development and analysis of an accident scenario should include:

(a) Analysis of the actual site conditions and conditions expected in the future.
(b) Identification of workers and members of the public (i.e. the representative person living in the vicinity of the R&D facility) who could possibly be affected by accidents, allowing for demographic variations.
(c) Specification of the accident configurations, with the corresponding operating procedures and administrative controls for operations.
(d) Identification and analysis of conditions at the R&D facility, including internal and external initiating events that could lead to a release of material or of energy with the potential for adverse effects, the time frame for emissions and the exposure time, in accordance with reasonable scenarios.
(e) Specification of the SSCs important to safety that are credited with reducing the likelihood of, and/or mitigating the consequences of, accidents. These SSCs that are credited in the safety assessment should be qualified to perform their functions in the accident conditions.

(f) Characterization of the source term (material, mass, release rate, temperature).
(g) Identification and analysis of transport pathways for released material within the facility.
(h) Identification and analysis of pathways by which material that is released could be dispersed in the environment.
(i) Quantification of the consequences for the representative person identified in the safety assessment.

4.124. Analysis of the actual conditions at the site and the conditions expected in the future involves a review of the meteorological, geological and hydrological conditions at the site that may influence facility operations or contribute to transporting material or transferring energy that may be released from the facility; see section 5 of NS-R-5 (Rev. 1) [1].

4.125. Environmental transport of material should be calculated with qualified models and codes or using data derived from qualified codes, with account taken of the meteorological and hydrological conditions at the site that would result in the highest exposure of the public.

EMERGENCY PREPAREDNESS AND RESPONSE

4.126. The hazards associated with an R&D facility and potential consequences, if an emergency occurs, should be assessed to provide a basis for adequate emergency arrangements in accordance with GSR Part 7 [11], GS-G-2.1 [12] and para. 9.62 of NS-R-5 (Rev. 1) [1]. The on-site and off-site emergency arrangements, including emergency plan(s) and procedures, that take into account the potential hazards assessed for the facility (the plant and experimental equipment) should be developed for a range of postulated emergencies. Such emergencies include, but are not limited to, criticality accidents and nuclear or radiological emergencies coincident with external hazards affecting the infrastructure in the vicinity of the R&D facility (e.g. widespread fires, earthquakes and tsunamis).

4.127. The R&D staff running experiments should inform management of the hazards and shutdown arrangements for all experiments in the facility, for both Case 1 and Case 2 facilities.

4.128. For Case 2 R&D facilities, an expanded list of hazards is defined in the IAEA Safety Guides related to the corresponding type of nuclear fuel cycle facilities, for example in SSG-6 [6], SSG-5 [18], SSG-7 [19] and Safety of Nuclear Fuel Reprocessing Facilities, IAEA Safety Standards Series No. SSG-42 [24]. These should be considered in the hazard assessment used for developing the emergency arrangements.

MANAGEMENT OF RADIOACTIVE WASTE

General

4.129. Requirements for managing radioactive waste from R&D facilities are established in paras 6.31–6.34 in NS-R-5 (Rev. 1) [1]. General requirements on predisposal management of radioactive waste are established in Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5 [25] and further guidance is provided in The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSG-3 [26]. Further information on the optimization of protection for radioactive waste is provided in Refs [27, 28]. Specific guidance on predisposal management of radioactive waste from nuclear fuel cycle laboratories is provided in Predisposal Management of Radioactive Waste from the Use of Radioactive Material in Medicine, Industry, Agriculture, Research and Education, IAEA Safety Standards Series No. SSG-45 [29], while guidance that may be relevant to pilot plants can be found in Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40 [30] and Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-41 [31]. IAEA safety standards require the generation of radioactive waste to be minimized in volume and activity, as far as practicable. The following aspects should be considered in design:

(a) Generation of waste: Requirement 8 of GSR Part 5 [25] establishes general design requirements for radioactive waste generation and control. These requirements include the proper characterization of wastes in terms of total activity, concentrations of relevant radionuclides and other hazards at the generation stage. A record keeping system should be implemented to ensure the proper identification, traceability and accounting for the radioactive waste generated, and the avoidance of criticality conditions should be ensured when fissile material becomes waste and during its subsequent processing. In fume hoods, gloveboxes and hot cells it is possible to reduce waste by reducing the amount of material introduced into these installations.

(b) Handling of waste: Requirement 10 in GSR Part 5 [25] states that adequate containers are required to be provided for radioactive waste removed from R&D facilities. It is good practice to minimize the spread of contamination by control at the point of origin. Guidance on the handling of waste containing fissile material, including guidance on mass control, is provided in SSG-27 [10]. Special requirements apply to such waste, as stated in para. V.15 of NS-R-5 (Rev. 1) [1], including a requirement for engineered features providing containment and control of geometry. Examples include filters from fume hoods, gloveboxes, hot cells and ventilation systems.

(c) Collection of waste: Design features should reduce the risk of damage to waste containers that can potentially lead to loss of confinement. For the predisposal management of radioactive waste, consideration should be given to a central waste management area. In this central area, the waste should be characterized (including any fissile content) and classified. The waste may subsequently be treated and placed in containers in this area, for interim storage. The mixing of wastes that are chemically or radiologically incompatible in the same containers or storage areas should be avoided by design where possible.

(d) Storage of waste: The design of storage areas and waste containers should take account of radioactivity and other hazards of the waste, even if the storage is intended to be short term. Requirement 11 of GSR Part 5 [25] states that "Waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management." Measures to guarantee the integrity of the facility and the waste containers considering low probability events should be taken even for interim storage.

(e) Processing of waste: Subsequent processing of the waste outside the R&D facility can include pretreatment (i.e. segregation, chemical adjustment and decontamination), treatment (i.e. volume reduction, removal of radionuclides from the waste and change of composition) and conditioning (i.e. immobilization and packaging), before longer term storage. The preferred techniques and procedures for treatment and conditioning provide waste forms and/or waste packages in line with the established or anticipated waste acceptance requirements for storage and eventual disposal.

Management of gaseous and liquid discharges

4.130. The discharge of gaseous effluents from an R&D facility should be controlled by an air purification system, which normally consists of a number of high efficiency particulate air (HEPA) filters in series. Performance standards should be set for the air purification system, in accordance with an appropriate safety assessment.

4.131. Monitoring equipment such as the following should be installed and used:

(a) Differential pressure gauges for detecting when filters need to be changed;
(b) Activity or gas concentration measurement devices and discharge flow measuring devices with continuous sampling;
(c) Injection and sampling equipment for testing filter performance.

4.132. Liquid effluents to the environment should be treated to reduce the discharge of radioactive material and hazardous chemicals to levels authorized by regulatory bodies. The use of filters, ion exchange beds or other technology should be considered where appropriate.

OTHER DESIGN CONSIDERATIONS

Gloveboxes and hot cells

4.133. Fume hoods, gloveboxes and hot cells should be designed to facilitate the use of dry cleaning methods (e.g. with criticality safe filtered vacuum cleaners). Features such as easily cleanable surfaces, strippable coatings and rounded corners should be considered.

Radiation protection shielding

4.134. The materials handled in an R&D facility can generate significant dose rates (neutron, beta/gamma) depending on the isotopic composition of the material processed. Therefore, consideration should be given at the design stage to the need for neutron and gamma shielding.

4.135. Effective gamma and neutron shielding can be applied to the faces of hot cells and gloveboxes but this can restrict visibility and increase occupancy periods of workers. The choice and type of shielding should therefore be based on a prediction of the total occupational exposure during normal operation and maintenance.

Design for fresh fuel storage

4.136. Storage facilities for fresh fuel should be designed with fixed, dry and marked locations for the fuel, in accordance with the conclusions of the criticality safety analysis. Racks, fixings and handling arrangements should be capable of accommodating fuel of the required dimensions while maintaining the required stability. Fuels should be clearly identifiable. Necessary provisions for physical protection should be included in the design.

4.137. In designing storage facilities for fresh fuel, consideration should also be given to provisions for:

(a) Weighing items for inventory control and verification without the need to transfer fuel to and from storage;
(b) Space and facilities for packaging, with an inert atmosphere if appropriate.

Design for maintenance

4.138. Design for maintenance should include the following aspects:

(a) Consideration of whether maintenance can be carried out remotely instead of manually using personal protective equipment.
(b) Measures to maintain criticality safety conditions such as limiting the introduction of liquids, solvents, plastics and other moderators.
(c) Prevention of the spread of contamination when maintaining or replacing equipment (e.g. motors and drives can be located outside gloveboxes).
(d) The R&D facility design should aid good housekeeping. Gloveboxes and hot cells can become dusty unless cleaned regularly. Tools should be stored in designated locations. Waste accumulation should be avoided.
(e) Removal of shielding material. Shielding on gloveboxes is often provided for normal process operations and may need to be removed for maintenance access. Consideration should be given to removing all radioactive sources before removing any shielding.
(f) The facility design should minimize sharp edges and the need for sharp equipment in gloveboxes to minimize the potential to cause wounds that could become contaminated.
(g) The design of replaceable parts should facilitate segregation and handling of mixed and hazardous waste.
(h) Surveillance and monitoring requirements for ageing and degradation.

Decontamination and dismantling

4.139. Floor, wall and ceiling surfaces should be selected, particularly in wet chemical areas, to facilitate decontamination and future decommissioning. Surfaces in areas where contamination may exist should be made non-porous and easy to clean, particularly in rooms containing hot cells and gloveboxes, as well as within the hot cells and gloveboxes themselves. Appropriate methods include the application of coverings or coatings to such surfaces, for instance by using paint, resins or stainless steel liners. They should be designed without corners or crevices that may be difficult to access. In addition, all potentially contaminated surfaces should be made readily accessible to allow for periodic and eventual decontamination (e.g. by stripping of paint or coatings).

5. CONSTRUCTION

5.1. Paragraph 7.1 of NS-R-5 (Rev. 1) [1] states "Before the construction of a fuel cycle facility begins, the operating organization shall satisfy the regulatory requirements regarding the safety of the facility design", and the construction of an R&D facility will also require authorization by the regulatory body.

5.2. For a complex R&D facility, authorization should be sought in several stages. Each stage may conclude with a hold point at which approval by the regulatory body is required before the subsequent stage may commence. The extent of regulatory involvement during construction should be commensurate with the potential hazards posed by the R&D facility during its expected lifetime.

5.3. Current good practices should be used for building construction, and for fabrication and installation of facility equipment. Effective means should be put in place to prevent the installation of counterfeit, fraudulent or suspect items, as well as non-conforming or sub-standard components, because such items or components could impair safety even after the commissioning of the R&D facility.

5.4. Modularized components (e.g. gloveboxes, hot cells, fume hoods, monitoring systems) should be used in the construction of complex R&D facilities for fundamental research and analysis (Case 1 facilities). This enables equipment to be tested and proven at the manufacturer's premises before installation in the R&D facility. In addition, this approach also aids commissioning, maintenance and decommissioning.

5.5. The construction of parts of the R&D facility and the commissioning or operation of other parts of the R&D facility can overlap. Construction in a radioactive environment can be significantly more difficult and time consuming than when no active material is present. When this occurs, the R&D facility organization should take measures to prevent:

(a) Construction personnel from receiving unnecessary exposure to radiation;
(b) Damage caused by construction activities to SSCs necessary for operating the R&D facility;
(c) Transfer of radioactive material to the part of the facility under construction;
(d) Any harm to personnel in the operating part of the facility.

5.6. Preventative measures should also include the training of construction personnel for their own safety and the safety of others on simulated installations prior to performing actual construction.

5.7. Consideration should be given to the quality assurance programme during the construction of an R&D facility. The programme should be prepared early in the construction stage and should include:

(a) Applicable codes and standards;
(b) The organizational structure;
(c) Design change programme (configuration control);
(d) Procurement control;
(e) Records maintenance;
(f) Equipment testing;
(g) Coding and labelling of safety relevant components, cables, piping and other pieces of equipment.

5.8. Further guidance on safety in the construction of nuclear installations can be found in Construction for Nuclear Installations, IAEA Safety Standards Series No. SSG-38 [32].

6. COMMISSIONING

6.1. Section 8 of NS-R-5 (Rev. 1) [1] sets out the requirements applicable to the commissioning of an R&D facility. A commissioning programme should be prepared and implemented to demonstrate that the R&D facility conforms to its designed objectives and safety performance criteria as well as to familiarize the operating personnel with the particular characteristics of the facility. The establishment of a good safety culture should start at the earliest possible stage of commissioning.

6.2. Paragraph 8.9 of NS-R-5 (Rev. 1) [1] establishes the requirement for commissioning to be divided into stages; this requirement is also applicable to an R&D facility at the plant or experimental level.

Cold commissioning

6.3. In this stage, the facility's systems are tested in the absence of radioactive material. The facility is tested systematically, as individual items of equipment and as systems in their entirety (see para. 8.9 of NS-R-5 (Rev. 1) [1]). As it is relatively easy to take corrective actions at this point, as much verification and testing as possible should be carried out in this stage. Operators should take the opportunity to prepare the set of operational documents and to learn the details of systems. Leaktightness and the stability of control systems are best tested at this stage.

Warm commissioning

6.4. The emergency arrangements for the facility should be in place prior to the next stage of commissioning, in accordance with GSR Part 7 [11]. Natural or depleted uranium should be used in this stage as necessary, to avoid criticality risks, to minimize occupational radiation exposure and to limit possible needs for decontamination. This stage also provides the opportunity to initiate the control regimes that will be necessary when higher activity materials such as plutonium, other actinides or fission products are introduced.

6.5. Safety tests performed during this commissioning stage should mainly be devoted to confinement checking. These should include: (i) checking for airborne radioactive material; (ii) smear checks on surfaces; and (iii) checking for gaseous discharges and liquid releases. Checks for unexpected accumulations of hazardous material should also be carried out.

Hot commissioning

6.6. This stage enables administrative and engineered systems to be progressively and cautiously brought into full operation, with radioactive material present. Paragraphs 8.5 and 8.10 in NS-R-5 (Rev. 1) [1] establish requirements to fully exercise radioactive systems and reinforce safety culture to ensure that operating personnel are fully trained in handling radioactive material and the associated emergency arrangements.

6.7. The licence to operate the R&D facility is generally issued to the operating organization just before this third stage. The regulatory body should define hold points and/or witness points as licence obligations, coordinated with the proposed commissioning programme; see Licensing Process for Nuclear Installations, IAEA Safety Standards Series No. SSG-12 [33]. At this stage, hot commissioning will be performed under the responsibility, safety procedures and organization of the licensed operator. Hot commissioning may be considered part of the operational stage of the R&D facility.

6.8. The safety committee of the R&D facility (or an equivalent review body) should be established before active commissioning commences, if one has not been established already. Lessons learned from similar facilities should be applied especially for the commissioning of a new R&D facility.

6.9. During commissioning and later, during operation of the R&D facility, predicted estimates of doses to workers should be assessed against actual dose rates. If, in operation, the actual doses are higher than the predicted doses, corrective actions should be taken, including making any necessary changes to the licensing documentation (e.g. the safety case) or adding or changing safety features or work practices (see also Sections 6 and 7). The Fundamental Principles 4, 5 and 6 of Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1 [34] apply.

6.10. For R&D facilities, the review of worker doses starts during the commissioning stage but continues throughout the lifetime of the facility as new experiments and materials are introduced or parts of the facility are brought into operation.

7. OPERATION

CHARACTERISTICS OF R&D FACILITIES

7.1. Paragraph 9.3 of NS-R-5 (Rev. 1) [1] states:

> "The operating organization shall have the overall responsibility for the safety of the facility during operation. The operating organization shall establish an appropriate management structure for the facility and shall provide the necessary infrastructure for operations to be conducted safely."

Paragraphs 9.4 and 9.5 in NS-R-5 (Rev. 1) [1] detail responsibilities for operations, maintenance and control of modifications. These requirements and the general guidance in GS-G-3.5 [4] are relevant to R&D facilities. This section provides specific guidance on good practices and additional considerations in meeting the safety requirements for an R&D facility, including operations and experiments that may be undertaken by different teams, or by different organizations. Paragraph 1.2 of this Safety Guide outlines some distinctive hazards for an R&D facility that should be taken into account in meeting the safety requirements.

7.2. Safety should be coordinated between the operational functions and the research functions of the R&D facility. The interface between operations and research provided by the safety committee should not be used as a substitute for procedures for everyday communication and cooperation on safety between these functions, which should also be documented. Responsibilities that should be coordinated carefully include the management of radioactive material, waste management and the monitoring of experiments. The safety committee (or equivalent body) of the R&D facility should comprise representatives of operations, safety and research functions.

7.3. Research programmes should comply with the existing safety case or be considered as a modification. Research requires flexibility in the materials and processes used and the safety case should anticipate a variety of research needs; see para. 2.7. The domain of safe operation defined through the operational limits and conditions should be sufficiently large to avoid frequent modifications of the safety case or of the regulatory authorization. Any modification should be reviewed and made subject to approval by the appropriate authority, in accordance with regulatory requirements.

7.4. Some of the operational activities performed in an R&D facility are more appropriate for Case 1 facilities and others are more appropriate to Case 2 facilities, as described in Annexes I and II. Some paragraphs in this section refer to these cases and to the Annexes.

QUALIFICATION AND TRAINING OF PERSONNEL

7.5. The general safety requirements relating to the qualification and training of R&D facility personnel are defined in paras 4.10, 4.24, 8.4 and 9.8–9.13 of NS-R-5 (Rev. 1) [1].

7.6. The diversity of R&D facility personnel should be accommodated by the training programmes for safety. All training programmes linked with the R&D facility should aim to establish a common safety culture.

7.7. In such training programmes, emphasis should be given to individual responsibility for safe operation, organization, human factors, lessons learned from events (both at the facility and at other facilities), defence in depth and assessment of the safety of specific R&D facility programmes or operations.

7.8. The operating organization should consider the effect of changes in research and operating personnel and work programmes when planning training programmes.

7.9. Many processes relating to glovebox and hot cell operations involve manual intervention. Therefore, special attention should be paid to training R&D facility personnel operating gloveboxes and hot cells, including reaction to anticipated operational occurrences (e.g. a punctured glove in a glovebox or a loss of ventilation in a hot cell).

FACILITY OPERATION

7.10. Paragraph 9.6 of NS-R-5 (Rev. 1) [1] establishes requirements related to interdependencies and communication between facilities on the same site. Different organizational units within an R&D facility should hold regular work planning meetings to achieve a common work plan and to coordinate activities. Clear definitions of individual assignments should be documented and made subject to approval at a suitable level of authorization.

7.11. To ensure that the R&D facility operates well within its operational limits and conditions under normal circumstances, a set of lower level sub-limits and conditions should be defined. Such sub-limits and conditions should be clearly defined and understandable and should be made available to the personnel operating the facility. Where there is flexibility for different groups to set their own sub-limits, the management system should ensure that these are notified to all relevant personnel.

7.12. Operating documents should be prepared that list all the limits and conditions under which the R&D facility is operated. Annex IV gives examples of operational limits and conditions applicable to facilities for fundamental research (Case 1 facilities) and processing at a pilot scale (Case 2 facilities), which can be used for defining operational limits and conditions in the various R&D facility areas.

7.13. Generic limits should also be set for the facility. Examples of such limits are:

(a) The allowed ranges of mass control of fissile material during operation, transfer and storage to avoid criticality; for example, the inventory limit for fissile material in gloveboxes;
(b) Specified limits on concentrations, geometry and moderators in solutions containing fissile materials;
(c) Specified inventory limits of radioactive material and isotopic compositions in gloveboxes or interim storage areas;
(d) Maximum heat loads specified for locations such as hot cells or gloveboxes;
(e) Maximum quantities of additives at different steps in R&D facility processes;
(f) Specified limits on combustible material in gloveboxes and hot cells;
(g) Specified limits for flammable atmospheres in enclosed equipment, for example, for hydrogen in a furnace.

7.14. Programmes should be prepared for the routine surveillance of airborne and surface contamination, radiation protection and, more generally, for ensuring an adequate level of housekeeping.

7.15. The values of the key safety variables in operational limits and conditions should be recorded at all times for auditing purposes and to support periodic safety reviews. There should be an investigation and learning process triggered by non-compliances with the operational limits and conditions. The findings of such investigations should be recorded and any lessons identified should be disseminated (operating experience feedback).

7.16. The operating organization should define procedures to ensure a proper level of safety when phases of R&D facility operation are limited and are followed by long periods of shutdown. Training programmes should be capable of coping with such situations and should reflect such procedures.

7.17. Procedures should also include actions required to ensure criticality safety, chemical safety, fire safety, emergency response[6] and environmental protection. Operating procedures should be defined for the ventilation system in fire conditions. Periodic testing and drills should be performed. Operating instructions and procedures should be reviewed periodically and should be updated and authorized as appropriate.

7.18. In the R&D facility measures should be taken to ensure that experiments and processes can be placed in a safe shutdown condition. Some systems, such as ventilation used for confinement, will normally continue to operate. Specific operating procedures should be used for the shutdown of particular processes to prevent, for example, exothermic reactions, hydrogen explosions and criticality. Formal systems of communication should be established to ensure that the facility configuration, including the status of SSCs important to safety, the operational limits, conditions and other key safety information, is known, recorded and accessible at all times.

7.19. An inspection programme for the facility should be established, the purpose of which is periodically to confirm that the R&D facility is operating in accordance with the prescribed operational limits and conditions; see paras 7.24–7.26.

7.20. The management of the R&D facility should arrange for pre-job briefings and a risk assessment briefing at the start of each day and before new operations or experiments are undertaken, to identify potential safety issues and define the best options for safety, as well as to review and assess procedures; see para. 2.37 in GS-G-3.5 [4]. All personnel of the R&D facility should participate in such meetings, as far as possible.

[6] Emergency procedures are part of overall emergency arrangements to be established in accordance with the guidance in paras 4.126–4.128.

MAINTENANCE AND PERIODIC TESTING

7.21. The safety requirements relating to maintenance, calibration, periodic testing and inspection of nuclear fuel cycle facilities are established in NS-R-5 (Rev. 1) [1], paras 9.28–9.34.

7.22. When carrying out maintenance in an R&D facility, particular consideration should be given to the potential for surface contamination or airborne radioactive material, as well as to any chemical or biological hazards. The R&D facility should not be placed in an unsafe or unanalysed condition in order to perform periodic testing or routine maintenance.

7.23. Maintenance should follow good practices with particular consideration given to the following:

(a) A suitable maintenance programme should be developed and implemented for all equipment and devices used in work control, for example, handover and handing back of approved documents, means of communication and visits to job sites, changes to the planned scope of work, suspension of work and ensuring safe access.
(b) Equipment isolation, for example, de-energizing and disconnecting electrical cabling, hot or pressurized piping and draining, venting and purging of equipment.
(c) Testing and monitoring, for example, checks of workplace and tools before commencing work (see para. 5.67 in GS-G-3.5 [4]), monitoring during maintenance and checks for re-commissioning, and communications as above.
(d) Safety precautions for work, for example, specifications ensuring the availability and use of personal protective equipment.
(e) Continued monitoring systems for control of criticality and radiation protection.
(f) Reinstallation of equipment, for example, reassembly, reconnection of pipes and cables, testing, cleaning the job site and monitoring should be performed after maintenance and before re-commissioning.

7.24. A programme of periodic inspections of the R&D facility should be established, as a minimum for fume hoods, hot cells, gloveboxes and entrances to containment areas. The pressure drop across filter banks should be checked on a regular basis. There should be routine programmes of inspection and maintenance designed to prevent the spread of contamination or a release of hazardous material. These programmes should include, for example:

(a) Inspection and maintenance to detect glove material degradation and prevent glove failures;
(b) Maintenance of master–slave manipulators and their sleeves in hot cells.

7.25. Periodic testing of the fire detection and suppression systems for the R&D facility should be carried out. The operational compliance of ventilation systems with fire protection requirements should also be verified on a regular basis.

7.26. Regular verification of the availability of materials necessary for maintenance should be conducted. For continuity of safe operations of the R&D facility, a programme for the provision of spare parts for safety features including radiation monitoring equipment should be established and implemented.

CONTROL OF MODIFICATIONS

7.27. R&D facilities are normally established for a variety of different R&D programmes. It may nevertheless be necessary to modify the facility and its safety case if a new programme of work or item of equipment not covered by the existing authorization is to be implemented or installed. As part of the management system, a process for the control of modifications should be applied in an R&D facility, in accordance with para. 9.35 of NS-R-5 (Rev. 1) [1].

7.28. According to the safety significance of the modification and in agreement with the regulatory body, modifications should be assessed and then registered or otherwise authorized by the regulatory body before the modifications are implemented. The reassessment of the safety of the facility and the formal authorization by the regulatory body identified in para. 3.10 of NS-R-5 (Rev. 1) [1] should consider, in particular, the need to assess human factors, e.g. the human–machine interface, alarm systems, procedures and the qualification or requalification of personnel.

7.29. The control of modifications should be managed in accordance with a process established by the operating organization. A modification control form, which may be an electronic record, should be used as an overall means of monitoring the progress of modifications through the system and as a means of ensuring that all modification proposals receive an equivalent and sufficient level of scrutiny. The modification control form should be used to describe the proposed change and the purpose of the change, and to identify its potential impact on safety. All aspects of safety that may be affected by the modification should be described, with a demonstration that adequate and sufficient safety provisions are in place to control the potential hazards. For example, changes to the materials and thickness of shielding, quantities of hydrogenated and non-hydrogenated materials, and locations of equipment that may affect criticality safety analyses or radiation safety should be described.

7.30. Modification control forms should be scrutinized, and be subject to approval by qualified and experienced persons to verify that the arguments used to demonstrate safety are suitably robust and that the modification meets the requirements of the regulatory body. The depth of the safety arguments and the degree of scrutiny to which they are subjected should be commensurate with the safety significance of the modification.

7.31. The modification control form should also specify which documentation would need to be updated as a result of the modification. Procedures for the control of documentation should be put in place to ensure that documents are changed and distributed within a reasonable time, allowing operating personnel to review, adopt and apply modified procedures when modifications are commissioned. The modification control form should also specify the functional checks that are required before the modified system may be declared fully operational again.

7.32. The modifications made in an R&D facility should be reviewed by the operating organization on a regular basis. This is to ensure that the combined effect of a number of minor modifications do not have hitherto unforeseen effects on the overall safety of the facility. Depending upon national regulatory practices, the results of such a review may also be reported to the regulatory body; see Section 2 of this Safety Guide.

CRITICALITY SAFETY

7.33. Where there is fissile material in an R&D facility, it is particularly important that procedures for controlling criticality hazards (paras 9.49 and 9.50 of NS-R-5 (Rev. 1) [1]) are strictly applied.

7.34. Operational aspects of criticality control in an R&D facility should include consideration of the following:

(a) Unexpected changes in conditions that could increase the risk of a criticality accident, for example, unplanned accumulation of fissile material (e.g. in gloveboxes or ventilation ducts) or hydrogenated materials;
(b) Unexpected accumulation of water due, for example, to fire suppression sprays or leaks from water pipes;
(c) Management of moderating materials, particularly hydrogenated materials such as those used for decontamination of gloveboxes and leakages of oils from gear boxes;
(d) Management of the transfer of fissile material (procedures, mass measurement, systems and records) where mass control is used;
(e) Reliable methods for detecting the onset of unsafe conditions with respect to criticality control;
(f) Evacuation drills and/or exercises (see paras 7.68–7.71 on emergency preparedness);
(g) Periodic calibration or testing of criticality control and monitoring systems (e.g. material movement control, balances and scales).

7.35. The tools used for the purposes of accounting for and control of nuclear material, such as mass, volume or isotope measurements and accounting software, may also have some use in the field of criticality safety. However, where there is any uncertainty about the characteristics of fissile material, conservative values should be used for parameters such as fissile material content and isotopic composition. This arises particularly when handling cell floor or glovebox sweepings and similar waste material.

7.36. Additional safety measures may be required for activities such as maintenance work. For example, "if fissile material has to be removed from equipment only approved containers shall be used", (para. V.14 in NS-R-5 (Rev. 1) [1]). Also, waste and residues arising from experiments or pilot processes, decontamination and maintenance activities should be collected in containers with a favourable geometry approved for the work, and should be stored in dedicated criticality safe areas.

RADIATION PROTECTION

7.37. Paragraphs 9.36 and 9.37 of NS-R-5 (Rev. 1) [1] state:

> "The measures for protection against radiation exposure of operating personnel, including contractors, and members of the public shall comply with the requirements of the regulatory body and with the requirements established in [GSR Part 3 [7]]. For all operational states, the radiation protection measures shall be such as:
>
> (a) To ensure that exposures are kept below regulatory limits;
> (b) To optimize radiation protection."

In an R&D facility, the radiological hazards to both workers and members of the public include intakes (inhalation or ingestion of particulates, aerosols and gases) and external exposure. To ensure effectiveness of the radiation protection measures, action levels and effluent discharge limits should be predefined for comparison with results of monitoring.

7.38. Paragraphs 9.38–9.43 of NS-R-5 (Rev. 1) [1] require the establishment of an appropriate radiation protection programme. For an R&D facility, account should be taken of its complexity and size, as well as the diversity of inventories. In addition, the physical and chemical properties of the inventory may change inadvertently and result in unforeseen consequences.

7.39. Equipment outside of gloveboxes and hot cells, the rooms in the facility and the surrounding environment should be monitored systematically and regularly. Any deviation of the radiation levels above the normal ranges (e.g. hot spots or slow incremental increases of radiation level) should be detected, its origin should be identified and prompt corrective and/or mitigating actions should be taken.

7.40. Radiation protection personnel should be part of the decision making process in an operating R&D facility so that requirements for the optimization of exposures can be applied. Such requirements include the early detection of problems and proper housekeeping for material storage and waste segregation. Any zones with high levels of contamination or high radiation levels should be recorded and marked.

7.41. Intrusive maintenance and modifications should be regarded as major activities requiring justification by facility management and the optimization of protection and safety as required by GSR Part 3 [7]. The procedures for such activities should include:

(1) Estimation of doses (external doses) prior to the activity.
(2) Preparatory activities to minimize the dose, including:
    (a) Identification of specific risks associated with the activities;
    (b) The use of additional shielding, remote devices or mock-ups;
    (c) Definition of specific procedures within the work permit (individual and collective protections requirements such as the use of masks, clothing and gloves, and time limitations).
(3) Measurement of the doses during the activities.
(4) Implementation of feedback to derive possible improvements.

Control of internal exposure

7.42. During operation of an R&D facility (including maintenance and modifications) internal exposure should be controlled by the following means:

(a) Performance standards should be set for all parameters potentially affecting internal exposure, for example, contamination levels.
(b) Regular contamination surveys of facility areas and equipment should be carried out to confirm the adequacy of cleaning programmes.
(c) To aid personnel in considering the level of risk involved in any task and assigning radiation protection personnel to routine workplace surveys, facility areas should be classified into radiation and contamination zones. The boundaries between such zones should be regularly checked and adjusted to match current conditions.
(d) Radiation and contamination zones should be delineated with proper signage.
(e) Continuous air monitoring should be carried out to alert facility operators if airborne contamination is present.
(f) Contamination levels should not be permitted to exceed predetermined action levels.
(g) Mobile air samplers should be deployed where there are sources of airborne contamination, as necessary.
(h) Prompt investigation should be carried out when high levels of airborne contamination have been detected.

(i) Personnel should be trained in putting on, using and taking off personal protective equipment with the assistance of radiological protection personnel.
(j) Personal protective equipment should be maintained in good condition and be regularly inspected.
(k) A high standard of housekeeping should be maintained within the facility. Cleaning techniques should be used that do not give rise to airborne contamination.
(l) The effectiveness of the ventilation system should be checked regularly and rebalanced if necessary, following the isolation or de-isolation of boxes and fume hoods.
(m) Waste arising from maintenance or similar interventions should be segregated by type (i.e. by treatment and disposal route), collected and directed to the appropriate waste route.
(n) Careful consideration should be given to the combination of radiological and industrial hazards (e.g. oxygen deficiency, heat stress) with particular attention paid to the risk/benefit analysis of the use of personnel protective equipment, especially for air-fed systems.
(o) Personnel and equipment should be checked for contamination and should be decontaminated, if necessary, prior to crossing boundaries between contamination zones.

7.43. The method for assessing internal exposure may be based on the collection of air sampling data. In vivo (whole body) monitoring and biological sampling (for example, nose blow, faecal and periodic urine samples) should also be available as necessary for normal and accident conditions as complementary measures to monitor workers' exposure.

7.44. The extent of monitoring should be sufficient to achieve low levels of airborne activity and contamination in workplaces, taking account of the characteristics of specific radionuclides potentially present.

7.45. Entry into and exit from work areas should be controlled to prevent the spread of contamination. In particular, clothing changing and decontamination stations should be available.

7.46. During periodic testing, inspection and maintenance of R&D facilities, precautions should be taken to limit the spread of contamination by means of temporary enclosures and additional ventilation systems.

7.47. On completion of maintenance work, areas should be decontaminated and air sample and smear checks should be carried out to confirm that the area can be returned to normal use. Consideration should be given to grouping similar activities between work periods, in order to optimize protection and ensure that temporary area categorizations are maintained.

7.48. There should be careful preparation before entry into hot cells or gloveboxes that have contained radioactive materials (such as gloveboxes under maintenance). Radiation levels and non-fixed contamination levels should be measured inside the hot cell or glovebox before entry to inform the selection of personal protective equipment and to determine if working time restrictions are required. Such operations necessitate appropriate authorizations, depending on local rules (see GSR Part 3 [7], para. 3.94) and industrial safety requirements for confined space entries.

7.49. Access to areas designated as controlled areas due to the presence of contamination should be avoided by R&D facility personnel with skin wounds.

7.50. On the basis of effluent monitoring data, regular estimates of doses to the public (to a representative person) living near the facility should be made.

Control of external exposure

7.51. There are dedicated areas in an R&D facility where specific arrangements are required to control external radiation exposure. Typically, these will be areas in pilot processing facilities where bulk quantities of radioactive material and other radioactive sources are stored and handled.

7.52. Radiation levels should be controlled at the worksite by:

(a) Ensuring that areas of high occupancy are remote or appropriately shielded from significant quantities of radioactive material;
(b) Removal of radioactive material from areas adjacent to the work area for extended maintenance work;
(c) Handling and operating of instrumentation that contains radiation sources only by suitably qualified and experienced persons;
(d) Performance of routine radiation dose rate surveys.

7.53. External radiation exposure should be controlled by:

(a) Training personnel on radiation hazards and in the use of appropriate workplace monitoring equipment;
(b) Avoiding unnecessary occupation of controlled areas, for example, by limiting working time near radiation sources;
(c) Using individual shielding (e.g. lead aprons) and temporary shielding;
(d) Maintaining a safe distance from radiation sources where practicable.

7.54. Because of the proximity of hands to radioactive material when doing work in gloveboxes, hands are susceptible to receiving a higher dose than other parts of the body. Therefore, the exposure of extremities should be monitored closely (e.g. by the use of finger dosimeters).

7.55. Additional controls may be necessary if radioactive material with higher specific activity is used. This could also introduce additional radionuclides into waste streams. A comprehensive assessment of doses due to occupational and public exposure should be carried out before introducing such radioactive material.

INDUSTRIAL AND CHEMICAL SAFETY

7.56. Paragraph 6.54 of NS-R-5 (Rev. 1) [1] lists conventional hazards to be considered in the design of a fuel cycle facility. The conventional chemical hazards found in R&D facilities and experiments that should be considered include the following:

(a) Chemical hazards due to compounds, such as acids, bases and toxic organic or metallic compounds;
(b) Explosion and fire hazards due to flammable organics, pyrophoric metals, hydrogen, ammonium nitrate and ammonia;
(c) Asphyxiation hazard due to the presence of nitrogen, carbon dioxide or inert gases.

Requirements and guidance for these are provided in international and national standards on chemical safety.

7.57. In a fire, dynamic confinement systems should continue operation (including filtration) to remove smoke, heat and particulates and to compensate for potential overpressure if appropriate. Operation of the dynamic confinement system should be maintained for as long as temperatures at filters do not exceed the threshold at which containment would be lost, as determined by the safety analysis. A fire hazards analysis should be conducted at periodic intervals to incorporate changes that may affect the likelihood of a fire. Computer modelling may be used to support the fire hazards analysis.

7.58. A health surveillance programme should be set up in accordance with national regulations, for routinely monitoring the health of R&D facility workers; see paras 3.76(f), 3.108 and 3.109 in GSR Part 3 [7]. Both the radiological and the chemical effects of chemicals and materials used and produced should be considered as necessary, as part of the health surveillance programme.

7.59. The national and international standards that apply to non-nuclear chemical laboratories also apply to nuclear chemical laboratories. Guidelines should be developed for scientific staff, covering the types of chemical hazards to be expected and the prevention of associated accidents. Much of the guidance may overlap with standard practice for radiation protection and there will be areas where there should be guidance specific to chemical hazards. These may cover topics such as eye protection, reaction hazards and toxicity and may refer to documentation provided by chemical and equipment suppliers or contained in the relevant international and national standards.

MANAGEMENT OF RADIOACTIVE WASTE

7.60. The requirements relating to the management of radioactive waste and effluents in operation are established in paras 9.54–9.57 of NS-R-5 (Rev. 1) [1]. General requirements on the predisposal management of radioactive waste are established in GSR Part 5 [25]. Specific guidance on the predisposal management of radioactive waste from nuclear fuel cycle laboratories is provided in SSG-45 [29], while guidance that may be relevant to pilot plants can be found in SSG-40, SSG-41 [30, 31] and The Management System for the Processing, Handling and Storage of Radioactive Waste, IAEA Safety Standards Series No. GS-G-3.3 [35].

7.61. Performance standards set for air purification systems should specify performance levels at which filters or scrubber media should be changed. After filter changes, tests should be carried out to ensure that filters are not damaged and are correctly seated; smoke tests may be used.

7.62. The generation of solid radioactive waste can be reduced by removing unnecessary packaging from articles before transfer into contamination areas. Processes such as incineration, metal melting and compaction may also be used to reduce the volume of waste [30, 31]. Such processes should be selected on the basis of the characteristics of the waste after segregation. According to national regulations and as far as reasonably achievable, waste material resulting from processing should be recycled or re-used or cleared from regulatory control where possible. Cleaning methods should be adopted that reduce and/or minimize the generation of waste, for instance, the reuse of washings from clean areas when cleaning more contaminated areas.

7.63. As part of the management system, measures for quality assurance and control should be applied for the processing of all waste streams to ensure, as far as achievable, compliance with the waste acceptance criteria for the selected or anticipated disposal option.

7.64. Mixing of waste streams should be limited to those streams that are radiologically and chemically compatible. If the mixing of chemically different waste streams is considered, the chemical reactions that could occur should be evaluated in order to avoid uncontrolled or unexpected reactions.

7.65. The operating organization should characterize radioactive waste as it is generated. Relevant records and reports should be created and managed according to the proper management system; see SSG-40, SSG-41 and GS-G-3.3 [30, 31, 35].

7.66. When legacy materials exist for which there are no data from chemical and/or radiological analyses, reports on the R&D programmes that produced these wastes should be collected or prepared and stored, to be used in subsequent safety assessments. Where necessary to fill gaps in historical information, former employees should be interviewed and published scientific and annual reports on legacy materials should be evaluated. In the absence of relevant radiological or chemical records, legacy material should be monitored for different types of radiation, its radiological and chemical properties should be characterized and any hazards should be quantified.

7.67. Before the clearance of equipment for recycling or for disposal, it should be decontaminated to the level required by the regulatory body. Criteria for clearance applicable to many R&D facilities are set out in Schedule I of GSR Part 3 [7].

7.68. Paragraphs 7.69–7.71 provide guidance on the requirements and supporting recommendations on emergency preparedness and response contained in GSR Part 7 [11], GS-G-2.1 [12] and Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-2 [36] (as appropriate) and in paras 9.62–9.67 and in V.17 and V.18 of NS-R-5 (Rev. 1) [1] as they apply to R&D facilities.

7.69. The emergency arrangements established in accordance with paras 4.126–4.128 of this Safety Guide should consider the layout of the R&D facility site (i.e. the site may be composed of a large number of buildings and facilities).

7.70. The operating organization should carry out regular emergency exercises, some of which should involve off-site resources, to check the adequacy of the emergency arrangements, including the training and preparedness of on-site and off-site personnel and services including communications.

7.71. The emergency arrangements should be periodically reviewed and updated. Any lessons identified from operating experience, emergency exercises, modifications, periodic safety reviews, emergencies that have occurred at similar facilities, emerging knowledge and changes to regulatory requirements should be taken into account.

8. PREPARATION FOR DECOMMISSIONING

8.1. Decommissioning activities are to be performed with an optimized approach to achieving a progressive and systematic reduction in radiological hazards, and are undertaken on the basis of planning and assessment to ensure the safety of workers and the public and the protection of the environment, both during and after decommissioning operations; see Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6 [37], which establishes general safety requirements for the decommissioning of facilities.

8.2. The following measures should be applied at the design, construction and operational stages in the lifetime of an R&D facility to facilitate its eventual decommissioning:

(a) Design measures to prevent contamination from penetrating structural materials, such as pond liners;
(b) Physical and procedural methods to prevent the spread of contamination;
(c) Design features to facilitate decommissioning;
(d) Consideration of the implications for decommissioning resulting from modifications and experiments in the facility, when they are proposed;
(e) Identification of reasonably achievable changes to the facility design to facilitate or accelerate decommissioning;
(f) Comprehensive preparation of records for all significant activities and events at all stages of the facility's lifetime, archived in a secure and readily retrievable form, and indexed in a documented, logical and consistent manner;
(g) Minimization of the eventual generation of radioactive waste during decommissioning;
(h) Ensuring adequate financial resources for safe decommissioning.

8.3. The radiological hazard associated with the preparation for decommissioning of R&D facilities depends upon the type of work performed. Either this work should already be addressed by the existing decommissioning plan for the facility and experiments, or the plan should be subject to an appropriate review and modification before the decommissioning work begins. It should normally be expected that any temporary experimental apparatus inside Case 1 facilities would be dismantled and removed before operations cease. In terms of dealing with contaminated equipment, the following should be taken into account:

(1) In high activity cells or units, beta/gamma surface contamination may exist that requires prior decontamination by chemical or mechanical means (such as chemical rinses, sand blasting and using specialized tools). The objective should be to remove contamination where possible in order to reduce radiation levels to as low as possible to allow direct access to the equipment. If, after decontamination, dose rates remain high, remote handling should be used.
(2) In alpha liquid units, alpha surface contamination may require rinsing with chemical materials other than those used during operation.
(3) In alpha powder units, deposits of powder may remain that can be managed with appropriate personal protective equipment.

8.4. Where fissile material could be present, the requirements on criticality safety in paras V.19 and V.20 of NS-R-5 (Rev. 1) [1] apply.

PREPARATORY STEPS

8.5. The preparatory steps for the decommissioning process should include:

(1) Post-operational clean-out to remove all bulk quantities of radioactive material and other hazardous materials;
(2) Identification of contaminated parts of buildings and equipment, and radionuclides;
(3) Characterization of the types and levels of contamination;
(4) Decontamination of the facility to reach the levels required by the regulatory body for final decommissioning, or the lowest reasonably achievable level of residual contamination;
(5) Preparation of risk assessments and method statements for the licensing of the decommissioning process; Safety Assessment for the Decommissioning of Facilities Using Radioactive Material, IAEA Safety Standards Series No. WS-G-5.2 [38], contains recommendations on safety assessment for decommissioning.

8.6. In the event of decommissioning being significantly delayed after an R&D facility has been permanently shut down, safety measures should be applied to maintain the R&D facility in a safe and stable state, including measures to prevent criticality, spread of contamination and fire, and to maintain appropriate radiological monitoring. Consideration should be given to the need for a revised safety assessment for the shut down facility state and to apply knowledge management methods to ensure that the knowledge and experience of operators is retained in a durable and retrievable form. Efforts should be made to remove as much radioactive material or hazardous material from the facility as possible, before it is permanently shut down.

DECOMMISSIONING PROCESS

8.7. Specific guidance on the decommissioning process for R&D facilities is provided in Decommissioning of Medical, Industrial and Research Facilities, IAEA Safety Standards Series No. WS-G-2.2 [39]. Guidance that may be relevant to pilot plants can be found in Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-47 [40]. It should be ensured that personnel deployed for decommissioning of the R&D facility (the plant or the experimental equipment) are suitably experienced and qualified for such work. They should clearly understand the control regime under which they are working in order to maintain acceptable environmental conditions and to apply applicable health and safety standards.

8.8. During the decommissioning of contaminated areas, particular attention should be paid to:

(1) Avoiding the spread of contamination through the use of appropriate techniques and procedures. In particular, the amounts of liquids (such as water and chemicals) used for decontamination should be minimized in order to minimize the generation of secondary radioactive waste.
(2) Appropriate waste handling and packaging as well as planning for appropriate disposal of the waste.
(3) The safe processing and storage of contaminated waste material that cannot be disposed of immediately.
(4) Minimizing the generation of airborne contamination, rather than simply relying on personal protective equipment.

8.9. The extent of decontamination applied to enable the recycling of equipment or release of buildings or facilities from regulatory control should meet the criteria established by the regulatory body, in accordance with GSR Part 6 [37] and Schedule I of GSR Part 3 [7].

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Annex I

PROCESS ROUTE IN AN R&D FACILITY: LABORATORY SCALE (CASE 1)

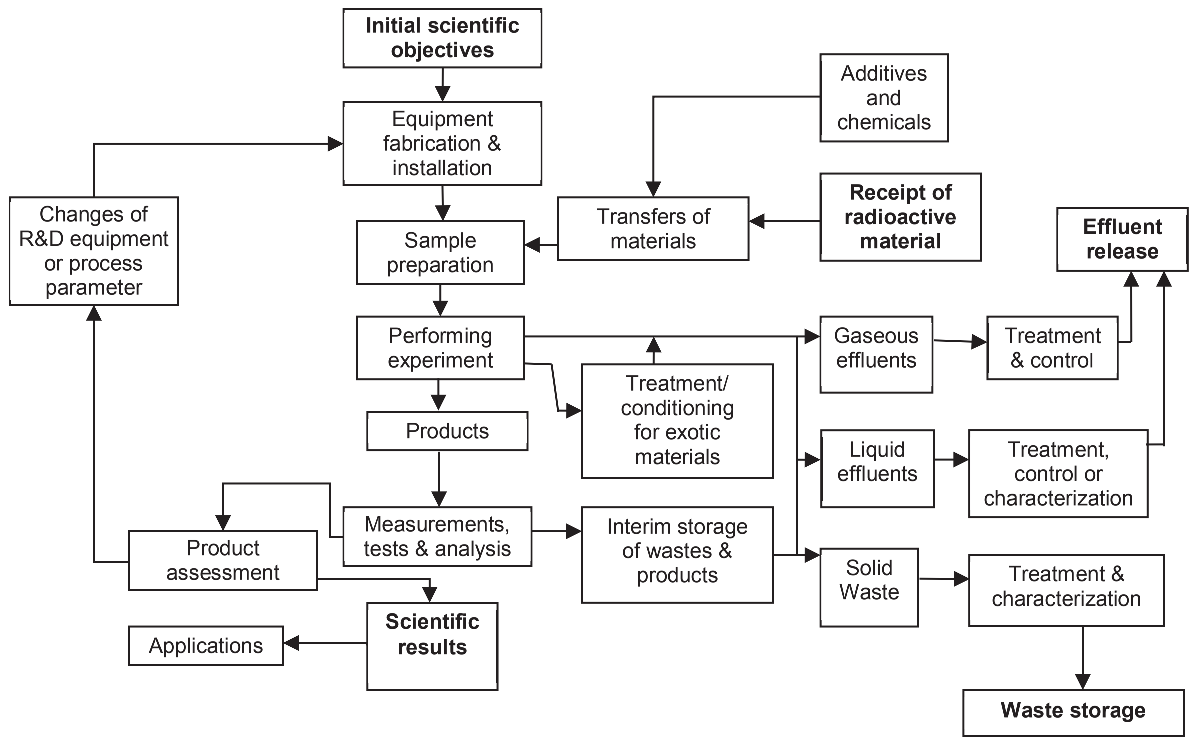

FIG. I–1. Diagram showing the general processes in an R&D facility operating at laboratory scale (Case 1).

Annex II

PROCESS ROUTE IN AN R&D FACILITY: PILOT SCALE (CASE 2)

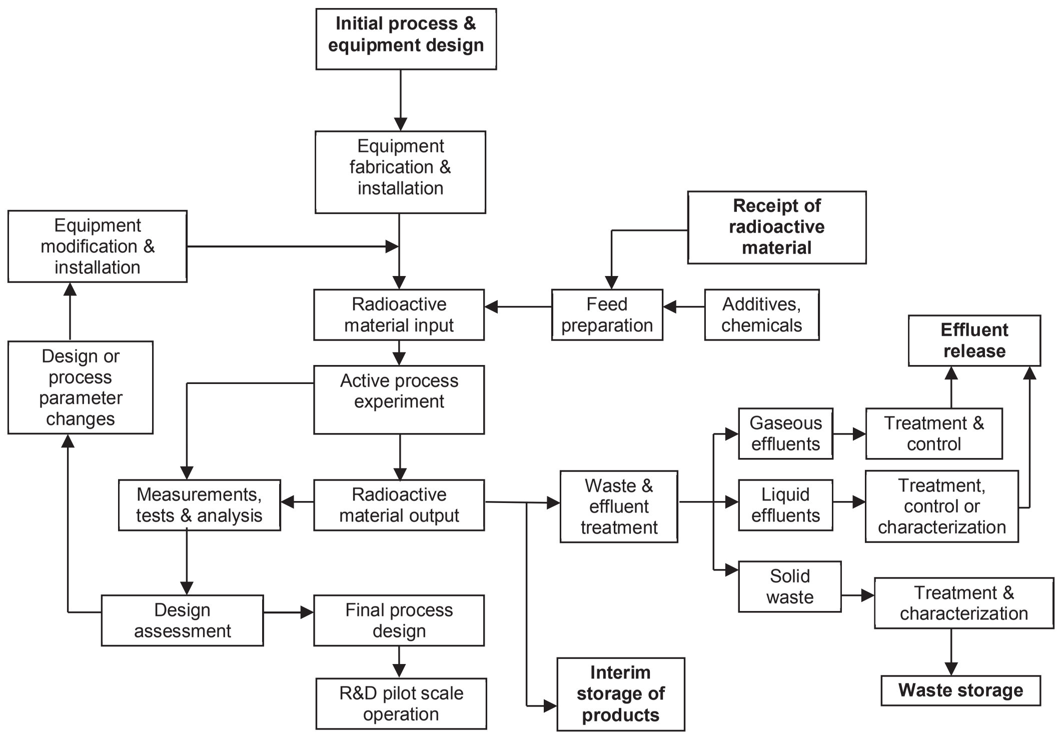

FIG. II–1. Diagram showing the general processes in an R&D facility operating at a pilot scale (Case 2).

Annex III

STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS

Main safety function:
(1) Prevention of criticality;
(2) Confinement of harmful materials, including the removal of decay heat, for the prevention of releases;
(3) Protection against external radiation exposure.

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Initial scientific objectives			1, 2 and 3	Application of Safety Principles Nos 4–9[1] Safety assessment of programmes and activities

[1] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Equipment fabrication and installation	Equipment ensuring geometry and moderation control Reflectors Neutron absorbers Detection and alarm systems	Criticality accident	1	Quality of the design and construction Installation according to the safety case and set procedures Accessibility/visibility to allow for periodic inspection, maintenance and checks
	Equipment ensuring mass, and concentration	Criticality accident	1	Quality of the design and construction with diverse and robust control of key parameters Installation according to the safety case and set procedures with realistic commissioning tests

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Building, fume hoods, gloveboxes, hot cells and interim storage Ventilation, filters	Contamination Loss of integrity	2	Quality of the design and construction Use of fail-safe designs where possible Installation according to safety case and set procedures Realistic commissioning tests Measurement points for airflow/pressure Accessibility/visibility to allow for periodic inspection, maintenance and checks of structural integrity
	Hot cells or shielded gloveboxes	Insufficient shielding	3	Quality of the design and construction Operational limits and conditions on radiation protection Validation of the shielding suitability during commissioning

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Receipt of radioactive material	Transportation means	Degradation of criticality safety margin	1 (fissile material only)	Transport rules, regulations and procedures[a] Verification by recipient in accordance with operational limits and conditions
	Measurement devices for isotopic and chemical composition	Violation of acceptance criteria Unexpected or exotic material (see para. 2.2(e))	1, 2 and 3	Suitably qualified and experienced personnel Non-destructive analysis or sampling of imported fissile material for isotopic or chemical characterization Calibration of the measurement devices
	Transportation means	Collision Fire Exposure	2 and 3	Transport rules, regulations and procedures On-site transportation rules Authorized personnel Smear tests, brake tests

[a] Rules for the safe transport of radioactive materials and samples at the facility are defined by the operator or IAEA safety standards for transport[2] may be applied in a graded manner.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. SSR-6, IAEA, Vienna (2012).

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Licensed container	Leakage Overpressure or explosion, e.g. hydrogen due to radiolysis effect	2	On-site transportation rules Suitably qualified and experienced personnel Verification of use of right container Visual inspection of container and its seals Correct labelling Smear tests, pressure tests
	Shielding Licensed container	Increased dose to R&D facility personnel	3	Transport rules, regulations and procedures On-site transportation rules Suitably qualified and experienced personnel Verification of use of right container Verification by recipient Visual inspection and radiation monitoring
Additives and chemicals including gases	Engineering fittings e.g. gas bottles Standardized containers	Fire, explosion and toxicity	2 (industrial safety)	Positive identification of supplies Checks of material safety data sheets Suitably qualified and experienced personnel for receipt, storage, use and disposal of chemicals

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Transfers of nuclear and non-nuclear materials	For nuclear materials; fume hoods or coupling device to hot cells or gloveboxes For chemicals: as defined by the materials safety data sheets	Breach of the integrity of containment leading to inadvertent release	2 and 3	For nuclear materials: R&D facility safety case limits Operating procedures consistent with safety analysis For chemicals, conformation to material safety data sheets Radiation protection controls Chemical hazard controls
Sample/feed preparation	Chemical analysis, weighing devices	Non-acceptable k_{eff}	1	Procedures, criticality control measures, moderator limits, etc. Calibration of structures systems and components
	Criticality accident alarm system	Unavailability of alarm	1	Procedures controlling transfers of fissile materials, personnel access and egress
	Fume hoods, hot cells or gloveboxes	Breach of containment	2	Maintenance and periodic testing Permissible pressure
	Fume hoods, hot cells or shielded gloveboxes	Insufficient shielding	3	Maintenance and periodic checks for purposes of radiation protection

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Performance of experiments	Calibrated equipment Diverse equipment ensuring mass, geometry, moderation control Reflectors Neutron absorbers Detection and alarm systems	Non-acceptable k_{eff} Double batching Inadvertent accumulation of fissile material	1	Operational limits and conditions where necessary Independent double check by suitably qualified and experienced persons especially for mass and concentration of fissile materials Stringent implementation of quality assurance including maintenance and periodic inspection, e.g. of reflectors Questioning attitude
	Fume hoods, hot cells or gloveboxes Pressure monitoring/ recording	Breach of containment	2	Effective isolation procedures Maintenance and periodic testing
	Emergency power supply	Loss of power	2	System dependent procedures e.g. for low battery voltage Maintenance and periodic testing

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Fire protection system	Uncontrolled fire Accumulations of flammable materials, blocked exits	2	Note any potential for pyrophoric materials Maintenance and periodic testing Good housekeeping
	Fume hoods, hot cells or shielded gloveboxes	Insufficient shielding Buildup of radioactive materials	3	Maintenance and periodic checks for the purposes of radiation protection Good housekeeping
Products	Criticality detection and alarm system or neutron measurement device Criticality accident alarm system	Non-acceptable k_{eff}	1	Anticipation and verification of characteristics of products in line with operational limits and conditions —assessment if significant change in density, chemical and physical form e.g. precipitation Maintenance and periodic testing of equipment

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Control of discharge of powders or fluids from the equipment to hot cell, glovebox or waste Containers, cabinet, well, wet storage	Fire and explosion Breach of containment	2	Operational limits and condition Implementation of conservative procedures Checks for purposes of radiation protection, smear tests, pool water activity etc. Put the R&D facility in a safe state Maintenance and periodic testing Potential bio-hazards
Measurements, tests and analysis	Safety related instruments and control systems	Unexpected outcome. Non-acceptable k_{eff}	1	Criticality assessment defining operational limits and conditions Double contingency principle Calibration
	Safety related instrumentation and control systems e.g. pressure, radiation	Unexpected outcome	2	Adequacy of the material with the safety case Hazard assessment defining operational limits and conditions Calibration, regular inspections Maintenance and periodic testing

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Application	None	Hazard transferred to a third party (customer of the facility)	1, 2 and 3	Quality assurance applied to work conducted by the R&D facility with some transfer of knowledge and safety information to the user: — Product identified (labelled) and capable of being safely handling — Documentation and training of third parties and customers — Checks on export packages prior to use Responsibility for the subsequent safety of the product and its application transferred from the R&D facility to user or third party
Gaseous effluents	Off-gas treatment units, iodine filters and HEPA filters Differential pressure measurements and controls	Breach of containment Fan malfunction	2	Periodic monitoring and testing as defined by procedures and regulatory limits

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Scrubbers, HEPA filters, connections and casings	Contact dose on filter casing Deposition of radioactive particulate	3	Periodic checks for the purposes of radiation protection, as defined by procedures and regulatory limits
Liquid effluents	Ion exchange resins and extraction	Abnormal presence of fissile material	1	Periodic testing by gamma/neutron counting Accountability Smear tests Criticality controls
	Connections, equipment for retention of filtering medium or resin, e.g. prevention of backflow	Presence of leak	2	Measurements, periodic testing as defined by procedures and regulatory limits Tightness, fail-safe design Checks for the purposes of radiation protection
	Filters, ion exchange resins, extraction evaporation	Buildup of radioactive materials on media and increasing risk to R&D facility operators	3	Good planning, periodic checks for the purposes of radiation protection, as defined by procedures and regulatory limits

TABLE III–1. STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
	Containers	Contact dose on containers Breach of containment	2	Measurements, e.g. smear test, periodic testing as defined by procedures and regulatory limits
	Shielding on containers	Exposure from packaging and increased risk to R&D facility operators	3	Checks for the purposes of radiation protection, as defined by procedures, records of radioactive materials and regulatory limits for discharges

Annex IV

EXAMPLES OF OPERATIONAL LIMITS AND CONDITIONS

Area or operation	Example operational limit or condition
Radiation protection in hot cells or shielded gloveboxes	No more than 100 millilitres of radioactive product or 1 TBq iodine-131 equivalent allowed in a particular cell at any one time
Verification of receipt for fissile material	The consignment number, weight and isotopic composition on the label are recorded in the ‘samples-in’ system, and the sample’s as-received weight is measured and recorded, enrichments over 4.0% or discrepancies in the weight greater than 100 mg are reported to the supervisor
Criticality control of process	The H/U atomic ratio not exceeding 8.4 at any time
Criticality control of process product	No more than 10 mg/L solids in daily product sample as measured by the analytical service department
Individual experiment	No more than 10 L of hydrogen used in the glovebox in any one experiment
X ray machines	The X ray machine is not energized unless the door to the X ray cell is closed and the interlock is functional

CONTRIBUTORS TO DRAFTING AND REVIEW

Bardelle, P.	Commissariat à l’énergie atomique et aux énergies alternatives, France
Carr, V.M.	International Atomic Energy Agency
Gater, R.	International Atomic Energy Agency
Kotzalas, M.	Nuclear Regulatory Commission, United States of America
Mester, W.	Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH, Germany
Nocture, P.	International Atomic Energy Agency
Norato, M	Nuclear Regulatory Commission, United States of America
Orabi, G.	National Centre of Nuclear Safety and Radiation Control, Egypt
Sigit	National Nuclear Energy Agency, Indonesia
Williamson, R.	National Nuclear Laboratory, United Kingdom
Uchiyama, G.	Japan Atomic Energy Agency, Japan

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