IAEA

International Atomic Energy Agency

IAEA SAFETY STANDARDS

No. SSG-43 (Rev. 1)

for protecting people and the environment

Safety of Nuclear Fuel Cycle Research and Development Facilities

SPECIFIC SAFETY GUIDE

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 web site:

http://www-ns.iaea.org/standards/

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.

RELATED PUBLICATIONS

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.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

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

SAFETY OF NUCLEAR FUEL CYCLE RESEARCH AND DEVELOPMENT FACILITIES

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS, THE
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CABO VERDE
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
COMOROS
CONGO
COOK ISLANDS
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ESWATINI
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GAMBIA, THE
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUINEA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS, KINGDOM OF THE
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORTH MACEDONIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAINT KITTS AND NEVIS
SAINT LUCIA
SAINT VINCENT AND THE GRENADINES
SAMOA
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOMALIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
TOGO
TONGA
TRINIDAD AND TOBAGO
TUNISIA
TÜRKİYE
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

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 (Rev. 1)

SAFETY OF NUCLEAR FUEL CYCLE RESEARCH AND DEVELOPMENT FACILITIES

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2025

Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
tel.: +43 1 2600 22529 or 22530
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
June 2025
STI/PUB/2107
https://doi.org/10.61092/iaea.lgmj-zs2t

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Safety of nuclear fuel cycle research and development facilities / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2025. | Series: IAEA safety standards series, ISSN 1020–525X ; no. SSG-43 (Rev. 1) | Includes bibliographical references.
Identifiers: IAEAL 25-01735 | ISBN 978-92-0-102025-3 (paperback : alk. paper) | ISBN 978-92-0-102125-0 (pdf) | ISBN 978-92-0-102225-7 (epub)
Subjects: LCSH: Nuclear facilities — Safety measures. | Research and development projects. | Hazardous substances — Safety measures. | Nuclear fuels.
Classification: UDC 621.039 | STI/PUB/2107

FOREWORD

by Rafael Mariano Grossi
Director General

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

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

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

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the Safety Requirements. The principles are expressed as 'must' statements.

Safety Requirements

Safety Requirements are governed by the objective and principles of the Safety Fundamentals. They establish the requirements to be met to ensure the protection of people and the environment, both now and in the future. The format and style of the Safety Requirements facilitate their use for the establishment of a national regulatory framework. Requirements are presented as 'overarching' requirements[2] in bold, followed by a number of associated requirements; all are equally important and are expressed as 'shall' statements.

Safety Guides

Safety Guides provide recommendations on how to comply with the Safety Requirements, indicating an international consensus that it is necessary to take the

---

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

[2] The IAEA Regulations for the Safe Transport of Radioactive Material do not include overarching requirements.

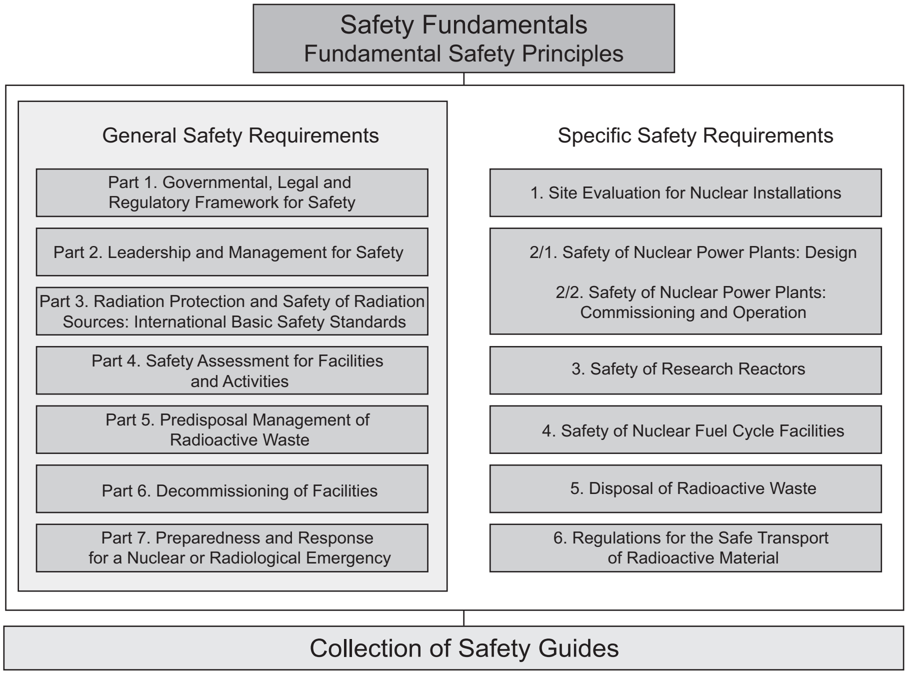

FIG. 1. The long term structure of the IAEA Safety Standards Series.

measures recommended (or alternative measures that achieve the same level of protection). Safety Guides present international good practices and, increasingly, best practices. The recommendations provided in Safety Guides are expressed as 'should' statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

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

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

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

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

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

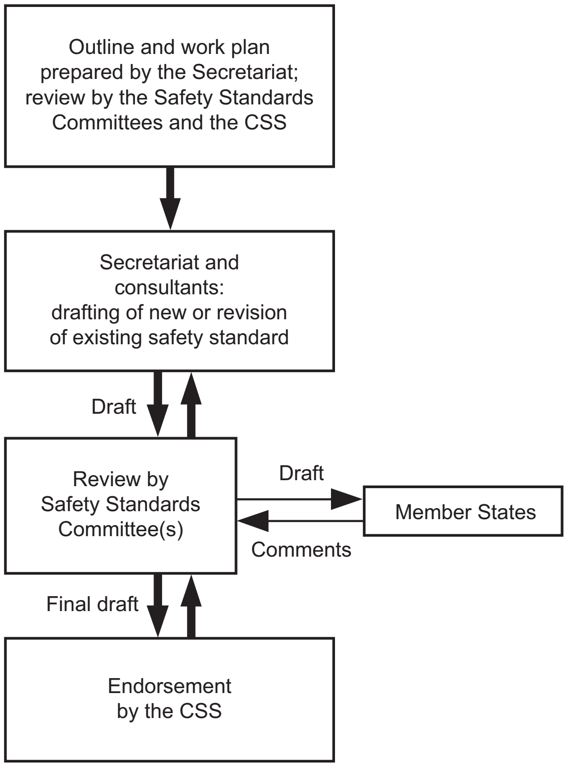

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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

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

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

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

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Requirements for safety in all stages of the lifetime of a nuclear fuel cycle facility are established in IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [1].

1.2. This Safety Guide provides specific recommendations on the safety of nuclear fuel cycle research and development (R&D) facilities.

1.3. Nuclear fuel cycle R&D facilities may receive, handle, process and store various nuclear and other radioactive materials, including uranium, other actinides, fission products or activated materials in multiple physical forms such as powders, liquids and gases. These materials can present diverse hazards, such as nuclear and radiological hazards, toxic and chemical hazards (e.g. hydrofluoric acid, uranium hexafluoride, ammonia), and explosive or flammable hazards from reactive materials (e.g. hydrogen, nitric acid, metallic powders). Another common feature of such facilities is the diversity of researchers and operating personnel, organized in different teams with potentially different training, expertise, experience, expectations and goals.

1.4. Nuclear fuel cycle R&D facilities can operate over extended periods of time to provide analytical services, materials and testing services, and the inventories of radioactive and other hazardous materials in such facilities can be significant. Such facilities are subject to the safety requirements established in SSR-4 [1] relating to the management of nuclear fuel cycle facilities and activities in general, as well as to the safety requirements for specific types of nuclear fuel cycle facility at which similar operations are performed.

1.5. Activities at nuclear fuel cycle R&D facilities may relate to any stage of the nuclear fuel cycle, from fundamental and applied research to fuel processing, material examination, fuel safety, chemical analysis and the development of instrumentation. A variety of physicochemical processes may be employed to study different types of fuel or material that might be hazardous. Particular care is needed when researching new or novel processes and when establishing the safety of processes under development, to ensure that the safety assessment and safety measures are appropriate. The normal practice of eliminating unknown factors relating to safety is not always possible in some nuclear fuel cycle R&D activities.

In such cases, additional margins of safety and a more cautious application of a graded approach are appropriate.

1.6. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-43, Safety of Nuclear Fuel Cycle Research and Development Facilities[1].

OBJECTIVE

1.7. The objective of this Safety Guide is to provide recommendations on safety in the siting, design, construction, commissioning, operation, and preparation for decommissioning of nuclear fuel cycle R&D facilities to meet the relevant requirements established in SSR-4 [1].

1.8. The recommendations in this Safety Guide are aimed primarily at operating organizations of nuclear fuel cycle R&D facilities, regulatory bodies, designers and other relevant organizations.

SCOPE

1.9. Safety requirements for nuclear fuel cycle facilities (i.e. facilities for uranium ore refining, conversion, enrichment, reconversion[2], storage of fissile material, fabrication of fuel (including mixed oxide fuel), storage and reprocessing of spent fuel, associated conditioning, and storage of waste, and facilities for fuel cycle related R&D) are established in SSR-4 [1]. This Safety Guide provides recommendations on meeting these requirements for nuclear fuel cycle R&D facilities.

1.10. This Safety Guide applies to two types of nuclear fuel cycle R&D facility, denoted as Case 1 and Case 2. These are described below and illustrated in Annexes I and II:

(a) Case 1: Facilities involving 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

---

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Research and Development Facilities, IAEA Safety Standards Series No. SSG-43, IAEA, Vienna (2017).

[2] Also referred to as ‘deconversion’.

(before and after reactor irradiation), and investigations of nuclear materials and wastes arising from new processes;

(b) Case 2: Facilities involving R&D on processes and equipment envisaged for use on an industrial scale (e.g. pilot facilities for waste treatment).

This Safety Guide also applies to the individual experiments (activities) undertaken within Case 1 and Case 2 facilities, using a graded approach.

1.11. This Safety Guide does not apply to irradiators, accelerators, research reactors, subcritical assemblies or radioisotope production facilities.

1.12. This Safety Guide covers the safety of nuclear fuel cycle R&D facilities and the protection of workers, the public and the environment. It does not consider ancillary processing facilities in which waste and effluents are treated, conditioned, stored or disposed of, except insofar as all waste generated has to comply with Requirement 24 (and paras 6.94–6.99) and Requirement 68 (and paras 9.102–9.108) of SSR-4 [1] and with the requirements established in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [2].

1.13. The recommendations on ensuring criticality safety in a nuclear fuel cycle R&D facility in this Safety Guide supplement the more detailed recommendations provided in IAEA Safety Standards Series No. SSG-27 (Rev. 1), Criticality Safety in the Handling of Fissile Material [3].

1.14. The implementation of safety requirements on the governmental, legal and regulatory framework and in relation to regulatory oversight (e.g. requirements for the authorization process, regulatory inspection and regulatory enforcement), as established in IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [4], is not addressed in this Safety Guide.

1.15. Additional recommendations relevant to Case 2 nuclear fuel cycle R&D facilities are provided in the IAEA Safety Guides for the corresponding type of nuclear fuel cycle facility. For example, additional recommendations applicable to fuel fabrication pilot facilities are provided in IAEA Safety Standards Series No. SSG-6 (Rev. 1), Safety of Uranium Fuel Fabrication Facilities [5].

1.16. This Safety Guide does not include nuclear security recommendations for a nuclear fuel cycle R&D facility. Recommendations on nuclear security are provided in IAEA Nuclear Security Series No. 13, Nuclear Security Recommendations on

Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [6], and guidance is provided in IAEA Nuclear Security Series No. 27-G, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5) [7] and in IAEA Nuclear Security Series No. 35-G, Security During the Lifetime of a Nuclear Facility [8]. However, this Safety Guide does include recommendations on managing interfaces between safety, nuclear security and the State system of accounting for and control of nuclear material.

STRUCTURE

1.17. Section 2 provides general safety recommendations for a nuclear fuel cycle R&D facility. Section 3 provides recommendations on the development of a management system for such a facility and the activities associated with it. Section 4 provides recommendations on the safety aspects to be considered in the evaluation and selection of a site for a nuclear fuel cycle R&D facility to minimize any environmental impact. Section 5 provides recommendations on safety in the design stage of a nuclear fuel cycle R&D facility, including recommendations on the safety analysis for operational states and accident conditions and on radioactive waste management and other design considerations. Section 6 provides recommendations on safety in the construction stage of a nuclear fuel cycle R&D facility, and Section 7 provides recommendations on safety in the commissioning stage. Section 8 provides recommendations on safety in the operation of a nuclear fuel cycle R&D facility, including recommendations on the management of operations; maintenance and periodic testing; control of modifications; criticality control; radiation protection; fire, chemical and industrial safety; the management of waste and effluents; and emergency preparedness and response. Section 9 provides recommendations on preparing for the decommissioning of a nuclear fuel cycle R&D facility.

1.18. Annexes I and II show the typical process routes for Case 1 and Case 2 nuclear fuel cycle R&D facilities, respectively. Annex III provides examples of structures, systems and components (SSCs) important to safety in nuclear fuel cycle R&D facilities, grouped in accordance with the process areas. Examples of operational limits and conditions for nuclear fuel cycle R&D facilities are provided in Annex IV.

2. HAZARDS IN NUCLEAR FUEL CYCLE R&D FACILITIES

2.1. In nuclear fuel cycle R&D facilities, fissile material and other radioactive material are present in different forms with diverse physical and chemical characteristics. The main hazards are potential criticality, loss of confinement, radioactive contamination, radiation exposure (both internal exposure and external exposure), fire, flooding, chemical hazards and explosion hazards.

2.2. Nuclear fuel cycle R&D facilities are often highly reliant on human operations. Notwithstanding this, the systems that should be designed to continue operating in order to maintain the facility in a safe state[3] during and after an event include the following:

(a) Heat removal systems that 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 the release of radioactive material from the facility;
(c) Criticality safety systems;
(d) Systems that provide chemical safety under high temperature conditions;
(e) Inert gas feed systems, for example, to hot cells or gloveboxes;
(f) Real time radiation monitoring systems.

2.3. Factors relevant to the safety of nuclear fuel cycle R&D facilities include the following:

(a) The radiological consequences of a release of radioactive material under accident conditions can be significant.
(b) Fissile material (if present) can 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 SSG-27 (Rev. 1) [3]).
(c) The radiation levels and the risk of internal exposure and external exposure are significantly increased when irradiated fuel is used.

---

[3] As defined in SSR-4 [1], a safe state is a facility state, following an anticipated operational occurrence or accident conditions, in which the nuclear fuel cycle facility is subcritical and the main safety functions can be ensured and maintained stable for a long time.

(d) The chemical toxicity of material used in nuclear fuel cycle R&D facilities has to be considered. For example, uranium hexafluoride, if released, reacts with the moisture in the air to form hydrogen fluoride and soluble uranyl fluoride. Therefore, the safety analysis should also address the impacts resulting from these chemicals and their potential mixing (e.g. in liquid effluent streams).
(e) The presence of products, subproducts or waste arising from R&D programmes involving exotic nuclear materials such as the following:
  (i) Non-standard mixed oxide or uranium dioxide fuel or new fuel matrices (e.g. carbides, nitrides, metallic forms);
  (ii) Isotopes with particular constraints for disposal (e.g. long half-life transuranic isotopes, fission products and activated materials such as trace materials in fuel cladding);
  (iii) Materials without an agreed national disposal route (e.g. graphite and aluminium in waste);
  (iv) Uranium with enrichment levels higher than 5%;
  (v) Materials in the thorium fuel cycle that contain high energy gamma emitters (e.g. some ^{232}U decay products).

2.4. Nuclear fuel cycle R&D facilities range from small scale academic research facilities to large nuclear pilot facilities. As such, the application of a graded approach to meeting safety requirements is very important (see paras 1.10 and 2.15 of SSR-4 [1]).

3. MANAGEMENT SYSTEM FOR NUCLEAR FUEL CYCLE R&D FACILITIES

3.1. A management system that integrates the safety, health, environmental, security, quality, human and organizational factor, societal and economic elements is required to be implemented by the operating organization (see Requirement 4 of SSR-4 [1]). The integrated management system should be established early in the lifetime of a nuclear fuel cycle R&D facility to ensure that safety measures are specified, implemented, monitored, audited, documented and periodically reviewed throughout the lifetime of the facility.

3.2. Requirements for the management system are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [9]. Associated recommendations are provided in IAEA Safety Standards Series

Nos GS-G-3.1, Application of the Management System for Facilities and Activities [10]; GS-G-3.5, The Management System for Nuclear Installations [11]; GSG-16, Leadership, Management and Culture for Safety in Radioactive Waste Management [12]; and TS-G-1.4, The Management System for the Safe Transport of Radioactive Material [13].

3.3. The management system is required to take into account the interfaces between safety and nuclear security (see para. 1.3 of GSR Part 2 [9]). Requirement 75 of SSR-4 [1] states:

> "**The interfaces between safety, security and the State system of accounting for, and control of, nuclear material shall be managed appropriately throughout the lifetime of the nuclear fuel cycle facility. Safety measures and security measures shall be established and implemented in a coordinated manner so that they do not compromise one another.**"

The activities for ensuring safety throughout the lifetime of a nuclear fuel cycle R&D facility involve different groups as well as interfaces with other areas, such as those relating to nuclear security and to the State system of accounting for and control of nuclear material. Activities with such interfaces should be identified in the management system and should be coordinated, planned and conducted to ensure effective communication and clear assignment of responsibilities. Communications regarding safety and nuclear security should ensure that confidentiality of information is maintained. This includes communications in relation to the system of nuclear material accounting and control, for which information security should be coordinated in a manner ensuring that safety and security measures are not compromised. Potential conflicts between the transparency of information relating to safety matters and the protection of information for nuclear security reasons are required to be addressed (see para. 4.10 of GSR Part 2 [9]).

3.4. In determining how the requirements of the management system for the safety of a nuclear fuel cycle R&D facility are to be applied, a graded approach based on the relative importance to safety of each item or process is required to be used (see Requirement 7 and para. 4.15 of GSR Part 2 [9]).

3.5. The management system is required to support the development and maintenance of a strong safety culture (see Requirement 12 of GSR Part 2 [9]), and should address all aspects of safety (including radiation safety, criticality safety[4], chemical safety, fire safety and industrial safety). Special consideration should be given to all processes covered by the management system that are associated with handling plutonium, including (where appropriate) transition to hot commissioning and assignment of new staff to activities involving plutonium handling (see also para. 8.27 of SSR-4 [1]).

3.6. In accordance with paras 4.15–4.23 of SSR-4 [1], the management system is required to address four functional areas: management responsibility; resource management; process implementation; and measurement, assessment, evaluation and improvement. These areas may be summarized as follows:

(a) Management responsibility includes the support and commitment of management necessary to achieve the safety objectives of the operating organization in such a manner that safety is not compromised by other priorities.
(b) Resource management includes the measures necessary to ensure that the resources essential to the implementation of the safety policy and the achievement of the safety objectives of the operating organization are identified and made available.
(c) Process implementation includes the activities and tasks necessary to achieve the safety goals of the organization.
(d) Measurement, assessment, evaluation and improvement provide an indication of the effectiveness of management processes and work performance compared with objectives or benchmarks; it is through measurement and assessment that opportunities for improvement can be identified.

MANAGEMENT RESPONSIBILITY FOR A NUCLEAR FUEL CYCLE R&D FACILITY

3.7. The prime responsibility for the safety of a nuclear fuel cycle R&D facility, including criticality safety, rests with the operating organization (see Requirement 2 of SSR-4 [1]). The senior management of such a facility is required to demonstrate leadership for and commitment to safety (see para. 3.1 of GSR Part 2 [9]). In accordance with para. 4.11 of GSR Part 2 [9], the management system for a nuclear fuel cycle R&D facility is required to clearly specify the organizational structures, processes, responsibilities, accountabilities, levels of authority and interfaces within the organization and with external organizations.

[4] Further recommendations on the management system for criticality safety are provided in paras 2.17–2.40 of SSG-27 (Rev. 1) [3].

3.8. The documentation of the management system is required to describe the interactions among the individuals managing, performing and assessing the adequacy of the processes and activities important to safety (see para. 4.16 of GSR Part 2 [9]). The documentation should also cover other management measures, including planning, scheduling and resource allocation (see also para. 9.9 of SSR-4 [1]).

3.9. Paragraph 4.15 of SSR-4 [1] states:

> "[T]he management system shall include provisions for ensuring effective communication and clear assignment of responsibilities, in which accountabilities are unambiguously assigned to individual roles within the organization and to suppliers, to ensure that processes and activities important to safety are controlled and performed in a manner that ensures that safety objectives are achieved."

The management system should include arrangements for empowering relevant personnel to stop unsafe operations at the nuclear fuel cycle R&D facility.

3.10. The operating organization of a nuclear fuel cycle R&D facility is required to ensure that safety assessments and analyses are conducted, documented and updated (see Requirement 5 of SSR-4 [1]). Requirements for safety assessment are established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [14].

3.11. The operating organization of a nuclear fuel cycle R&D facility is required to audit all safety related matters on a regular basis (see paras 4.2(d) and 4.23 of SSR-4 [1]). This includes examination of the arrangements for emergency preparedness and response at the facility, such as emergency communications and evacuation routes (including signage). Checks should be performed by the nuclear criticality safety staff who performed the criticality safety analyses to confirm that the data used and the implementation of criticality safety measures are correct. Audits should be performed by personnel who are independent of those that performed the safety assessments or conducted the safety activities. The data from these audits should be documented and submitted for management review and for action, if necessary.

RESOURCE MANAGEMENT FOR A NUCLEAR FUEL CYCLE R&D FACILITY

3.12. The senior management of the operating organization is required to determine the competences and resources (both human and financial) for the safe operation of the nuclear fuel cycle R&D facility (see Requirement 9 of GSR Part 2 [9]). The senior management is also required to ensure that suitable training is conducted (see para. 4.23 of GSR Part 2 [9]). The management of the operating organization should also have frequent personal contact with personnel, including observing work in progress.

3.13. Requirement 58 of SSR-4 [1] states that "**The operating organization shall ensure that all activities that may affect safety are performed by suitably qualified and competent persons.**" The operating organization is required to ensure that these personnel receive training and refresher training at suitable intervals, appropriate to their level of responsibility (see paras 9.38–9.47 of SSR-4 [1]). In particular, personnel involved in activities with fissile material (both uranium and plutonium), with radioactive material including waste, or with chemicals should understand the nature of the hazard posed by these materials and how the risks are controlled by the established safety measures, operational limits and conditions, and operating procedures.

3.14. Requirement 11 of GSR Part 2 [9] states that "**The organization shall put in place arrangements with vendors, contractors and suppliers for specifying, monitoring and managing the supply to it of items, products and services that may influence safety.**" The management system for a nuclear fuel cycle R&D facility is required to include arrangements for procurement (see paras 4.33–4.36 of GSR Part 2 [9]). The operating organization is also required to ensure that suppliers of items and resources important to safety have an effective management system (see para. 4.16(b) of SSR-4 [1]). To meet these requirements, the operating organization should conduct audits of the management systems of suppliers.

PROCESS IMPLEMENTATION FOR THE MANAGEMENT SYSTEM FOR A NUCLEAR FUEL CYCLE R&D FACILITY

3.15. Requirement 63 of SSR-4 [1] states:

> "**Operating procedures shall be developed that apply comprehensively for normal operation, anticipated operational occurrences and accident conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body.**"

Paragraph 9.66 of SSR-4 [1] states that "Operating procedures shall be developed for all safety related operations that may be conducted over the entire lifetime of the facility." The operating procedures should specify all the parameters at the nuclear fuel cycle R&D facility that are intended to be controlled and the performance criteria that should be fulfilled.

3.16. Any proposed modification to an existing nuclear fuel cycle R&D facility, or any proposed new activity, is required to be assessed in terms of its implications for existing safety measures and approved prior to implementation (see para. 9.57 of SSR-4 [1]). Modifications of safety significance are required to be subjected to safety assessment and regulatory review and, where necessary, they are required to be authorized by the regulatory body before they are implemented (see paras 9.57(d), 9.57(h) and 9.59 of SSR-4 [1]). The documentation for the facility or activity is required to be updated to reflect modifications (see paras 9.57(f) and 9.57(g) of SSR-4 [1]). All relevant operating personnel, including supervisors, should receive adequate training on the modifications.

MEASUREMENT, ASSESSMENT, EVALUATION AND IMPROVEMENT OF THE MANAGEMENT SYSTEM FOR A NUCLEAR FUEL CYCLE R&D FACILITY

3.17. Requirement 13 of GSR Part 2 [9] states that "**The effectiveness of the management system shall be measured, assessed and improved to enhance safety performance, including minimizing the occurrence of problems relating to safety.**"

3.18. The audits performed by the operating organization (see para. 3.11), as well as proper control of modifications (see para. 3.16), are particularly important for ensuring the safety of the nuclear fuel cycle R&D facility. The results of audits are required to be evaluated by the operating organization and corrective actions are required to be taken where necessary (see para. 4.2(d) of SSR-4 [1]).

3.19. Deviation from operational limits and conditions, deviations from operating procedures, and unforeseen changes in process conditions that could affect criticality safety are required to be reported and promptly investigated by the operating organization of the nuclear fuel cycle R&D facility, and the operating organization is required to inform the regulatory body (see paras 9.34, 9.35 and 9.84 of SSR-4 [1]). The depth and extent of the investigation should be proportionate to the safety significance of the event, in accordance with a graded approach. The investigation should cover the following:

(a) An analysis of the causes of the deviation to identify lessons and to determine and implement corrective actions to prevent a recurrence;
(b) An analysis of the operation of the facility or of the conduct of the activity, including an analysis of human factors;
(c) A review of the safety assessment and analyses that were previously performed, including the safety measures that were originally established.

3.20. Requirement 73 of SSR-4 [1] states that "**The operating organization shall establish a programme to learn from events at the facility and events at other nuclear fuel cycle facilities and in the nuclear industry worldwide.**" Recommendations on operating experience programmes are provided in IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [15].

VERIFICATION OF SAFETY AT A NUCLEAR FUEL CYCLE R&D FACILITY

3.21. The safety of a nuclear fuel cycle R&D facility is required to be verified by means of comprehensive safety assessment and to be systematically assessed throughout the lifetime of the facility, for example, by periodic safety reviews (see Requirement 5 of SSR-4 [1]). The operating organization should establish a process for periodic safety reviews as part of the management system.

3.22. Requirement 6 of SSR-4 [1] states that "**An independent safety committee (or an advisory group) shall be established to advise the management of the operating organization on all safety aspects of the nuclear fuel cycle facility.**" The safety committee of a nuclear fuel cycle R&D facility should have members, or access to persons, who are suitably qualified and experienced in relevant areas, including human factors, criticality safety and radiation protection. Such persons should be available during commissioning and operation (including modifications) of the facility.

4. SITE EVALUATION FOR NUCLEAR FUEL CYCLE R&D FACILITIES

4.1. Requirements for site evaluation for nuclear fuel cycle R&D facilities are provided in IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [16], and recommendations are provided in associated Safety Guides, such as IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations [17].

4.2. The site evaluation process for a nuclear fuel cycle R&D facility will depend on a large number of variables. At the earliest stage of planning a facility, a list of potential hazards due to external events (e.g. earthquakes, accidental aircraft crashes, fires, nearby explosions, floods, extreme weather conditions) is required to be developed, all significant hazards are required to be evaluated and the design basis for the facility is required to be carefully determined (see section 5 of SSR-4 [1]). In addition, the radiological risk posed by the facility to workers, the public and the environment in both operational states and accident conditions is required to be evaluated (see Requirement 12 of SSR-1 [16]).

4.3. The scope of the site evaluation for a nuclear fuel cycle R&D facility is established in Requirement 3 of SSR-1 [16] and paras 5.1–5.14 of SSR-4 [1] and should also reflect the hazards described in Section 2 of this Safety Guide.

4.4. A nuclear fuel cycle 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 nuclear fuel cycle R&D facilities are part of a larger site for which site evaluation criteria have already been determined. Interactions with facilities nearby should be considered, as follows:

(a) In the case of an existing nuclear facility, the criteria will normally be encompassed by the site evaluation studies for the existing facility. These existing evaluation studies should be verified.
(b) In the case of a non-nuclear site (e.g. a hospital, university or research centre), the main siting issue can often be the feasibility of the necessary

emergency arrangements, such as the arrangements for evacuation. This may involve specific design provisions or other emergency provisions in order to meet the requirements of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [18], and the associated recommendations provided in IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [19].

4.5. SSR-1 [16] and section 5 of SSR-4 [1] establish the requirements for site evaluation for a new nuclear fuel cycle facility as well as for existing facilities, to be applied in accordance with a graded approach. The application of a graded approach is expected to be especially relevant for nuclear fuel cycle R&D facilities; nevertheless, care should be taken and an adequate review and justification should be made for any graded application of the requirements for site evaluation. Particular attention should be paid to the following throughout the lifetime of the nuclear fuel cycle R&D facility:

(a) The appropriate monitoring and systematic evaluation of site characteristics;
(b) The periodic review of all identified natural and human induced external hazards, and their credible combinations, and of the site conditions in the design basis for the facility;
(c) The identification of and the need to take account of all foreseeable variations in the site evaluation data (e.g. new or planned significant industrial development, infrastructure or urban developments);
(d) Revision of the safety assessment report (in the course of a periodic safety review or the equivalent) to take account of on-site and off-site changes that could affect safety at the nuclear fuel cycle R&D facility, with account taken of all current site evaluation data and the development of scientific knowledge and evaluation methodologies and assumptions;
(e) Consideration of future changes to site characteristics that could have an impact on emergency arrangements and the ability to perform emergency response actions for the facility.

4.6. The population density and population distribution in the vicinity of a nuclear fuel cycle R&D facility are required to be considered in the site evaluation process to minimize any possible health consequences for people in the event of a release of radioactive material and hazardous chemicals (see Requirements 4 and 12 of SSR-1 [16]). Also, in accordance with Requirement 25 and paras 6.1–6.7 of SSR-1 [16], the dispersion in air and water of any radioactive material released from a nuclear fuel cycle R&D facility is required to be assessed, taking into account the orography, land cover and meteorological features of the region.

The environmental impact from the facility under all facility states is required to be evaluated (see para. 5.4 of SSR-4 [1]) and should meet the applicable site evaluation criteria.

4.7. Security advice is required to be taken into account in the selection of a site for a nuclear fuel cycle R&D facility (see para. 11.4 of SSR-4 [1]). For nuclear fuel cycle R&D facilities in which plutonium is handled, special attention should be given to the management of the interface between safety and nuclear security during site evaluation (see para. 5.2(d) and Requirement 75 of SSR-4 [1]). The selection of a site should take into account both safety and nuclear security aspects, to ensure that they do not compromise one another, and should be facilitated by experts from both safety and security.

4.8. The operating organization should maintain a full record of the decisions taken on the selection of a site for a nuclear fuel cycle R&D facility and of the reasons behind those decisions.

4.9. The site characteristics are required to be reviewed periodically for their adequacy and continued applicability during the lifetime of a nuclear fuel cycle R&D facility (see paras 5.13 and 5.14 of SSR-4 [1]). Any changes to these characteristics that might require a revision of the safety assessment should be identified and evaluated.

5. DESIGN OF NUCLEAR FUEL CYCLE R&D FACILITIES

MAIN SAFETY FUNCTIONS AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.1. Requirement 7 of SSR-4 [1] states:

> “**The design shall be such that the following main safety functions are met for all facility states of the nuclear fuel cycle facility:**
>
> **(a) Confinement and cooling of radioactive material and associated harmful materials;**
> **(b) Protection against radiation exposure;**
> **(c) Maintaining subcriticality of fissile material.**”

All these safety functions are likely to be applicable to Case 2 nuclear fuel cycle R&D facilities (see para. 1.10). The safety measures identified in the design of a nuclear fuel cycle R&D facility should comprise those items important to safety and operational limits and conditions that, when taken as a whole, fulfil these main safety functions.

5.2. Requirements for the confinement of radioactive material are established in Requirement 35 and paras 6.123–6.128 of SSR-4 [1]. In normal operation, internal exposure should be avoided by design, including by static and dynamic barriers and adequate zoning. The need to rely on personal protective equipment is required to be minimized (see para. 3.93 of IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [20]).

5.3. Requirements for heat removal are established in Requirement 39 and paras 6.157–6.159 of SSR-4 [1]. If significant decay heat is generated in the nuclear fuel cycle R&D facility, all thermal loads and processes should be given appropriate consideration in the design. Particular care should be paid to the provision of adequate cooling (using passive design features, if possible) in accident conditions. 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. The possibility of chemical reactions at high temperature or high pressure in sealed containers should also be considered, and provisions to manage this situation should be provided.

5.4. Requirements for protection against external exposure in the design of nuclear fuel cycle facilities are established in Requirement 36 and paras 6.129–6.134 of SSR-4 [1]. Depending on the specific design of a nuclear fuel cycle R&D facility and the inventory of radioactive material, a combination of source limitation, shielding, distance and time may be necessary for the protection of personnel within the facility. Particular attention (in both design and operation) should be paid to provisions for maintenance (see Requirements 26 and 65 of SSR-4 [1]).

5.5. Requirements for maintaining subcriticality in nuclear fuel cycle facilities are established in Requirement 38 and paras 6.138–6.156 of SSR-4 [1]. Recommendations on ensuring subcriticality in the handling of fissile material are provided in SSG-27 (Rev. 1) [3].

5.6. The handling of various types of radioactive material should be taken into consideration in the design of nuclear fuel cycle R&D facilities. Owing to the nature of the work done in such facilities, there are often design and engineering provisions that are flexible and adaptable to future uses, including the possible dismantling and reconfiguration of parts of the facility. These provisions should be designed with the following in mind:

(a) They should enhance safety.
(b) They should take into account the potential for ageing and degradation of items important to safety.
(c) They should ensure safety is maintained over the lifetime of the facility.
(d) They should be used for handling new types of radioactive material only if a modification proposal and safety assessment have been submitted.
(e) They should take into account the future decommissioning of the facility.

Design basis and safety analysis for a nuclear fuel cycle R&D facility

5.7. A design basis accident is a postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and a conservative methodology and for which releases of radioactive material are kept within acceptable limits (see Requirement 17 of SSR-4 [1]). All estimates of source terms should include allowance for the ingrowth of radioactive decay products (e.g. ^{241}Am) over the lifetime of the facility.

5.8. Requirements relating to the design basis for items important to safety and for the design basis analysis for a nuclear fuel cycle R&D facility are established in Requirements 14 and 20 of SSR-4 [1], respectively.

5.9. The specification of the design basis will depend on the potential radiological hazard associated with the nuclear fuel cycle R&D facility and will need to comply with design requirements as well as siting and other regulatory requirements. Consideration should be given to all internal hazards, external hazards and their credible combinations selected in the site evaluation phase and associated with the design basis for the facility. These hazards might include internal and external explosions (in particular, hydrogen explosions), chemical and toxic releases, internal and external fires, dropped loads and handling errors, earthquakes, extreme meteorological phenomena (in particular, flooding and tornadoes), accidental aircraft crashes, and other applicable external hazards as defined in the site evaluation report. A list of postulated initiating events to be considered for nuclear fuel cycle facilities is provided in the appendix to SSR-4 [1].

5.10. The specification for the design basis should take account of events that might be the consequence of other events, such as a flood following an earthquake, or multiple events initiated by an external event, such as fire or multiple leaks within the facility caused by an earthquake (see para. 6.61 of SSR-4 [1]).

Structures, systems and components important to safety at a nuclear fuel cycle R&D facility

5.11. Paragraph 6.21 of SSR-4 [1] states:

> "The design of the nuclear fuel cycle facility:
>
> …….
>
> (e) Shall provide for structures, systems and components and procedures to control the course of and, as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems."

Annex III to this Safety Guide presents examples of SSCs important to safety and possible challenges to safety functions for nuclear fuel cycle R&D facilities.

Confinement of radioactive material at a nuclear fuel cycle R&D facility

5.12. Containment is required to be the primary method for protection against the spreading of contamination at a nuclear fuel cycle facility (see para. 6.124 of SSR-4 [1]). To meet Requirement 35 of SSR-4 [1] for a nuclear fuel cycle R&D facility, both static and dynamic confinement need to be considered, as determined by the safety analysis, as follows:

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

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 the rooms around the fume hoods, hot cells and gloveboxes, and/or the building walls. The design of the static containment should take into account openings between the different confinement zones (e.g. doors, penetrations).

5.13. The nuclear fuel cycle R&D facility should be designed to promptly detect and retain any leakage of liquids from equipment, vessels and pipes and to recover the volume of liquid to the primary containment. This is important for both design and operation, especially where the first static barrier provides other safety functions (e.g. favourable geometry for criticality avoidance or exclusion of air for flammable liquids).

5.14. 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. 5.19).

5.15. Dynamic containment cannot be provided in some circumstances; for example, sealed containers and isolated equipment 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 of radioactive material 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 with (and at) the time and place of origin to avoid an unexpected release of contamination. Labels and containers may be colour coded to match equipment and pipework. Labels and barcodes may 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.

5.16. 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.

5.17. The design of confinement areas should include contamination monitoring devices covering all locations inside the nuclear fuel cycle 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.

5.18. The design of a nuclear fuel cycle R&D facility is required to facilitate maintenance and decontamination (see Requirement 26 and para. 6.96(a) of SSR-4 [1]). The design of the facility should employ compartmentalization as one of the means of optimizing protection and safety for such activities.

5.19. Airborne contamination (from liquids or dispersible solids) is required to be prevented or the level kept as low as reasonably achievable (see paras 6.120 and 6.123 of SSR-4 [1]). The ventilation system for a nuclear fuel cycle R&D facility should include filters, in series, to protect workers, the public and the environment by filtering the air in all facility states and to ensure the integrity of the static barriers (see also paras 6.127 and 6.128 of SSR-4 [1]). Filters should also be used when airflow passes through confinement barriers, for example, at cooling inlets and where air exits the facility.

5.20. Paragraph 6.123 of SSR-4 [1] states that "the design performance of ventilation 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 in accordance with accepted standards, such as those of the International Organization for Standardization and relevant national requirements.

5.21. The potential for the failure of a fully loaded filter in the ventilation system of a nuclear fuel cycle R&D facility should be considered. Additional standby fans and filters should be provided as specified in the safety analysis. These fans and filters should be capable of maintaining ventilation during filter changing. Fans should be supplied with emergency power so 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 the initiation of the standby ventilation; this period may be used to define an operational limit or condition. Local monitoring and alarm systems should be installed to alert operating personnel to system malfunctions that result in high or low flows or unintended differential pressures.

5.22. The number of transfer operations involving radioactive material should be minimized in the design of the facility. To reduce the complexity of transfer operations, nuclear fuel cycle R&D facilities should be designed to accommodate standardized means of movement and 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.

Radiation protection of persons and protection of the environment at a nuclear fuel cycle R&D facility

5.23. Protection against radiation exposure relies on an appropriate combination of controls on the magnitude of the source, on the dispersion of the source (i.e. confinement; see paras 5.12–5.22) and on parameters that contribute to internal exposure (see paras 5.31–5.34) and external exposure (see paras 5.35–5.37).

5.24. In the design of a nuclear fuel cycle R&D facility, consideration should be given to maintenance, calibration, periodic testing and inspection, with the aim of minimizing the dose to workers and other persons. Requirements for the design of items important to safety to facilitate maintenance of nuclear fuel cycle facilities are established in Requirement 26 of SSR-4 [1]. Examples of provisions to meet these requirements in a nuclear fuel cycle R&D facility include connection junctions at containment boundaries and easily cleanable surfaces.

5.25. The design of a nuclear fuel cycle facility is required to ensure that the accumulation of radioactive material — for example, in process equipment, fume hoods, gloveboxes, hot cells, and secondary systems such as ventilation ducts — is avoided (see paras 6.119(c) and 9.84 of SSR-4 [1]). Where necessary, provisions should be made for the removal or reduction of any such accumulated radioactive material.

5.26. In the design of a nuclear fuel cycle R&D facility, consideration is required to be given to the remote operation of services and experimental equipment where possible (see para. 6.130 of SSR-4 [1]).

5.27. Requirements for the designation of controlled areas and supervised areas are established in paras 3.88–3.92 of GSR Part 3 [20]. The classification assigned should be based initially on that used in the facility design (see para. 6.121 of SSR-4 [1]) and should be developed on the basis of advice from radiation protection personnel, as necessary. Individual contamination zones and the boundaries between them should be regularly checked and adjusted, if necessary, to reflect the radiological conditions. The designation of areas inherently uses a graded approach based on the radiation and contamination levels. However, the use of a graded approach should be carefully considered, as even small quantities of alpha emitting radioactive material might represent a significant contamination hazard.

5.28. Radiation protection in a nuclear fuel cycle R&D facility often relies on analytical data from samples. If possible, a monitoring method that does not involve 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. Requirement 67 and paras 9.90–9.101 of SSR-4 [1], which establish requirements for radiation protection during operation, including control of occupational exposure and control of contamination, also apply to equipment and procedures used for sample analysis at a nuclear fuel cycle R&D facility.

5.29. Paragraph 6.132 of SSR-4 [1] states that "Means of monitoring radiation levels shall be provided so that any abnormal conditions would be detected in a timely manner and personnel may be evacuated." Depending on the results of the safety assessment, the monitoring system for radiation protection in a nuclear fuel cycle R&D facility should consist principally of the following:

(a) Fixed area monitors (for gamma and neutron radiation) and stationary air samplers (for beta and gamma activity, and for alpha activity) for access and evacuation purposes;
(b) Mobile area monitors (for gamma and neutron radiation) and mobile air samplers (for beta and gamma activity, and for alpha activity) for personnel protection purposes and for evacuation purposes during maintenance;
(c) Personal dosimeters consistent with the types of radiation present in the nuclear fuel cycle R&D facility.

5.30. The design of a nuclear fuel cycle R&D facility should provide measures for continuous monitoring and control of the stack exhaust and for the periodic monitoring of the environment around the facility (see also Requirement 25 and paras 6.100–6.104 of SSR-4 [1]).

Protection of personnel against internal exposure

5.31. In a nuclear fuel cycle R&D facility, the static barriers (see para. 5.12) normally protect personnel from internal exposure and external exposure. The design of such barriers should be specified to ensure their integrity and effectiveness and, where appropriate, to facilitate maintenance. The design specifications of such barriers should include the following, as appropriate: weld specifications; 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 closed systems, leaktightness should be used to achieve a high standard of confinement.

5.32. For fume hoods, 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 might escape the first confinement barrier and be inhaled by workers.

5.33. The design of a nuclear fuel cycle R&D facility is required to include equipment to monitor airborne radioactive material (see para. 6.120 of SSR-4 [1]). This equipment should provide an immediate alarm on detection of airborne contamination above a specified threshold. The system design and the location of monitoring points should be chosen with account taken of the following:

(a) The most likely locations of personnel;
(b) Airflows and air movement within the facility;
(c) Evacuation zoning and evacuation routes;
(d) The use of mobile monitoring equipment for temporarily controlled areas (e.g. for maintenance).

5.34. Where radioactive powders or liquids are handled in the nuclear fuel cycle R&D facility or experiment, the installation of collection equipment (e.g. drip trays) should be considered to prevent the accidental spreading of radioactive material or hazardous material and for geometry control.

Protection of personnel against external exposure

5.35. The aims of protection against external radiation exposure are to ensure that exposures are below the dose limits established in schedule III of GSR Part 3 [20], and to optimize protection and safety (see paras 2.7 and 6.6 of SSR-4 [1]), through a combination of source reduction, distance, shielding and administrative controls. Provision of shielding should also be considered in storage areas for radioactive material and waste. The application of the requirement for optimization of occupational exposure should also consider maintenance personnel.

5.36. In areas containing high levels of beta and/or gamma activity (e.g. areas where spent fuel is handled), the protection of personnel 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. In areas containing lower levels of beta/gamma activity (e.g. a teaching laboratory), a combination of shielding and administrative controls should be used for protection of persons (i.e. from exposure to the whole body and to extremities). In general, shielding should be installed as close to the source as possible.

5.37. The potential for exposure 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 (e.g. stainless steel) or should be covered or coated to prevent the accumulation of deposits of processed materials or their decay products. The installation of local shielding (or provisions to add shielding easily) should be considered in locations where radionuclides might accumulate.

Prevention of criticality at a nuclear fuel cycle R&D facility

5.38. Prevention of criticality is an important topic, with various aspects to be considered during the design and operation of a nuclear fuel cycle R&D facility. Requirement 38 of SSR-4 [1] states: "**The design shall ensure an adequate margin of subcriticality, under operational states and conditions that are referred to as credible abnormal conditions, or conditions included in the design basis.**" Detailed recommendations on criticality safety are provided in SSG-27 (Rev. 1) [3].

5.39. The criticality safety analysis should demonstrate that the design of equipment and the related safety measures are such that the facility is in a subcritical state at all times (i.e. the values of the controlled parameters are always maintained in the subcritical range). Safety margins should be derived and applied in accordance with paras 2.8–2.12 of SSG-27 (Rev. 1) [3].

5.40. Paragraph 6.142 of SSR-4 [1] states that "For the prevention of criticality by means of design, the double contingency principle shall be the preferred approach."

5.41. Any system interfaces at which there is a change in the state of the fissile material or in the method of criticality control are required to be specifically assessed (see para. 6.147 of SSR-4 [1]). Particular care should also be taken to assess all transitional, intermediate or temporary states that occur, or could reasonably be expected to occur, under all operational states and accident conditions.

5.42. In many nuclear fuel cycle R&D facilities in which fissile material is handled, 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 restricting the mass in an area should be preferred (i.e. compared with other parameters, as listed in para. 5.43(b)–(j)). Several such areas may coexist independently in a single facility, provided there are suitable controls.

5.43. For Case 2 nuclear fuel cycle R&D facilities, the recommendations on the control of criticality provided in relevant facility specific Safety Guides (i.e. IAEA Safety Standards Series Nos SSG-5 (Rev. 1), Safety of Conversion Facilities and Uranium Enrichment Facilities [21]; SSG-6 (Rev. 1) [5]; SSG-7 (Rev. 1), Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities [22]; and SSG-42 (Rev. 1), Safety of Nuclear Fuel Reprocessing Facilities [23]) should be implemented. In any case, the recommendations for the prevention of criticality in SSG-27 (Rev. 1) [3] should be followed. Examples of the parameters that should be controlled in nuclear fuel cycle R&D facilities to prevent criticality include the following:

(a) Mass: The mass margin[5] should be based on a representative material with the lowest critical mass. The mass margin should not be less than 100% of the normal value in operation (unless the likelihood of double batching is demonstrated to be sufficiently low) or equal to the physical mass that can be accumulated.
(b) Geometry or shape: The safety analysis should give consideration to the layout of the facility and the dimensions and locations of pipes, vessels and other laboratory equipment. Geometry control could be used, for example, in the design of furnaces and dissolvers.
(c) Density and forms of materials: The safety analysis should consider the range of densities for different forms of materials (e.g. powder, pellets, rods) used in a nuclear fuel cycle R&D facility.
(d) Concentration and density of material in analytical laboratories and in liquid effluent units: The safety analysis should consider the range of fissile material in solution as well as any potential precipitates (e.g. plutonium recovered from waste streams).
(e) Moderation: The safety 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 nuclear fuel cycle R&D facilities or that might be present under accident conditions (e.g. water from firefighting). The possibility of non-homogeneous distributions of moderators with fissile material should be considered (e.g. organic binders and porosity enhancing agents used in the pelletizing process).

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

(f) Moisture content in powders: The safety analysis should consider the range of moisture content for the powders used in a nuclear fuel cycle R&D facility.
(g) Reflection: The most conservative margin resulting from different assumptions should be retained; example assumptions include the following:
    (i) A hypothetical thickness of water around a processing unit;
    (ii) Consideration of the actual neutron reflection effect due to, for example, the presence of personnel, organic materials, shielding materials, or the 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. Neutron absorbers such as cadmium and boron may be used in nuclear fuel cycle R&D facilities, and the safety analysis should address their effect as neutron absorbers; however, ignoring their effects would still yield conservative results. The use of mobile or easily displaced or removed solid absorbers should be avoided. Care should be taken where there is the possibility of precipitation or dilution of soluble absorbers in fissile solutions.
(i) Neutron interaction: Consideration should be given to neutron interactions between fissile material in all locations in the nuclear fuel cycle R&D facility and all potential locations that might be involved. Specific consideration should be given to the layout of the nuclear fuel cycle 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. Physical locators should be designed to remain in place under natural phenomena such as earthquakes, tsunamis, flooding and tornadoes.
(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.

5.44. For a process where fissile material is handled in a discontinuous manner (including batch processing or waste packaging), the process and its equipment are required to meet the requirements established in Requirement 66 and paras 9.83–9.85 of SSR-4 [1] for criticality control. The design of the nuclear fuel cycle R&D facility, including any support systems, should provide the necessary equipment for accounting for and control of nuclear material and should have clear and easily identifiable boundaries. Particular consideration is required to be given to the interface between two areas to ensure that transfers of fissile material meet criticality control requirements for both areas (see para. 6.147 of SSR-4 [1]). 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.

5.45. Requirements for criticality detection and alarm systems and associated provisions are established in paras 6.149, 6.172 and 6.173 of SSR-4 [1]. Information regarding the need to install criticality accident alarm systems can be found in Ref. [24]. Where such systems are installed, the nuclear fuel cycle R&D facility design is required to include clearly marked evacuation routes and personnel regrouping areas (see para. 6.149 of SSR-4 [1]). Personnel should be trained in criticality evacuation procedures.

5.46. The areas in a nuclear fuel cycle R&D facility containing fissile material for which criticality detection and alarm systems are necessary to initiate immediate evacuation[6] should be defined in accordance with the layout of the facility, the process being undertaken in the area, the criticality safety analysis and regulatory requirements.

5.47. The need for additional shielding, remote operation and other design measures to mitigate the consequences of a criticality accident, if one should occur, should be assessed in terms of the application of the concept of defence in depth, as described in paras 6.19–6.27 of SSR-4 [1]. For example, consideration should be given to the provision of remote mitigation devices to empty a vessel containing the solution initiating the event or to absorb the neutron flux.

POSTULATED INITIATING EVENTS FOR A NUCLEAR FUEL CYCLE R&D FACILITY

5.48. In accordance with Requirement 19 and paras 6.1 and 6.60–6.76 of SSR-4 [1], postulated initiating events from the list of internal hazards and external hazards for a nuclear fuel cycle R&D facility, and credible combinations thereof, are required to be identified for detailed further analysis.

[6] The immediate activation of the alarm system is to minimize doses to personnel in case of repeated or multiple criticality events or events with slow criticality kinetics.

Internal hazards at a nuclear fuel cycle R&D facility

5.49. The design of a nuclear fuel cycle R&D facility is required to take into account the nature and severity of internal hazards (see Requirement 15, paras 6.43–6.48 and the appendix to SSR-4 [1]).

Internal fires and explosions

5.50. Requirements for fire safety at a nuclear fuel cycle facility are established in Requirement 41 and paras 6.162–6.167 of SSR-4 [1].

5.51. In a nuclear fuel cycle R&D facility, fire hazards are 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.

5.52. Fire in a nuclear fuel cycle R&D facility might 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 or moderation or by modifying the control system.

5.53. An analysis of fire and explosion hazards in a nuclear fuel cycle R&D facility is required to be conducted (see Requirement 22 and paras 6.77–6.79 of SSR-4 [1]). Fire hazard analysis should identify potential causes of fires, such as any fuels or oxidizing agents present, sources of open fire and heat, or electrical cables. The potential consequences of fires should be assessed, where appropriate, with an estimation of the frequency or probability of the occurrence of the consequences. The analysis should also assess the inventory of radioactive materials, ignition sources and combustible materials nearby and should determine the adequacy of measures for fire protection. Computer modelling of fires may be used in support of the fire hazard analysis. 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 is low, a potential fire might have significant consequences with regard to safety and, as such, certain protective measures are likely to be necessary.

5.54. An important aspect of the fire hazard analysis for a nuclear fuel cycle R&D facility is the identification of areas of the facility that warrant special consideration (see Requirement 22 of SSR-4 [1]). In particular, the fire hazard analysis should consider the following:

(a) Areas where radioactive material is processed and stored;
(b) Areas in which radioactive and/or other hazardous powders are produced or processed;
(c) Workshops, laboratories and storage areas containing flammable and/or combustible liquids, solvents, resins and reactive chemicals, or involving the mechanical treatment of pyrophoric metals or alloys (e.g. cuttings, shavings);
(d) Areas with high fire loads, such as waste storage areas;
(e) Waste treatment areas, especially if incineration is used;
(f) Rooms containing items important to safety (e.g. rooms containing the last stage filters of the ventilation system, electrical switch rooms) whose failure might lead to radiological consequences or consequences that are unacceptable in terms of criticality safety;
(g) Process control rooms and supplementary control rooms, where appropriate;
(h) Evacuation routes.

5.55. Paragraph 6.162 of SSR-4 [1] states:

> "The design shall include provisions to:
>
> (a) Prevent fires and explosions;
> (b) Detect and quickly extinguish those fires that do start, thus limiting the damage caused;
> (c) Prevent the spread of those fires that are not extinguished, and prevent fire induced explosions, thus minimizing their effects on the safety of the facility."

5.56. Requirements for measures to accomplish the dual aims of fire prevention and mitigation of the consequences of a fire are established in paras 6.162–6.167 and 9.109–9.115 of SSR-4 [1]. For a nuclear fuel cycle R&D facility, these measures include the following:

(a) Minimization of the combustible load of individual areas, including fume hoods, gloveboxes and hot cells.
(b) Segregation of the areas where non-radioactive hazardous material from process areas is stored.

(c) Use of inert atmospheres with oxygen monitoring alarms in gloveboxes and hot cells in which there is a high likelihood of fire (e.g. from cutting metal clad fuel elements).
(d) Selection of materials in accordance with their functional requirements and fire resistance ratings.
(e) Compartmentation of buildings and ventilation ducts as far as possible to prevent the spread of fires. The higher the fire risk, the greater the number of fire compartments that 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. Attention should be paid to the potential spread of contamination due to fire degrading the boundaries of the compartments. This is of particular importance when the compartment boundary is also the last barrier of confinement. Thus, the preferable option is to have separate boundaries for the confinement function and for the fire compartment.
(f) Minimization of the number of possible ignition sources such as open flames or electrical sparks, and their segregation from combustible material to the extent practicable.
(g) Insulation of hot or heated surfaces.
(h) Installation of fire detection systems inside rooms where radioactive material is handled. Provision of detectors inside cells, gloveboxes and ventilation ducts should also be considered.
(i) Selection of suitable fire extinguishing media consistent with the findings of other safety analyses, especially with the requirements for criticality control (see Requirement 38 and para. 6.146(c) of SSR-4 [1]).
(j) Avoidance of the possible spread of contamination due to dynamic containment acting in reverse or due to uncontrolled water flows where extinguishing devices are installed inside fume hoods, gloveboxes or cells.
(k) Consideration of the potential for operator asphyxiation and of the integrity of the gas supply, where inert gas is used as a fire suppressant.

5.57. The design of ventilation systems in a nuclear fuel cycle R&D facility should be given particular attention with regard to fire prevention. Dynamic containment comprises ventilation ducts and filter units, which might 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. The 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.

5.58. Suitable monitoring equipment for the ventilation system in a nuclear fuel cycle R&D facility 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 therefore be given to the need to provide dampers and other design means to reduce the challenge to filters in the event of a fire.

5.59. Requirements relating to the prevention of explosions at a nuclear fuel cycle R&D facility are established in Requirements 22 and 41 and paras 6.77–6.79 and 6.162–6.167 of SSR-4 [1]. Explosions caused by explosive chemicals can cause a release of radioactive material. The potential for explosion can result from the use of extraction solvents, hydrogen, hydrogen peroxide, nitric acid, degradation products and pyrophoric materials (e.g. metallic hydrides, dust or particles).

5.60. To prevent a release of radioactive material as a result of an internal explosion, the following provisions should be considered in the design of a nuclear fuel cycle R&D facility:

(a) The need to maintain the separation of incompatible chemical materials in normal operation and anticipated operational occurrences (e.g. recovery of leaks);
(b) The use of blow-out panels to mitigate the effects of explosions;
(c) The control of parameters (e.g. concentration, temperature, pressure, flow rate) to prevent conditions that might lead to explosion;
(d) The use of inert atmospheres;
(e) Control of humidity levels;
(f) Effective airlocks between areas containing flammable atmospheres and other areas;
(g) The use of ventilation systems (e.g. to prevent the accumulation of combustible gases, to prevent temperature rises).

Handling errors

5.61. Requirements relating to handling of fissile material and other radioactive material are established in Requirement 51 and paras 6.192–6.195 of SSR-4 [1]. Mechanical or electrical failures or human errors in the handling of such materials might result in the degradation of criticality controls, confinement or shielding or a reduction in defence in depth. The following should be achieved in the design of a nuclear fuel cycle R&D facility:

(a) Elimination of the need to lift loads, where practicable, especially within the facility, by using track-guided transport or another stable means of transport;

(b) Limitation of the consequences of drops and collisions (e.g. by minimizing the heights of lifts (see para. 6.194 of SSR-4 [1]), qualifying containers against the maximum drop, designing floors to withstand the impact of dropped loads and installing shock absorbing features, and specifying safe travel paths);
(c) Minimization of the failure frequency of mechanical handling systems (e.g. cranes, carts) by appropriate design, including through control systems with multiple fail-safe features (e.g. brakes, wire ropes, action on power loss, interlocks).

These measures should be supported by ergonomic design (see para. 6.11 of SSR-4 [1]), human factors analysis (see Requirement 27 of SSR-4 [1]) and appropriate administrative controls (see paras 9.36 and 9.37 of SSR-4 [1]).

Equipment failures

5.62. Paragraphs 6.80–6.89 of SSR-4 [1] establish requirements to address equipment failure in the design of a nuclear fuel cycle R&D facility. Thus, a nuclear fuel cycle R&D facility is required to be designed to cope with the failure of equipment that would result in a degradation of confinement, shielding or criticality control or a reduction in defence in depth. As part of the design, the failure of all SSCs important to safety is required to be assessed (see paras 6.1 and 6.80 of SSR-4 [1]), and consideration is required to be given (in accordance with the results of safety assessment) to the design or procurement of items that fail to a safe configuration. Where no safe configuration can be assured, the functionality of SSCs important to safety is required to be maintained (see para. 6.89 of SSR-4 [1]); for example, through diversity, redundancy, physical separation and/or independence, as necessary.

5.63. Failure due to fatigue, corrosion or lack of mechanical strength should be considered in the design of containment systems for a nuclear fuel cycle R&D facility.

5.64. To prevent failure of equipment containing hazardous materials, effective programmes for maintenance, periodic testing and inspection should be established at the design stage of a nuclear fuel cycle R&D facility (see also paras 5.149–5.151).

5.65. In evaluating failure and fail-safe conditions, special consideration should be given to the failure of computer systems, computerized control, and software systems, through the application of appropriate national or international codes and standards or through a functional analysis of the systems and their failure frequencies (see also Requirement 45 of SSR-4 [1]).

Loss of services

5.66. A nuclear fuel cycle R&D facility should be designed to cope with loss of services that might have an impact on safety. The loss of services should be considered both for individual items of equipment and for the facility as a whole and, on multifacility sites, for the nuclear fuel cycle R&D facility's ancillary and support facilities (e.g. waste treatment and storage facilities and other facilities on the site). Requirements for electrical power supply systems and compressed air systems are established in Requirements 49 and 50 of SSR-4 [1].

5.67. To meet the requirements established in Requirements 49 and 50 and para. 6.89 of SSR-4 [1], electric power supplies and other support services in a nuclear fuel cycle R&D facility should be of high reliability[7]. In the event of a loss of normal power, and depending on the status of the facility, an emergency power supply is required to be provided to certain SSCs important to safety (see para. 6.187 of SSR-4 [1]). For a nuclear fuel cycle R&D facility, these SSCs include the following:

(a) Criticality detection and alarm systems;
(b) Ventilation fans and monitoring systems for the confinement of radioactive material;
(c) Heat removal systems;
(d) Emergency control systems;
(e) Fire detection and alarm systems;
(f) Radiation monitoring systems;
(g) Instrumentation and control associated with the above items;
(h) Adequate lighting (see also para. 6.182 of SSR-4 [1]).

5.68. The loss of services such as compressed air, water for process equipment and ventilation systems, heating, and breathing air might also have consequences

---

[7] Contributions to reliability include the use of diverse and redundant electric power sources, switching and connections; the design of power supplies to withstand external hazards; and the use of uninterruptible power sources when necessary.

for safety. Examples of suitable measures that should be addressed in the design of a nuclear fuel cycle R&D facility include the following:

(a) In accordance with the safety assessment, the design of supply systems[8] should be of adequate reliability, with diversity and redundancy, as necessary.
(b) The maximum period that a loss of support supplies can be sustained with acceptable levels of safety should be assessed and considered in the design provisions for all such supplies.
(c) As far as practicable, pneumatically actuated valves should be designed to be fail-safe in the event of a loss of air supply, in accordance with the safety analysis.
(d) Adequate backup capacity or a redundant supply should be provided to cope with a loss of water or heating.
(e) With regard to a 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.

5.69. Consideration should be given to the possible loss, lack or excess of process media or additives that might have safety consequences. Examples include the following:

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

5.70. Consideration should be given to processes that generate heat and to ventilation systems that need 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 Case 2 facility can have significant heat loads and might need to 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 Case 2 facilities with large heat sources.

5.71. Other functions of the ventilation system should be considered in the safety analysis, such as the maintenance of cooling to prevent heat stress to operating

---

[8] Examples of supply systems include air reservoirs, uninterruptible power supplies and diverse cooling.

personnel or the control of humidity where materials are handled. These can have an indirect effect on the safety of operations.

Leaks and spills

5.72. Requirement 35 and para. 6.120 of SSR-4 [1] establish requirements for confinement and leak detection for radioactive material. At a nuclear fuel cycle R&D facility, leaks from equipment and components such as pumps, valves and pipes might lead to the dispersion of radioactive material, fissile material or toxic chemicals and to the creation of unnecessary waste. Leaks of hydrogenous fluids (e.g. water, oil) can change the neutron moderation of fissile material and reduce the criticality safety margin. Leaks of flammable gases (e.g. hydrogen, natural gas, propane) or liquids might lead to explosions and/or fire. Leak detection systems should be used if such fluids are present.

5.73. Vessels containing significant quantities of fissile material in liquid form should be equipped with alarms and interlocks to prevent overfilling and subsequent overflow or spillage. The area beneath the vessels should include means to ensure that spilled fissile material will be safely contained, for example, with drip trays configured to ensure criticality safety and of a capacity that can safely accommodate the volume of the vessel. The subcriticality of collected leaks and spills is required to be demonstrated (see para. 6.146(a) of SSR-4 [1]).

5.74. Leakage of coolants where there might be physical or chemical incompatibility with the materials or equipment present should be considered. The possibility of an unintended chemical reaction causing the precipitation of fissile material should also be considered (see also para. 6.139(c) of SSR-4 [1]).

5.75. Spillage might occur from cans, drums or waste packages during transit within the nuclear fuel cycle R&D facility and in storage areas. Appropriate measures to ensure containment during material movements should be provided.

Flooding

5.76. Requirements relating to protection against internal flooding of a nuclear fuel cycle R&D facility are established in Requirement 15 of SSR-4 [1]. Flooding might lead to dispersion of radioactive material and/or changes in the moderation of any fissile material present. Rainwater, groundwater, condensates, and heating and cooling fluids are all capable of flooding a facility. Flooding, and even dew, might cause harm to equipment, including electrical damage and corrosion, and could infiltrate emergency supplies or fissile material.

5.77. For areas where fissile material is present, the criticality safety analysis should consider the risk and consequences of condensation and flooding. Full disconnection from the water supply or the use of limited water volumes should be considered. Equipment should not have water supply connections during normal operation unless the criticality assessment has taken into account the presence and potential leakage of water.

5.78. In nuclear fuel cycle 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 analysis should consider the presence of the maximum credible amount of liquid within each room, as well as the maximum credible amount of liquid that could flow from any connected rooms, vessels or pipework.

5.79. The potential hydraulic pressure and upthrust on large vessels, ducting and containment structures in the event of flooding should be considered in the design of a nuclear fuel cycle R&D facility.

Chemical hazards

5.80. Requirements for the management of chemical hazards in a nuclear fuel cycle facility are established in Requirement 42 and para. 6.168 of SSR-4 [1]. In a nuclear fuel cycle R&D facility, 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 generation of gas, especially hydrogen. These effects should be taken into account in the safety analysis for the following:

(a) Liquid effluents and organic solvents used in the facility;
(b) Contaminated oil and flammable waste;
(c) Process scraps containing hydrogenated additives.

The design of a nuclear fuel cycle R&D facility should prevent or mitigate the effects of hazards associated with radiolysis and irradiation.

External hazards at a nuclear fuel cycle R&D facility

5.81. The design of a nuclear fuel cycle R&D facility is required to take into account the nature and severity of external hazards (see Requirement 16 and paras 6.49–6.54 of SSR-4 [1]). Such external hazards, either natural or human induced, are required to be identified and evaluated in accordance with the requirements established in SSR-1 [16]. Detailed recommendations on the protection of nuclear installations against external hazards are provided in IAEA Safety Standards Series Nos SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [25]; SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [26]; SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations [27]; SSG-67, Seismic Design for Nuclear Installations [28]; SSG-68, Design of Nuclear Installations Against External Events Excluding Earthquakes [29]; and SSG-79, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations [30].

Earthquakes

5.82. To ensure that the design of the nuclear fuel cycle R&D facility provides the necessary degree of robustness, a seismic assessment is required to be performed (see Requirements 15 and 16 of SSR-1 [16]) using a graded approach. Recommendations on this assessment are provided in SSG-9 (Rev. 1) [25] and SSG-67 [28]. The assessment of seismic hazards for a nuclear fuel cycle R&D facility design should include the following seismically induced events, as applicable:

(a) Loss of cooling.
(b) Loss of support services, including utilities.
(c) Loss of confinement (static and dynamic).
(d) Loss of safety functions for ensuring the return of the facility to a safe state and maintaining the facility in a safe state after an earthquake, including structural functions and functions for the prevention of other hazards (e.g. fire, explosion, load drop, flooding).
(e) The effect of the following on criticality safety functions such as geometry, moderation, absorption and reflection:
    (i) Deformation (geometry control);
    (ii) Displacement (geometry control, fixed poisons);
    (iii) Loss of material (geometry control, soluble poisons);
    (iv) Ingress of moderating material (moderation control);
    (v) Accumulation of fissile material;
    (vi) Homogeneous or heterogeneous mixing of fissile material with a moderator.

5.83. In accordance with Requirement 14 and paras 6.49 and 6.50 of SSR-4 [1], a nuclear fuel cycle R&D facility is required to be designed to withstand the design basis earthquake. The design should also be evaluated for beyond design basis seismic events considered as design extension conditions (see para. 6.73 of SSR-4 [1]) to ensure that such an event will not impair the function of control rooms (where provided) and will not cause loss of confinement or a criticality accident and that there is an adequate seismic margin to avoid cliff edge effects.

External fires and explosions and external toxic hazards

5.84. Hazards from external fires and explosions could arise from various sources in the vicinity of a nuclear fuel cycle R&D facility, such as petrochemical installations; combustible vegetation; pipelines and road, rail or sea routes used for the transport of flammable material (e.g. gas, oil); and volcanoes.

5.85. The hazards associated with external fires and explosions and external toxic hazards are required to be evaluated (see para. 5.33 of SSR-1 [16]). To demonstrate that the risks associated with such external hazards are below acceptable levels, the operating organization should first identify all potential sources of hazard and then estimate the associated event sequences that might affect the safety of the nuclear fuel cycle R&D facility. The radiological consequences of any damage should be assessed, and it should be verified that they are within acceptance criteria.

5.86. The operating organization is required to consider potentially hazardous installations and transport operations for hazardous material in the vicinity of the nuclear fuel cycle R&D facility (see paras 5.36 and 5.37 of SSR-1 [16]). Toxic and asphyxiant hazards should be evaluated to verify that specific gas concentrations meet the acceptance criteria. It should be ensured that external toxic and asphyxiant hazards would not adversely affect the control of the facility. In the case of explosions, risks should be assessed for compliance with overpressure criteria. To evaluate the possible effects of flammable liquids, volcanic ash, falling objects (e.g. chimneys), and air shock waves and missiles resulting from explosions, the possible distance of these hazards from the facility, and hence their potential for causing physical damage, should be assessed.

Extreme meteorological phenomena

5.87. A nuclear fuel cycle R&D facility is required to be protected against extreme meteorological conditions as identified in the site evaluation (see Section 4) by means of appropriate design provisions (see para. 5.7(b) of SSR-4 [1] and Requirement 18 of SSR-1 [16]). These provisions should address the events consequential to extreme meteorological conditions and generally include the following:

(a) The ability of structures important to safety to withstand extreme weather loads;
(b) The prevention of flooding of the facility, including adequate means to remove water from the roof in cases of extreme rainfall;
(c) The ability to safely shut down experiments in the facility in accordance with the operational limits and conditions, followed by maintaining the facility in a safe and stable shutdown state, where necessary;
(d) Means of ensuring that high water levels during floods do not jeopardize the integrity and functionality of SSCs important to safety.

Tornadoes

5.88. Measures for the protection of a nuclear fuel cycle R&D facility against tornadoes will depend on the meteorological conditions for the area where the facility is located. The design of buildings and ventilation systems should comply with specific national regulations relating to hazards from tornadoes. If such regulations do not exist, the design should adhere to international good practices.

5.89. High winds are capable of lifting and propelling large, heavy objects (e.g. automobiles, telegraph poles). The possibility of impacts of such missiles is required to be taken into consideration during the design stage for the facility (see para. 5.14 of SSR-1 [16]). This should include consideration of both the initial impact and the effects of secondary fragments arising from collisions with concrete walls or from other forms of transfer of momentum.

Extreme temperatures

5.90. Extreme low or high temperatures, and their potential duration, are required to be taken into account in the design of the facility (see para. 5.11 of SSR-1 [16]). For a nuclear fuel cycle R&D facility, the aim should be to prevent effects such as the following:

(a) The freezing of cooling circuits (including cooling towers and outdoor actuators);
(b) The loss of efficiency of cooling circuits (i.e. during hot weather);
(c) Adverse effects on a building’s ventilation, heating and cooling systems that could cause poor working conditions and excess humidity in the buildings and adverse effects on SSCs important to safety.

Administrative controls to limit or mitigate the consequences of extreme temperatures should be relied on only if operating personnel have the necessary information and equipment (e.g. portable air-conditioning) and sufficient time to implement the measures.

5.91. If limits for humidity and/or temperature are specified in a building or a compartment, the air-conditioning system should be designed to also meet these limits during extreme weather conditions. Structural components of buildings, such as static containment, should also be designed to withstand extreme temperature and humidity and associated thermal stress effects, such as shrinkage in concrete.

Snowfall and ice storms

5.92. The occurrence of snowfall and ice storms and their effects are required to be taken into account in the design and the safety analysis for a nuclear fuel cycle R&D facility (see paras 5.11 and 5.27 of SSR-1 [16]). Snow and ice are generally taken into account as an additional load on the roofs of buildings. Icing in outdoor switchyards might lead to short circuits and thus a loss of off-site power. Snow can also block the inlets of ventilation systems and the outlets of drains. The flooding resulting from snow or ice accumulation and infiltration as well as the possibility that it could damage equipment important to safety (e.g. electrical systems) should be considered. The neutron reflecting effect and 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.

Flooding

5.93. A nuclear fuel cycle R&D facility is required to be protected against flooding (see para. 5.7(c) of SSR-4 [1] and Requirement 20 of SSR-1 [16]). For all potential flood events, such as extreme rainfall (for an inland site) or storm surge (for a coastal site), attention should be focused on SSCs important to safety. Equipment containing fissile material is required to be designed to prevent any criticality accident in the event of flooding (see para. 6.146(e) of SSR-4 [1]). Gloveboxes should be designed to be resistant (i.e. remain undamaged and static) to the dynamic effects of flooding, and all glovebox penetrations should be above any design basis flood levels. Electrical systems, instrumentation and control systems, emergency power systems (i.e. batteries and power generation systems) and control rooms should be protected by design.

5.94. With regard to extreme rainfall, attention should be focused on the stability of buildings (e.g. hydrostatic and dynamic effects), the water level and, where relevant, the potential for mudslides. In addition to the results of the flooding hazard assessment performed in accordance with the recommendations provided in SSG-18 [26], consideration should be given to the highest flood level historically recorded and to siting the facility above this flood level, at sufficient elevation and with sufficient margin to take into account uncertainties (e.g. in postulated effects of climate change), to avoid major damage from flooding.

Inundation events

5.95. Measures for the protection of the facility against natural and human induced inundation events (e.g. dam burst, flash flood, storm surge, tidal wave, seiche, tsunami), including static effects (e.g. floods) and dynamic effects (e.g. runup, drawdown), will depend on the data collected during site evaluation for the area in which the nuclear fuel cycle R&D facility is located. The design of buildings, electrical systems, and instrumentation and control systems should comply with specific national regulations for inundation hazards and with the recommendations provided in paras 5.93 and 5.94. Particular attention should be given to the rapid onset of inundation events, the probable lack of warning, and the potential for these events to cause widespread damage, disruption of utility supplies and common cause failures both within the nuclear fuel cycle R&D facility and at other facilities on the site (and potentially locally and regionally, depending on the magnitude of the event).

Accidental aircraft crashes or hazards from externally generated missiles

5.96. In accordance with the risk identified in the site evaluation (see Section 4), a nuclear fuel cycle R&D facility is required to be designed to withstand the design basis impact from accidental aircraft crashes or hazards from externally generated missiles (see para. 5.7(e) of SSR-4 [1] and para. 5.35 of SSR-1 [16]).

5.97. In evaluating the consequences of aircraft or secondary missile impacts on a nuclear fuel cycle R&D facility and the adequacy of the design to resist such impacts, only realistic crash scenarios, rotating equipment scenarios or structural failure scenarios should be considered, in accordance with a graded approach that is commensurate with the hazards associated with the facility. Knowledge of factors such as the possible angle of impact, the possible velocity or the potential for fire and explosion due to the aviation fuel load is needed to develop these scenarios. In general, fire cannot be ruled out following an aircraft crash. Therefore, specific design provisions for fire protection should be implemented, as necessary.

INSTRUMENTATION AND CONTROL SYSTEMS AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.98. Requirement 43 of SSR-4 [1] states:

> "**Instrumentation and control systems shall be provided for monitoring and control of all the process parameters that are necessary for safe operation in all operational states. Instrumentation shall provide for bringing the system to a safe state and for monitoring of accident conditions. The reliability, redundancy and diversity required of instrumentation and control systems shall be proportionate to their safety classification.**"

Therefore, instrumentation is required to be provided for measuring all the main parameters whose variation might affect the safety of processes at a nuclear fuel cycle R&D facility. As stated above, monitoring and control systems are required to cover normal operation, anticipated operational occurrences and accident conditions to ensure that adequate information can be obtained on the status of the operations and the facility and that proper actions can be undertaken in accordance with operating procedures, emergency procedures or accident management guidelines, as appropriate, for all facility states.

5.99. Instrumentation and control systems are required to be provided for criticality control and for hot cells, gloveboxes and hoods to fulfil the requirements for static and dynamic confinement (see paras 6.172–6.174 of SSR-4 [1]).

5.100. Passive and active engineering controls are more reliable than administrative controls and should be preferred for control in operational states and in accident conditions. Automatic systems are required to be designed to maintain process parameters in a nuclear fuel cycle R&D facility (or within individual experimental apparatus) within the operational limits and conditions or to bring the process to a predetermined safe state (see paras 6.21(d), 6.109 and 6.169 of SSR-4 [1]).

5.101. Appropriate information is required to be made available to operating personnel for monitoring the effects of automatic actions (see para. 6.170 of SSR-4 [1]). The layout of instrumentation and the manner of presentation of information should provide operating personnel with an adequate picture of the status and performance of the facility. Where necessary, devices should be installed that provide in an efficient manner visual and, as appropriate, audible indications of deviations from normal operation that could affect safety.

5.102. Control systems should be provided to ensure compliance with regulatory limits, for example, on discharges. Where appropriate, provision should be made for the automated measurement and recording of parameters that are important to safety, and manual periodic testing should be used to complement automated continuous testing of conditions.

Safety related instrumentation and control systems at a nuclear fuel cycle R&D facility

5.103. Safety related instrumentation and control at a nuclear fuel cycle R&D facility include systems for the following, as determined by the application of a graded approach:

(a) Criticality control, criticality detection and alarm:
   (i) Depending on the method of criticality control, the monitoring and control parameters include mass, concentration, acidity (which might have an impact on solubility, extraction, stripping or precipitation), isotopic composition or fissile content, burnup, and quantity of reflectors and moderators, as appropriate.
(b) Fire detection and extinguishing systems (see Requirement 41 of SSR-4 [1]):
   (i) All rooms with fire loads or significant amounts of fissile material and/or toxic chemicals should be equipped with provisions for fire detection and fire extinguishing.
   (ii) Gas detectors should be used in areas where a leakage of gas (e.g. hydrogen) could produce an explosive atmosphere.
(c) Process control and monitoring and control of equipment and supplies:
   (i) 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.
   (ii) A means of confirming correct concentrations of reactive media in supplies to hot equipment should be provided.
(d) Glovebox control and cell control:
   (i) For gloveboxes and cells under inert atmosphere, the gas concentration should be monitored and controlled for safety and possibly for product quality purposes.
   (ii) Temperatures should be monitored.
   (iii) Instrumentation and controls for ensuring negative pressure and fire control should be installed.

(e) Control of occupational radiation exposure, including provision of:
   (i) Electronic dosimeters with real time displays and/or alarms to monitor occupational exposure, including in areas with inspection equipment using X rays and sealed radiation sources.
   (ii) Installed (area) dose rate monitors for gamma and neutron radiation.
   (iii) Continuous air monitors to detect airborne radioactive material installed as close as possible to working areas.
   (iv) Devices for detecting surface contamination, installed or located close to relevant working areas and close to the exits from these areas.
(f) Control of liquid discharges and gaseous effluents, including:
   (i) Systems to monitor and control liquid discharges. This can be done by sampling and analysis and by measuring the volume of discharge.
   (ii) Systems to monitor and control gaseous discharges. This can be done by measurements of, for example, differential pressure to confirm that the filtration systems are working effectively and by continuous monitoring of discharges.
(g) Monitoring and control of airflows and air quality, including:
   (i) Systems to ensure that air in all areas of the nuclear fuel cycle R&D facility moves in the correct direction (i.e. from less contaminated to more contaminated areas).
   (ii) In work areas, the temperature, humidity and pollutants should be controlled to ensure comfort and hygiene.
   (iii) In some cases, local ventilation should be used; for example, in rooms housing backup batteries.

Control rooms at a nuclear fuel cycle R&D facility

5.104. Requirements for the design of control rooms for nuclear fuel cycle facilities are established in Requirement 46 and para. 6.180 of SSR-4 [1]. In a Case 2 nuclear fuel cycle R&D facility, control rooms should be provided to centralize the main data displays, controls and alarms for general conditions at the facility. For specific experiments in a Case 1 facility, it may be useful to have local control panels where relevant information can be gathered and monitored. Controls should be located in parts of the facility where risks to operating personnel can be minimized. Particular consideration should be given to identifying events, both internal and external to the control room, that might pose a direct threat to control room operators and to the operation of control rooms. Ergonomic principles are required to be applied in the design of control rooms and their displays and systems (see para. 6.108 of SSR-4 [1]).

HUMAN FACTORS ENGINEERING AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.105. Requirements relating to consideration of human factors are established in Requirement 27 and paras 6.107–6.110 of SSR-4 [1].

5.106. In accordance with Requirement 27 of SSR-4 [1], human factors in operation, inspection, periodic testing and maintenance are required to be considered at the design stage. Human factors that should be considered for nuclear fuel cycle R&D facilities include the following:

(a) The ease of intervention by operating personnel in all facility states;
(b) Possible effects on safety of inappropriate or unauthorized human actions (with account taken of tolerance of human error);
(c) The potential for occupational exposure.

5.107. In the design of a nuclear fuel cycle R&D facility, work locations should be evaluated for all modes of operation of the facility, including maintenance. The circumstances in which human intervention is necessary under abnormal conditions or accident conditions should be identified. The aim should be to facilitate the necessary actions of operating personnel and ensure that safety functions, and the SSCs that support them, are resistant to human error during such actions. This should include optimization of the design to prevent or reduce the likelihood of operator error (e.g. locked valves, segregation and grouping of controls, fault identification, logical displays, and segregation of displays and alarms for processes and safety systems). Particular attention should be paid to situations in which, in accident conditions, operating personnel need to make a rapid, accurate, fault tolerant identification of the problem and select an appropriate response or action.

5.108. Experts in human factors engineering and experienced operating personnel should be involved from the earliest stages of design. Areas that should be considered in the design of a nuclear fuel cycle R&D facility include the following:

(a) Application of ergonomic principles to the design of the workplace, considering the following aspects:
    (i) Design of human–machine interfaces (e.g. well laid out electronic control panels displaying all the necessary information);

(ii) The working environment (e.g. good accessibility and spacing of equipment, good lighting (including emergency lighting), surface finishes that allow areas to easily be kept clean);
(iii) Safety features of commercial equipment that has been adapted for nuclear use (e.g. in a glovebox).

(b) The location and the clear, consistent and unambiguous labelling of equipment and utilities 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 needed.
(d) Optimized layout of facilities and equipment and procedures to ensure ease of maintenance, inspection and testing activities.
(e) Task design and job organization, particularly during maintenance work, when automated control systems may be disabled.
(f) Minimization of the need to use personal protective equipment.
(g) Operational experience feedback relevant to human factors.

5.109. In the design and operation of fume hoods, gloveboxes (see para. 6.108 of SSR-4 [1]) and, where appropriate, hot cells, the following should be taken into account:

(a) In the design of equipment inside gloveboxes, the potential for accidents that might result in injuries to personnel, including internal radiation exposure through cuts in the gloves and/or wounds, and/or the possible failure of confinement.
(b) Ease of physical access to gloveboxes and adequate space and good visibility in the areas in which gloveboxes are located.
(c) The potential for damage to gloves and the provisions for glove change and, where applicable, filter change. Sharp edges and corners on equipment and fittings and associated tools should be avoided to minimize risks of glove damage.
(d) The training of operating personnel on procedures to be followed in operational states and in accident conditions (see para. 9.48 of SSR-4 [1]).

SAFETY ANALYSIS FOR A NUCLEAR FUEL CYCLE R&D FACILITY

5.110. Requirement 14 of GSR Part 4 (Rev. 1) [14] states that "**The performance of a facility or activity in all operational states and, as necessary, in the post-operational phase shall be assessed in the safety analysis.**" The safety analysis for a nuclear fuel cycle R&D facility should cover the various hazards for the whole facility (see Section 2 of this Safety Guide) and all the activities performed within the facility.

5.111. The list of postulated initiating events identified is required to take into account all the internal and external hazards and the resulting event scenarios (see Requirement 19 of SSR-4 [1]). The safety analysis is required to consider all the SSCs important to safety that might be affected by the postulated initiating events identified (see para. 4.20 of GSR Part 4 (Rev. 1) [14]).

5.112. For a nuclear fuel cycle R&D facility, the safety analysis should be performed iteratively with the development of the design, with the following objectives:

(a) Doses to workers and the public during operational states do not exceed dose limits and are as low as reasonably achievable, in accordance with Requirement 9 of SSR-4 [1].
(b) Doses to workers and the public during and following accident conditions remain below acceptable limits and are as low as reasonably achievable, in accordance with Requirement 9 of SSR-4 [1].
(c) Appropriate operational limits and conditions are developed, in accordance with Requirement 57 and paras 9.27–9.37 of SSR-4 [1].

Safety analysis for operational states at a nuclear fuel cycle R&D facility

5.113. For a nuclear fuel cycle R&D facility, a facility specific, enveloping and robust (i.e. conservative) assessment of occupational exposure and public exposure during normal operation and anticipated operational occurrences should be performed on the basis of the following assumptions:

(a) The bounding radiation source term (wherever it is located within the facility);
(b) The maximum annual working time at each workplace for both normal work activities and maintenance;
(c) Conservative assumptions about the efficiency of shielding.

5.114. The design of equipment and the layout of equipment and shielding in a nuclear fuel cycle R&D facility should be based on adequate coordination of process and mechanical designs, the safety assessment, and operating experience from other relevant facilities.

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

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

5.117. The calculation of estimated public dose should include all the exposure pathways associated with the facility, namely external exposure through direct or indirect radiation, and internal exposure through intakes of radioactive material (e.g. received through inhalation or through the food chain as a result of authorized discharges of radioactive material). The dose should be estimated for the representative person(s); detailed recommendations are provided in IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [31].

5.118. This Safety Guide addresses only those chemical hazards associated with a nuclear fuel cycle R&D facility that might give rise to radiological hazards (see para. 2.4 of SSR-4 [1]). Facility specific, realistic, robust (i.e. conservative) estimations of chemical hazards to personnel and releases of hazardous chemicals to the environment should be performed, in accordance with the standards applied in the chemical industry (see Requirement 42 and para. 6.168 of SSR-4 [1]).

Safety analysis for accident conditions at a nuclear fuel cycle R&D facility

5.119. The acceptance criteria associated with the safety analysis for accident conditions are required to be defined in accordance with Requirement 16 of GSR Part 4 (Rev. 1) [14] and with any regulatory requirements.

5.120. To estimate the on-site and off-site consequences of an accident at a nuclear fuel cycle R&D facility, the range of physical processes that could lead to a release of radioactive material to the environment needs to be considered, and bounding cases[9] encompassing the worst consequences should be determined.

---

[9] Bounding cases (also called ‘limiting cases’ or ‘enveloping cases’) are used for the estimation of consequences; see para. 6.62 of SSR-4 [1].

5.121. The main steps in the assessment of the possible radiological or chemical consequences of an accident at a nuclear fuel cycle R&D facility include the following:

(a) Analysis of the current site conditions (e.g. meteorological, geological, hydrogeological) and the conditions expected in the future.
(b) Specification of facility design and facility configurations, with the corresponding operating procedures and administrative controls for operations.
(c) Identification of individuals and population groups (for site personnel and members of the public) who might be affected by radiation risks and/or associated chemical risks arising from the facility.
(d) Identification and analysis, in accordance with reasonable scenarios, of conditions at the facility (including internal and external 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.
(e) Quantification of the consequences for site personnel and the representative person(s) identified in the safety assessment.
(f) Specification of the SSCs important to safety that may be credited to reduce the likelihood and/or to mitigate the consequences of accidents. The SSCs that are credited in the safety assessment are required to be qualified to perform their functions reliably in accident conditions (see Requirement 30 of SSR-4 [1]).
(g) Characterization of the source term (e.g. type of material, radionuclides and activity, mass, release rate, temperature).
(h) Identification and analysis of pathways by which material that is released could be dispersed in the environment.
(i) Identification of exposure pathways for both internal and external exposure.

5.122. The analysis of the 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 might influence facility operations or affect the dispersion of material or the transfer of energy that might be released from the facility. The operating organization is required to develop preparatory measures and guidelines to reduce the risk of accidents and return the facility to a controlled state (see paras 9.118 and 9.119 of SSR-4 [1]).

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

5.124. Further recommendations on the assessment of the potential radiological impact to the public are provided in GSG-10 [31]. Guidelines for assessing the acute and chronic toxic effects of chemicals used in nuclear fuel cycle R&D facilities are provided in Ref. [32]. Information on methods and practices, based on the IAEA safety standards and current international good practice, for performing safety analysis and preparing licensing documentation for nuclear fuel cycle facilities is provided in Ref. [33].

Analysis of design extension conditions

5.125. The safety analysis for a nuclear fuel cycle R&D facility is also required to identify design extension conditions and analyse their progression and consequences (see Requirement 21 and paras 6.73–6.75 of SSR-4 [1]). Paragraph 6.74 of SSR4 [1] states:

> "New facilities shall be designed such that the possibility of conditions arising that could lead to early releases of radioactive material or to large releases of radioactive material is practically eliminated. The design shall be such that, for design extension conditions, off-site protective actions that are limited in terms of times and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such actions. The postulated initiating events that lead to design extension conditions shall also be analysed for their capability to compromise the ability to provide an effective emergency response. Only those protective actions that can be reliably initiated within sufficient time at the location shall be considered available."

5.126. Design extension conditions include events more severe than design basis accidents that originate from extreme events or combinations of events that could cause damage to SSCs important to safety or that could challenge the fulfilment of the main safety functions at the nuclear fuel cycle R&D facility. The list of postulated initiating events provided in the appendix to SSR-4 [1], including combinations of these events, should be used, as should potential events with additional failures. Examples of design extension conditions that are applicable to nuclear fuel cycle R&D facilities are listed in Ref. [34]. Additional safety features or increased capability of safety systems (see para. 6.75 of SSR-4 [1]), identified during the analysis of design extension conditions, should be implemented in existing nuclear fuel cycle R&D facilities where practicable.

5.127. For analysing design extension conditions, best estimate methods with realistic boundary conditions are used. Acceptance criteria for the analysis, consistent with para. 6.74 of SSR-4 [1], should be defined by the operating organization and should be reviewed by the regulatory body.

5.128. The analysis of design extension conditions should also demonstrate that the nuclear fuel cycle R&D facility can be brought to a safe state in which the confinement function and subcriticality can be maintained in the long term.

MANAGEMENT OF RADIOACTIVE WASTE AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.129. Requirements for safety in radioactive waste management are established in GSR Part 5 [2]. Supporting recommendations are provided in IAEA Safety Standards Series Nos GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [35]; GSG-1, Classification of Radioactive Waste [36]; SSG-41, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities [37]; and GSG-16 [12].

5.130. In accordance with Requirement 24 of SSR-4 [1], the generation of radioactive waste from a nuclear fuel cycle R&D facility is required to be kept to the minimum practicable in terms of both activity and volume, by means of appropriate design measures. The following aspects should be considered in the design:

(a) Generation and classification of waste: Requirement 8 of GSR Part 5 [2] establishes general design requirements for radioactive waste generation and control. Requirement 9 of GSR Part 5 [2] establishes requirements for the characterization and classification of waste in terms of total activity, concentrations of relevant radionuclides and other hazards. The operating organization is required to maintain records to ensure proper identification, traceability and accounting for the radioactive waste generated (see para. 3.11 of GSR Part 5 [2]). In a nuclear fuel cycle R&D facility it is important to ensure that criticality is avoided 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.
(b) Handling of waste: In accordance with Requirement 10 of GSR Part 5 [2], appropriate containers are required to be provided for radioactive waste. In addition, measures to minimize the spread of contamination at the point at which waste is generated should be taken. Recommendations on the handling of waste containing fissile material, including on mass control, are provided in SSG-27 (Rev. 1) [3]. Examples of such waste at a nuclear fuel

cycle R&D facility include filters from fume hoods, gloveboxes, hot cells and ventilation systems.

(c) Collection of waste: Design features should be implemented to reduce the risk of damage to waste containers that could potentially lead to a loss of confinement. For the predisposal management of radioactive waste at a nuclear fuel cycle R&D facility, consideration should be given to a central waste management area in which the waste is 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 is required to take account of the type of radioactive waste, its characteristics and associated hazards, even if the storage is intended to be short term (see para. 4.20 of GSR Part 5 [2] and para. 6.95 of SSR-4 [1]). Requirement 11 of GSR Part 5 [2] 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 should be taken to ensure the integrity of the facility and the waste containers, taking into account low probability events, even for short term storage.

(e) Processing of waste: Subsequent processing of the waste outside a nuclear fuel cycle 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 storage or disposal. The techniques and procedures for treatment and conditioning are required to provide waste forms and/or waste packages that meet waste acceptance criteria for storage and disposal; see Requirement 12 of GSR Part 5 [2].

MANAGEMENT OF ATMOSPHERIC AND LIQUID RADIOACTIVE DISCHARGES AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.131. Nuclear fuel cycle facilities are required to be designed so that discharges to the environment are minimized (see para. 6.17 of SSR-4 [1]). If discharges cannot be avoided, the operating organization is required to ensure that authorized limits on such discharges are met in normal operation and in anticipated operational occurrences (see Requirement 25 of SSR-4 [1]).

5.132. The activity of gaseous effluent discharged from a nuclear fuel cycle R&D facility should be reduced by process specific ventilation treatment systems.

These systems should include, where necessary, equipment for reducing the discharges of radioiodine and other radioactive volatile or gaseous species. The final stage of treatment normally consists of dehumidification, spark arrestors and debris guards to protect filters, then filtration by 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. The ventilation treatment system for a specific nuclear fuel cycle R&D facility should be designed in accordance with a graded approach.

5.133. Equipment for monitoring the status and performance of filters at a nuclear fuel cycle R&D facility should be installed, including the following, as necessary:

(a) Differential pressure gauges to identify the need for filter changes;
(b) Activity or gas concentration measurement devices and discharge flow measuring devices with continuous sampling;
(c) Test (aerosol) injection systems and the associated sampling and analysis equipment for testing filter efficiency.

5.134. Liquid effluents to be discharged to the environment from a nuclear fuel cycle R&D facility are required to be monitored, treated and managed as necessary to reduce the discharge of radioactive material and hazardous chemicals (see para. 6.101 of SSR-4 [1]). The use of filters, ion exchange beds or other technology should be considered, where appropriate. Analogous provisions to those in para. 5.133 of this Safety Guide should be made to allow the efficiency of these systems to be monitored.

OTHER DESIGN CONSIDERATIONS FOR A NUCLEAR FUEL CYCLE R&D FACILITY

Fume hoods, gloveboxes and hot cells

5.135. 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 shielding

5.136. The materials handled in some nuclear fuel cycle R&D facilities can generate significant beta, gamma and neutron dose rates depending on the isotopic composition of the material processed. Therefore, consideration should be given at the design stage to the need for shielding for both neutron and gamma radiation.

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

Design for fresh fuel storage

5.138. 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 necessary dimensions while maintaining the necessary stability. Fuels should be clearly identifiable. Necessary provisions for physical protection should be included in the design.

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

(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

5.140. Design for maintenance of a nuclear fuel cycle R&D facility should include the following aspects:

(a) Consideration of whether maintenance can be performed remotely, instead of manually using personal protective equipment.
(b) Measures to maintain criticality safety, such as limiting the introduction of liquids, solvents, plastics and other moderators.
(c) The location of equipment to prevent the spread of contamination during maintenance or replacement (e.g. motors and drives can be located outside gloveboxes).
(d) Design provisions for isolation of the work area from the rest of the areas during maintenance or replacement.
(e) Facilitation of good housekeeping (see Requirement 64 of SSR-4 [1]). Gloveboxes and hot cells can become dusty unless cleaned regularly. Tools

should be stored in designated locations, and waste accumulation should be avoided.

(f) 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 radioactive material before removing any shielding.
(g) Minimization of sharp edges and the need to avoid sharp equipment in gloveboxes causing wounds that could become contaminated.
(h) The design of replaceable parts to facilitate segregation and handling of mixed and hazardous waste.
(i) Design provisions to facilitate surveillance and monitoring for ageing degradation.

Decontamination and dismantling

5.141. The types of floor, wall and ceiling surfaces selected in a nuclear fuel cycle R&D facility, particularly in wet chemical areas, are required to facilitate decontamination and future decommissioning (see paras 6.96(a) and 6.119(b) of SSR-4 [1]). Surfaces in areas where contamination might exist should be 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 of facilitating decontamination include the application of coverings or coatings to surfaces; for instance, by using paint, resins or stainless steel liners. Surfaces should be designed without corners or crevices that are difficult to access. In addition, all potentially contaminated surfaces should be made readily accessible to allow for periodic and final decontamination (e.g. by stripping of paint or coatings).

EMERGENCY PREPAREDNESS AND RESPONSE FOR A NUCLEAR FUEL CYCLE R&D FACILITY

5.142. Requirement 4 of GSR Part 7 [18] states that "**The government shall ensure that a hazard assessment is performed to provide a basis for a graded approach in preparedness and response for a nuclear or radiological emergency.**" The results of the hazard assessment provide a basis for identifying the emergency preparedness category relevant to the facility, as well as the on-site areas and, as relevant, off-site areas where protective actions and other response actions may be warranted in the case of a nuclear or radiological emergency. Further recommendations on emergency arrangements are provided in GS-G-2.1 [19].

5.143. Requirements for emergency preparedness and response at nuclear fuel cycle facilities are established in Requirements 47 and 72 and paras 6.181–6.183 and 9.120–9.132 of SSR-4 [1]. The operating organization of a nuclear fuel cycle R&D facility is required to establish arrangements for emergency preparedness and response that take into account the hazards identified and the potential consequences of an emergency associated with the facility (see Requirement 72 of SSR-4 [1]). The emergency plan and procedures and the necessary equipment and provisions are required to be based on the accidents analysed in the safety analysis report (see para. 9.124 of SSR-4 [1]). The conditions under which an off-site emergency response might need to be initiated include the internal hazards and external hazards identified as the postulated initiating events for a nuclear fuel cycle R&D facility (see paras 5.49–5.97 of this Safety Guide).

5.144. The emergency plan is required to cover all the functions to be performed in the response to an emergency (see para. 9.124 of SSR-4 [1]). It should also address the infrastructural elements (including training, drills and exercises) necessary to support these functions.

5.145. The R&D personnel running experiments should inform the operating organization of the hazards and the shutdown arrangements (i.e. to achieve a safe state) for the experiments under their control.

5.146. For Case 2 nuclear fuel cycle R&D facilities, the hazards listed in the relevant IAEA Safety Guide for a specific type of nuclear fuel cycle facility (e.g. SSG-6 (Rev. 1) [5], SSG-5 (Rev. 1) [21], SSG-7 (Rev. 1) [22], SSG-42 (Rev. 1) [23]) should be considered in the hazard assessment used for developing the emergency arrangements.

5.147. The safety analysis should identify those safety functions that should continue during and after events that might affect the operability of control rooms or control panels, for example fire or externally generated releases of hazardous chemicals. Appropriately located supplementary control rooms or panels (or alternative arrangements, such as emergency control panels) should be provided to ensure that the safety functions identified by this analysis can continue to be fulfilled.

5.148. The infrastructure for off-site emergency response (e.g. emergency centres, medical facilities) should be based on the site characteristics and the location of the nuclear fuel cycle R&D facility (see para. 9.122 of SSR-4 [1] and Requirement 24 of GSR Part 7 [18]).

AGEING MANAGEMENT AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.149. The design of a nuclear fuel cycle R&D facility is required to take into account the effects of ageing on SSCs important to safety to ensure their reliability and availability during the lifetime of the facility (see Requirement 32 of SSR-4 [1]).

5.150. The design of a nuclear fuel cycle R&D facility is required to facilitate the inspection of SSCs important to safety. This should include the detection of the effects of ageing (e.g. static containment deterioration, corrosion) and allow the maintenance or replacement of such items, if needed.

5.151. An ageing management programme is required to be implemented by the operating organization of a nuclear fuel cycle R&D facility (see Requirement 60 of SSR-4 [1]). This programme should be implemented at the design stage to allow equipment replacements to be anticipated.

6. CONSTRUCTION OF NUCLEAR FUEL CYCLE R&D FACILITIES

6.1. Requirements for the construction of a nuclear fuel cycle facility are established in Requirement 53 and paras 7.1–7.7 of SSR-4 [1]. Recommendations on the construction of nuclear installations are provided in IAEA Safety Standards Series No. SSG-38, Construction for Nuclear Installations [38].

6.2. For a complex nuclear fuel cycle R&D facility (e.g. a Case 2 facility), authorization by the regulatory body should be sought in several stages. Each stage may have a hold point at which approval by the regulatory body is necessary before the subsequent stage may be commenced, as described in para. 7.2 of SSR-4 [1].

6.3. Requirement 53 of SSR-4 [1] states that “**Items important to safety shall be constructed, assembled, installed and erected in accordance with established processes that ensure that the design specifications and design intent are met.**” The operating organization should implement effective processes to prevent the installation of counterfeit, fraudulent or suspect items, as well as non-conforming or substandard components. Such items or components could impair safety even after the commissioning of the nuclear fuel cycle R&D facility.

6.4. Modular components (e.g. gloveboxes, hot cells, fume hoods, monitoring systems) should be used, as far as practicable, in the construction of a nuclear fuel cycle R&D facility. This enables equipment to be tested and proven at the manufacturer's premises before installation in the facility. This approach also aids commissioning, maintenance and decommissioning.

6.5. The construction of parts of a nuclear fuel cycle R&D facility and the commissioning or operation of other parts of the same facility can overlap. Construction in areas where radioactive material is present can be significantly more difficult and time consuming. If this occurs, the operating organization for the facility should take measures to prevent the following:

(a) Unnecessary exposure of construction personnel to radiation;
(b) Damage to SSCs caused by construction activities;
(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 from construction activities.

These measures should also include the training of construction personnel on their own safety and the safety of others prior to the construction stage.

6.6. Consideration should be given to the quality assurance programme during the construction of a nuclear fuel cycle R&D facility. This programme should be prepared early in the construction stage and should include the following:

(a) Applicable codes and standards;
(b) The organizational structure;
(c) Design change programme (configuration control);
(d) Procurement control;
(e) Maintenance of records (see also para. 7.4 of SSR-4 [1]);
(f) Equipment testing;
(g) Coding and labelling of safety relevant components, cables, piping and other pieces of equipment;
(h) Receipt, handling, transport, storage, preservation and maintenance of SSCs.

7. COMMISSIONING OF NUCLEAR FUEL CYCLE R&D FACILITIES

7.1. Requirements for design provisions for the commissioning of nuclear fuel cycle facilities are established in Requirement 31 and para. 6.116 of SSR-4 [1]. Requirements for the commissioning programme for nuclear fuel cycle facilities are established in Requirement 54 and paras 8.1–8.27 of SSR-4 [1].

COMMISSIONING STAGES FOR A NUCLEAR FUEL CYCLE R&D FACILITY

7.2. In accordance with para. 8.12 of SSR-4 [1], the commissioning of a nuclear fuel cycle facility is required to be divided into stages, depending on the objectives to be achieved. For a nuclear fuel cycle R&D facility, this may involve three stages, which are described in paras 7.4–7.12 of this Safety Guide.

7.3. Some stages of commissioning may be subject to approval by the regulatory body, both prior to starting and at completion (see also paras 8.1 and 8.11 of SSR-4 [1]). The operating organization should define and agree with the regulatory body hold points (see para. 8.19 of SSR-4 [1]) and witness points to ensure proportionate inspection during commissioning. The purpose of these points should be principally to demonstrate safety in accordance with the safety analysis prior to advancement to the next stage of commissioning or operation. The operating organization is required to establish and maintain effective communication with the regulatory body throughout the commissioning process (see para. 8.11 of SSR-4 [1]) to ensure full understanding of the regulatory requirements and to maintain compliance with those requirements.

Stage 1: Cold commissioning

7.4. During cold (or ‘inactive’) commissioning, the nuclear fuel cycle R&D 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 SSR-4 [1]). Owing to the relative ease of taking corrective actions, as much verification and testing as possible should be performed at this stage.

7.5. At this stage, operating personnel should take the opportunity to learn the details of the systems and to further develop and finalize the operating procedures and associated documentation, including procedures relating to the operation and maintenance of the nuclear fuel cycle R&D facility and those relevant to any anticipated operational occurrences, including emergencies. The leaktightness of containment systems and the stability of control systems should be tested at this stage.

Stage 2: Warm commissioning

7.6. During warm (or 'trace active') commissioning, natural or depleted uranium should be used[10] as appropriate to avoid criticality risks, to minimize occupational exposure and to limit possible needs for decontamination. This stage provides the opportunity to initiate the control regimes that will be necessary when higher activity materials (e.g. plutonium, other actinides, fission products) are introduced into the nuclear fuel cycle R&D facility.

7.7. Safety tests performed during the warm commissioning stage should mainly be devoted to confinement checking. These tests should include checks for airborne radioactive material, smear checks on surfaces and checks for gaseous discharges and liquid releases. Checks should also be made for unexpected accumulations of hazardous material.

7.8. Prior to hot commissioning, the emergency arrangements (on-site and off-site arrangements, as appropriate) for the nuclear fuel cycle R&D facility are required to be established, including procedures, training, sufficient numbers of trained personnel, and emergency drills and exercises (see paras 8.14 and 8.15 of SSR-4 [1]).

Stage 3: Hot commissioning

7.9. The hot (or 'active' or 'hot processing') commissioning stage enables engineered systems and administrative controls to be progressively and cautiously brought into full operation, with radioactive material present. Paragraphs 8.16–8.18 of SSR-4 [1] establish requirements to fully confirm the performance of systems for radiation safety and criticality safety.

7.10. Hot commissioning should be performed under the responsibility, safety procedures and organization of the operating organization. Hot commissioning may be considered part of the operational stage of a nuclear fuel cycle R&D facility (see Section 8).

---

[10] In some States, the use of natural or depleted uranium may require regulatory approval.

7.11. The full operational radiation protection programme (see Requirement 67 of SSR-4 [1]) should be implemented, including individual monitoring.

7.12. The safety committee of the nuclear fuel cycle R&D facility is required to be established before hot commissioning commences (see Requirement 6 and paras 4.29 and 4.30 of SSR-4 [1]). Lessons learned from similar facilities should be applied, especially for the commissioning of a new nuclear fuel cycle R&D facility.

8. OPERATION OF NUCLEAR FUEL CYCLE R&D FACILITIES

ORGANIZATION OF OPERATION OF NUCLEAR FUEL CYCLE R&D FACILITIES

8.1. The specific hazards and their credible combinations associated with a nuclear fuel cycle R&D facility described in Section 2 should be taken into account in meeting the safety requirements for the operation of nuclear fuel cycle facilities established in section 9 of SSR-4 [1].

8.2. The activities relating to the operational functions and research functions of a nuclear fuel cycle R&D facility should be coordinated to ensure that safety is the overriding priority. The safety committee (see Requirement 6 of SSR-4 [1]) should provide an interface between operations and research; however, this interface should not be used as a substitute for procedures for everyday communication and cooperation on safety between these functions. Such procedures for communication and cooperation should be documented. Responsibilities that should be coordinated carefully include the management of radioactive material, the monitoring of experiments and the management of radioactive waste. The safety committee of the nuclear fuel cycle R&D facility should include representatives of operations, safety and research functions.

8.3. Research programmes should comply with the existing safety case or should be considered as a modification. Research involves flexibility in the materials and processes used, and the safety case should therefore anticipate a variety of research needs. 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.

8.4. Paragraph 9.3 of SSR-4 [1] establishes requirements relating to interdependencies and communication between facilities on the same site. Different organizational units within a nuclear fuel cycle 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 within the operating organization.

QUALIFICATION AND TRAINING OF PERSONNEL AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.5. Requirements for the qualification and training of nuclear fuel cycle facility personnel are established in Requirements 56 and 58 of SSR-4 [1]. Further recommendations are provided in paras 4.6–4.25 of GS-G-3.1 [10].

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

8.7. In 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 programmes or operations. The training should also include procedures for self-monitoring and the use of personal protective equipment.

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

8.9. Many processes relating to glovebox and hot cell operations involve manual intervention. Therefore, special attention should be paid to the training of nuclear fuel cycle R&D facility personnel who use gloveboxes and hot cells, including in the handling of tongs and manipulators. This training should include the actions to be taken in response to anticipated operational occurrences (e.g. a punctured glove in a glovebox, sleeve failure or loss of ventilation in a hot cell).

8.10. For nuclear fuel cycle R&D facilities containing a significant quantity of nuclear material, the complementary training of safety and nuclear security personnel and their mutual participation in both safety and nuclear security exercises should be part of the training programme to effectively manage the interface between safety and nuclear security. In particular, personnel with responsibilities and expertise in safety analysis and safety assessment should be provided with a working knowledge of the security arrangements at the facility. Similarly, security experts should be provided with a working knowledge of the safety considerations at the facility, so that potential conflicts between safety and nuclear security can be resolved effectively without safety and security measures compromising each other.

OPERATIONAL LIMITS AND CONDITIONS AND OPERATING PROCEDURES AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.11. Requirement 57 and paras 9.27–9.37 of SSR-4 [1] establish requirements for the operational limits and conditions to be developed for a nuclear fuel cycle facility. Operating personnel at a nuclear fuel cycle R&D facility should be clearly informed of the safety significance of the operational limits and conditions, including safety limits, safety system settings and limiting conditions for safe operation. Examples of SSCs relevant to defining operational limits and conditions for each process area are presented in Annex III to this Safety Guide.

8.12. To ensure that, under normal circumstances, the nuclear fuel cycle R&D facility operates well within its operational limits and conditions (see Requirement 57 of SSR-4 [1]), limiting conditions for safe operation are required to be defined by the operating organization (see para. 9.31 of SSR-4 [1]). The margins should be derived from the design considerations and from experience of operating the facility (both during commissioning and subsequently). The objective should be to set a sufficient safety margin while avoiding breaches of the limiting conditions for safe operation.

8.13. The authority to make operating decisions should be assigned to suitable levels of management, depending on the operational limits and conditions and the potential safety implications of the decision. The management system (see Section 3) should specify the authority and responsibilities at each management level. If an operational limit or condition is exceeded, the appropriate level of management should be informed (see also paras 9.34 and 9.35 of SSR-4 [1]). The circumstances that would necessitate an immediate decision or action for safety reasons should be defined, as far as practicable, in procedures developed in accordance with the management system. The appropriate personnel should be trained and authorized to make the necessary decisions and take the necessary actions in accordance with these procedures.

8.14. Any non-compliance with limits on operating parameters should be adequately investigated by the operating organization, and the lessons should be applied to prevent a recurrence. In accordance with regulatory requirements, the regulatory body should be notified in a timely manner of such non-compliances and of any immediate actions taken and should be kept informed of the subsequent investigations and their outcome.

8.15. A document that lists all the operational limits and conditions for the nuclear fuel cycle R&D facility should be prepared. Annex IV gives examples of operational limits and conditions applicable to Case 1 facilities and Case 2 facilities.

8.16. Operational limits and conditions that should be set for a nuclear fuel cycle R&D facility include the following, as applicable:

(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) Specific limits on concentrations, geometry and moderators in solutions containing fissile materials;
(c) Inventory limits of radioactive material and isotopic compositions in gloveboxes or interim storage areas;
(d) Limits on process parameters such as temperature, pressure and flow to ensure safe operation of the facility;
(e) Maximum heat loads specified for locations such as hot cells or gloveboxes;
(f) Maximum quantities of additives at different steps in the facility's processes;
(g) Specific limits on combustible material in gloveboxes and hot cells;
(h) Specific limits for flammable atmospheres in enclosed equipment, for example, hydrogen in a furnace;
(i) Specific limits on radiation and contamination levels in different areas.

8.17. The values of the key parameters in operational limits and conditions should be recorded for auditing purposes and to support periodic safety reviews. An investigation and learning process is required in the case of non-compliances with the operational limits and conditions (see paras 9.34 and 9.35 of SSR-4 [1]). The findings of such investigations should be recorded, and any lessons identified should be disseminated to the relevant operating personnel.

8.18. The operating organization should establish operating procedures to ensure safety during limited operation of the nuclear fuel cycle R&D facility, especially where this is followed by a long period of shutdown (see also para. 8.3 of SSR-4 [1]). Training programmes should reflect such procedures.

8.19. Operating procedures should include actions necessary to ensure criticality safety, radiation safety, chemical safety, fire safety, the protection of persons and the environment, and emergency preparedness and response.

8.20. Operating instructions and procedures are required to be reviewed periodically and updated as appropriate (see para. 9.68 of SSR-4 [1]).

8.21. In a nuclear fuel cycle R&D facility, measures should be taken to ensure that experiments and processes can be placed in a safe state. 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 and conditions, and other key safety information — is known, recorded and accessible at all times. Operating procedures should also be established for the use of the ventilation system in fire conditions.

8.22. The management of the nuclear fuel cycle R&D facility should arrange for pre-job briefings, including 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. All relevant personnel at the nuclear fuel cycle R&D facility should participate in such meetings. Post-job debriefings should also be conducted.

MAINTENANCE, CALIBRATION, PERIODIC TESTING AND INSPECTION AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.23. Requirements relating to maintenance, periodic testing and inspection for nuclear fuel cycle facilities are established in Requirement 65 and paras 9.74–9.82 of SSR-4 [1].

8.24. When conducting maintenance in a nuclear fuel cycle R&D facility, particular consideration should be given to the potential for surface contamination and airborne radioactive material, as well as to any chemical or biological hazards. The nuclear fuel cycle R&D facility should not be placed in an unsafe or unanalysed condition when performing periodic testing or routine maintenance.

8.25. Maintenance at a nuclear fuel cycle R&D facility should follow good practices, with particular consideration given to the following:

(a) The development of a suitable maintenance programme that includes all processes used in work control; for example, handover 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, the de-energizing and disconnection of electrical cables and hot or pressurized pipes, and the draining, venting and purging of equipment.
(c) Testing and monitoring; for example, checks of the workplace and tools before commencing work, monitoring during maintenance and checks for re-commissioning, and communications.
(d) Safety precautions for work; for example, specifications ensuring the availability and use of personal protective equipment.
(e) Continuous monitoring systems for control of criticality and for radiation protection.
(f) Reinstallation of equipment; for example, reassembly, reconnection of pipes and cables, testing, cleaning of the job site and monitoring after maintenance and before recommissioning.

8.26. A programme of periodic inspections of a nuclear fuel cycle R&D facility is required to be established and implemented (see Requirement 65 of SSR-4 [1]). At a minimum, this programme should include the periodic inspection of fume hoods, hot cells, gloveboxes and entrances to containment areas. The pressure drop across filter banks should be checked on a regular basis. Routine programmes of inspection and maintenance should be designed to prevent the spread of contamination or the release of hazardous material. These programmes should include activities such as the following:

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

8.27. Periodic testing of the fire detection and extinguishing systems for the nuclear fuel cycle R&D facility should be performed. The operational compliance of ventilation systems with fire protection requirements should also be verified on a regular basis.

8.28. The availability of materials necessary for maintenance should be verified regularly. To ensure continuity of safe operations of a nuclear fuel cycle R&D facility and to prevent the installation of counterfeit, fraudulent or suspect items, as well as non-conforming or substandard components, a programme for the provision of spare parts for items important to safety, including radiation monitoring equipment, should be established and implemented (see also para. 6.3).

8.29. The accurate and timely calibration of equipment is important for the safe operation of a nuclear fuel cycle R&D facility. Calibration procedures should cover equipment used by the nuclear fuel cycle R&D facility and by organizations that support the facility, such as analytical laboratories and suppliers of radiation protection equipment. The operating organization should satisfy itself that externally supplied or located equipment is properly calibrated at all times, in accordance with national or international standards, and that the records of calibration are traceable.

8.30. The frequency of calibration and periodic testing of instrumentation important to safety (including instrumentation located in analytical laboratories) should be defined in the operational limits and conditions, on the basis of the safety analysis.

AGEING MANAGEMENT FOR A NUCLEAR FUEL CYCLE R&D FACILITY

8.31. Requirements for an effective ageing management programme for nuclear fuel cycle facilities are established in Requirement 60 and paras 9.53–9.55 of SSR-4 [1]. In implementing these requirements, the operating organization of a nuclear fuel cycle R&D facility should take into account the following:

(a) Ensuring support for the ageing management programme by the management of the operating organization;
(b) Ensuring early implementation of an ageing management programme;
(c) Following a proactive approach based on an adequate understanding of the ageing of SSCs, rather than a reactive approach responding to the failure of SSCs;
(d) Ensuring optimal operation of SSCs to slow down the rate of ageing degradation;
(e) Ensuring the proper implementation of maintenance and testing activities in accordance with operational limits and conditions, design requirements

and manufacturers' recommendations, and following approved operating procedures;
(f) Minimizing human performance factors that could lead to premature degradation, through enhancement of staff motivation, fostering of a culture for safety (including a sense of ownership and awareness) and understanding of the basic concepts of ageing management;
(g) Ensuring the availability and use of correct operating procedures, tools and materials, and ensuring the availability of a sufficient number of qualified personnel for a given task;
(h) Collecting feedback on operating experience to learn from relevant ageing related events.

8.32. The ageing management programme should consider physical ageing as well as non-physical ageing (i.e. obsolescence or becoming out of date in comparison with current knowledge, codes, standards and regulations, and technology).

8.33. The surveillance undertaken as part of the ageing management programme (see para. 9.54 of SSR-4 [1]) should be implemented through regular checks performed by operating personnel, such as the following:

(a) Systematic monitoring of the condition of SSCs;
(b) Regular visual inspections of SSCs for evidence of deterioration due to ageing effects;
(c) Monitoring of operating conditions (e.g. taking heat images of electrical cabinets, checking the temperature of ventilator bearings).

CONTROL OF MODIFICATIONS AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.34. Nuclear fuel cycle R&D facilities are normally established in such a way that they can be utilized for a variety of 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. Where this modification involves a large increase in the scale of operations, the operating organization should plan the increase in stages, where possible, to permit the gathering of feedback and the validation of each stage.

8.35. Requirement 61 of SSR-4 [1] states that "**The operating organization shall establish and implement a programme for the control of modifications to the facility.**" The management system of a nuclear fuel cycle R&D facility should include a standard process for all modifications (see para. 3.18). A work control system, quality assurance procedures and appropriate testing procedures should be used for the implementation of modifications (including temporary modifications) at a nuclear fuel cycle R&D facility.

8.36. The operating organization of a nuclear fuel cycle R&D facility is required to inform the regulatory body of planned modifications, in accordance with regulatory requirements (see para. 9.57(h) of SSR-4 [1]). The impacts of modifications on the safety of the facility are required to be assessed by the operating organization, and the approval of the regulatory body may be required before the modifications are implemented (see paras 9.57(a) and 9.57(d) of SSR-4 [1]). The assessment of the impact on the safety of the facility should consider, in particular, human factors (e.g. the human–machine interface), alarm systems, operating procedures, and the qualification or requalification of personnel.

8.37. The operating organization of a nuclear fuel cycle R&D facility is required to prepare procedures and provide training to ensure that relevant personnel have the necessary competence and authority to ensure that modification projects are carefully controlled (see paras 9.56–9.59 of SSR-4 [1]). The safety of modifications should be assessed for potential hazards during installation, commissioning and operation.

8.38. Proposed modifications at a nuclear fuel cycle R&D facility should be reviewed in detail and be subject to approval by qualified and experienced persons to verify that the arguments used to demonstrate safety are suitably robust. This is particularly important if the modification could have an effect on criticality safety.

8.39. The level of detail of the safety assessments for modifications to a nuclear fuel cycle R&D facility and the degree of scrutiny to which they are subjected are required to be commensurate with the safety significance of the modification (see paras 9.58 and 9.59 of SSR-4 [1]).

8.40. The safety committee of the nuclear fuel cycle R&D facility is required to review any proposed modifications that might have significance for safety (see para. 4.31(d) of SSR-4 [1]). Suitable records should be kept of the committee's decisions and recommendations.

8.41. Safety related documentation is required to be updated to reflect modifications (see para. 9.57(f) of SSR-4 [1]). The plans for each modification at a nuclear fuel cycle R&D facility should also specify any documentation and training that will need to be updated (e.g. training programme, specifications, safety assessment, notes, drawings, engineering flow diagrams, process instrumentation diagrams, operating procedures).

8.42. Personnel involved in implementing a modification are required to be suitably trained and qualified (see para. 9.57(e) of SSR-4 [1]).

8.43. The management system for a nuclear fuel cycle R&D facility (see Section 3) should include a process for the overall monitoring of the progress of modifications and for ensuring that all proposals for modification receive a sufficient level of scrutiny. The documentation supporting the proposed modification should specify the functional (commissioning) checks that are necessary before the modified system is declared fully operational again.

8.44. Modifications to the design, layout, organization or procedures at a nuclear fuel cycle R&D facility might adversely affect nuclear security. The possible effects of such modifications on nuclear security are required to be considered to verify that safety measures and security measures do not compromise each other (see Requirement 75 of SSR-4 [1]).

8.45. The modifications made to a nuclear fuel cycle R&D facility (including modifications to the operating organization) should be reviewed on a regular basis to ensure that the cumulative effects of multiple modifications with minor safety significance do not have unforeseen effects on the overall safety of the facility. This review should be part of (or additional to) periodic safety review or an equivalent process.

CONTROL OF CRITICALITY HAZARDS AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.46. Requirements for criticality safety in the operation of a nuclear fuel cycle facility are established in Requirement 66 and paras 9.83–9.85 of SSR-4 [1]. Recommendations on criticality safety in all facilities and activities are provided in SSG-27 (Rev. 1) [3].

8.47. Operational aspects of criticality control in a nuclear fuel cycle R&D facility should be taken into consideration, including the following:

(a) Prevention of 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) Prevention of 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) Emergency drills and exercises (see paras 8.84–8.89);
(g) Periodic calibration or testing of criticality control and monitoring systems (e.g. material movement control, balances and scales).

8.48. 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 contribute to criticality safety. However, where there are any uncertainties in the characteristics of fissile material, conservative values are required to be used for parameters such as fissile material content and isotopic composition (see paras 6.140 and 6.156 of SSR-4 [1]). This is especially important when managing cell floor or glovebox sweepings and similar waste material.

8.49. Additional criticality safety measures may be necessary for activities such as maintenance work. Fissile material — including waste and residues arising from experiments, pilot processes, decontamination and maintenance activities — is required to be accumulated only in containers specifically designed and approved for that purpose (see para. 9.85(c) of SSR-4 [1]). Such containers should be stored in dedicated areas for which criticality safety is ensured.

RADIATION PROTECTION AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.50. Requirements for radiation protection during the operation of a nuclear fuel cycle facility are established in Requirement 67 and paras 9.90–9.101 of SSR-4 [1]. General requirements for radiation protection are established in Section 3 of GSR Part 3 [20]; recommendations on the implementation of requirements for the protection of workers are provided in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [39].

8.51. Requirement 67 of SSR-4 [1] states that "**The operating organization shall establish and implement a radiation protection programme.**" This programme should be established and maintained to fulfil the management's responsibility for protection and safety and should take into account the inventory and the variety of sources involved in the nuclear fuel cycle R&D facility. The radiation protection programme for a nuclear fuel cycle R&D facility is expected to include the following elements:

(a) Assignment of responsibilities (e.g. decision making; organizational arrangements, including for itinerant workers; safety committee);
(b) Designation and functions of qualified experts (e.g. in radiation protection, internal and external dosimetry, workplace monitoring, ventilation, occupational health, and radioactive waste management);
(c) Integration of radiation protection with other areas of health and safety (e.g. industrial hygiene, industrial safety, chemical safety, fire safety);
(d) Accountancy system for radiation generators and radioactive sources (providing their location, description, output, activity, and physical and chemical form, as appropriate);
(e) Designation of controlled areas and supervised areas;
(f) Local rules and procedures necessary for the protection and safety of workers and other persons;
(g) Provision of personal protective equipment;
(h) Arrangements for monitoring workers and the workplace;
(i) System for recording and reporting;
(j) Training programme;
(k) Methods for reviewing, auditing and correcting identified deficiencies;
(l) Emergency procedures;
(m) Programme for workers' health surveillance.

8.52. The operating organization of a nuclear fuel cycle R&D facility is required to ensure that doses are below authorized limits and are as low as reasonably achievable within any dose constraints set by the operating organization (see paras 9.91 and 9.93 of SSR-4 [1]). The operating organization should establish a policy to ensure that protection and safety is optimized using a systematic approach.

8.53. In a nuclear fuel cycle R&D facility, the possible exposure pathways (for workers and for members of the public) include internal exposure (through inhalation or ingestion of particulates, aerosols or gases) and external exposure. Paragraphs 9.90–9.94 of SSR-4 [1] require the establishment of an appropriate radiation protection programme to fulfil the operating organization's responsibility for protection and safety. For a nuclear fuel cycle R&D facility, the complexity and size of the facility, as well as the inventory of radioactive material, should be taken into account when establishing this programme. In addition, the possibility that the physical and chemical properties of the inventory might change inadvertently and result in unforeseen consequences should also be considered.

8.54. Equipment outside gloveboxes and hot cells, the rooms in the facility, and the surrounding environment should all be systematically and regularly monitored for dose rate and surface contamination. Any deviation above the normal ranges (e.g. hot spots, slow incremental increases in radiation level) should be noted, the reason for the increase should be identified and prompt corrective and/or mitigatory actions should be taken.

8.55. Radiation protection personnel (i.e. radiation protection managers, radiation protection officers and associated staff) should be involved in decision making associated with the optimization of protection and safety (e.g. the early detection and mitigation of hot spots) and proper housekeeping (e.g. waste segregation, packaging and removal).

8.56. Intrusive maintenance[11] and modifications should be regarded as major activities that involve justification by facility management. The procedures for such activities should include the following:

(a) Estimation of doses (external and internal) prior to the activity.
(b) Preparatory activities to minimize individual and collective doses, including:
    (i) Identification of specific risks associated with the activities;
    (ii) Use of additional shielding, remote devices or mock-ups, as appropriate;
    (iii) Definition of specific procedures within the work permit (e.g. on the use of respiratory protective equipment, protective clothing and time limitations).
(c) Measurement of the doses received during the activities.
(d) Use of feedback to identify possible improvements.

8.57. During the operation of a nuclear fuel cycle 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. The aim should be

[11] Intrusive maintenance is maintenance involving a significant reduction in shielding, the breaking of static containment or a significant reduction of dynamic containment, or a combination of these.

to achieve low levels of airborne activity and surface contamination in the facility, taking into account the physical, chemical and radiological characteristics of specific radionuclides potentially present.

(b) Regular contamination surveys of facility areas and equipment should be performed to confirm the adequacy of cleaning programmes.

(c) The operating organization is required to designate controlled areas and supervised areas (see para. 5.27). In addition, to further identify the risk involved in a task, facility areas should be classified into radiation and contamination zones that are demarcated with appropriate warning signs. The boundaries between such zones should be regularly checked and adjusted to match current conditions.

(d) Access to areas designated as controlled areas owing to the presence of contamination should be avoided by nuclear fuel cycle R&D facility personnel with skin wounds.

(e) Continuous air monitoring should be performed, as indicated by the safety assessment, to alert operating personnel if airborne contamination is present. Mobile air samplers should be deployed, as necessary. A prompt investigation should be performed if high levels of airborne contamination have been detected.

(f) Personnel should be trained in putting on, using and taking off personal protective equipment with the assistance of radiation protection personnel. Personal protective equipment is required to be maintained in good condition, periodically inspected and kept readily available (see para. 3.95 of GSR Part 3 [20]).

(g) A high standard of housekeeping is required to be maintained within the facility (see Requirement 64 of SSR-4 [1]). Cleaning techniques should be used that do not give rise to airborne contamination.

(h) The effectiveness of the ventilation system should be checked regularly; if necessary, the system should be rebalanced following the isolation or de-isolation of boxes and fume hoods.

(i) 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.

(j) Careful consideration should be given to the combination of radiological hazards and non-radiological hazards (e.g. oxygen deficiency, heat stress), with particular attention paid to balancing the risks and benefits associated with the use of personal protective equipment, especially for air-fed systems.

8.58. Entry into and exit from work areas should be controlled to prevent the spread of contamination. In particular, rooms for changing clothes and decontamination stations should be available. Personnel and equipment should be checked for contamination and should be decontaminated, if necessary, prior to crossing boundaries between contamination zones or exiting such zones.

8.59. During periodic testing, inspection and maintenance of a nuclear fuel cycle R&D facility, precautions should be taken to limit the spread of contamination by means of temporary enclosures and additional ventilation systems, as appropriate.

8.60. On completion of maintenance work, areas should be decontaminated, and air sampling and surface contamination checks should be performed 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.

8.61. There should be careful preparation before entry into hot cells or gloveboxes that have contained radioactive material. 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 necessary. Such activities should be subject to pre-job approval (e.g. using work permits) and are required to be performed in accordance with local rules (see para. 3.94 of GSR Part 3 [20]).

8.62. There may be areas in a nuclear fuel cycle R&D facility where specific arrangements are needed to control external radiation exposure. Typically, these will be areas in Case 2 facilities, such as pilot processing facilities, where bulk quantities of radioactive material and other radioactive sources are stored and handled. Radiation levels should be controlled within a nuclear fuel cycle R&D facility by the following means:

(a) Ensuring that areas of high occupancy are remote from, or appropriately shielded from, significant quantities of radioactive material;
(b) Ensuring the removal of radioactive material from the vicinity of areas in which extended maintenance work is planned;
(c) Ensuring that the instrumentation that contains radiation sources is only used by suitably qualified and experienced personnel;
(d) Performing regular radiation dose rate surveys.

8.63. External radiation exposure should be controlled within a nuclear fuel cycle R&D facility by the following means:

(a) Training personnel on radiation hazards and in the use of appropriate workplace monitoring equipment;

(b) Avoiding unnecessary occupancy of controlled areas, limiting the working time near radiation sources (e.g. through administrative control or by using direct reading dosimeters with alarm), and maximizing the distance from such sources;
(c) Using temporary shielding and, where appropriate, individual shielding (e.g. eye protection, lead aprons).

8.64. When working in gloveboxes, the hands can receive a much higher dose than other parts of the body. In such cases, the exposure of the extremities should be monitored (e.g. through the use of finger dosimeters).

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

8.66. Additional controls may be necessary if radioactive material with higher specific activity is used. A comprehensive assessment of doses (occupational exposure and public exposure) should be performed before introducing such radioactive material.

8.67. Where an assessment of occupational exposure is required (see Requirement 25 of GSR Part 3 [20]), this assessment should be based on personal dosimeters, as described in paras 5.29(c), 5.103(e) and 8.64 of this Safety Guide. The assessment of internal exposures, where necessary, may be based on the collection of air sampling data. In vivo (whole body) monitoring and biological sampling (e.g. nose blows, faecal and urine samples) should also be available, as necessary, for routine monitoring and/or accident conditions, as a complementary measure to monitor internal exposure.

8.68. Further recommendations on occupational radiation protection and the assessment of internal exposure and external exposure, including recommendations on decontamination, are provided in GSG-7 [39].

MANAGEMENT OF FIRE SAFETY, CHEMICAL SAFETY AND INDUSTRIAL SAFETY AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.69. Requirements for protection against fire and explosion are established in Requirement 69 and paras 9.109–9.115 of SSR-4 [1]. Requirements relating to industrial and chemical safety are established in Requirement 70 and paras 9.116 and 9.117 of SSR-4 [1].

8.70. The non-radiological hazards that could be present in a nuclear fuel cycle R&D facility 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 material, pyrophoric metals, hydrogen, ammonium nitrate or ammonia;
(c) Asphyxiation hazard due to the presence of nitrogen, carbon dioxide or inert gases.

8.71. In a fire, dynamic confinement systems (including filtration) should continue to operate effectively to remove smoke, heat and particulates and to compensate for potential overpressure, as appropriate. Operation of the dynamic confinement system should be maintained for as long as temperatures at filters are below the threshold at which containment would be lost, as determined by the safety analysis. The fire hazard analysis (see Requirement 22 of SSR-4 [1]) should be updated at periodic intervals to incorporate changes that might affect the likelihood of a fire. Computer modelling may be used to support the fire hazard analysis.

8.72. Personnel should be informed about the chemical hazards that exist within the nuclear fuel cycle R&D facility. Operating personnel are required to be properly trained with respect to the hazards associated with the process chemicals (see para. 9.117 of SSR-4 [1]) to enable them to adequately identify and respond to the problems that might lead to chemical accidents.

8.73. The exposure of personnel to chemical hazards should be assessed using a method similar to that used for the assessment of radiation exposure, which should be based on the collection of data from air sampling in the workplace, in combination with personnel occupancy data. This method should be assessed and reviewed, as appropriate, by the corresponding regulatory authority. Limiting values for exposure to various chemical hazards are provided in Ref. [32].

8.74. As required by national regulations, a health surveillance programme should be established to monitor the health of nuclear fuel cycle R&D facility personnel who might be exposed to harmful chemicals.

MANAGEMENT OF RADIOACTIVE WASTE AND EFFLUENTS AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.75. Requirements relating to the management of radioactive waste and effluents in the operation of a nuclear fuel cycle facility are established in Requirement 68 and paras 9.102–9.108 of SSR-4 [1].

8.76. All operating personnel at a nuclear fuel cycle R&D facility should be trained in the waste management hierarchy (i.e. eliminate, reduce, reuse, recycle and dispose; see para. 4.6 of GSR Part 5 [2]), the waste management programme for the facility and the relevant procedures. Waste minimization targets should be set and regularly reviewed, and a system for continuous improvement (i.e. minimization of waste volumes and waste activity in relation to the work performed) should be implemented.

8.77. All radioactive waste generated at a nuclear fuel cycle R&D facility should be treated and stored in accordance with pre-established criteria, taking into account any national waste classification schemes. Waste management involves consideration of both on-site and off-site waste storage capacity, as well as disposal options and available disposal facilities. Every effort should be made to characterize the waste as fully as possible, especially waste for which a disposal route has not yet been identified. Where a disposal route does exist, waste characterization should be performed in such a way that compliance with waste acceptance criteria can be demonstrated. The information characterizing the waste is required to be held and to be retrievable (see paras 9.104 and 9.106 of SSR-4 [1]).

8.78. Operational arrangements should be such that the requirement to minimize the generation of radioactive waste of all kinds (see para. 9.102 of SSR-4 [1]) is met (e.g. by reducing the generation of secondary waste and through the reuse, recycling and decontamination of materials). Trends in the generation of radioactive waste at a nuclear fuel cycle R&D facility should be monitored, and the effectiveness of the waste reduction and minimization measures applied should be demonstrated. Equipment, tools and consumable material entering hot cells, shielded boxes and gloveboxes should be minimized as far as practicable.

8.79. Any radioactive waste generated at a nuclear fuel cycle R&D facility is required to be characterized (see paras 6.94 and 9.103 of SSR-4 [1]). This characterization should include a determination of the waste's physical, chemical and radiological properties to allow its subsequent management (i.e. appropriate pretreatment, treatment, conditioning and selection or determination of a temporary storage or disposal route). To the extent possible, the management of waste should ensure that all waste will meet the specifications for temporary storage or disposal, as appropriate. Particular care should be taken to segregate waste containing fissile material and to ensure criticality safety for such waste (see also paras 9.84 and 9.85 of SSR-4 [1]).

8.80. 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 with the aim of avoiding uncontrolled or unexpected reactions. Mixed waste that also contains non-radioactive toxic substances should be managed properly based on the waste acceptance criteria of the storage or disposal facilities.

8.81. When legacy waste exists for which there are no data from chemical and/or radiological analyses, the reports from the R&D programmes that produced this waste should be collected and 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 waste should be evaluated. In the absence of relevant radiological or chemical records, legacy waste should be analysed to determine its radiological and chemical properties, and any hazards should be quantified.

8.82. Clearance procedures for radioactive waste should be provided in accordance with regulatory requirements. These procedures should be used as fully as practicable to minimize the volumes of radioactive waste and thus the size of disposal facility necessary. Before the clearance of equipment for reuse, recycling or disposal, the equipment should be decontaminated to the level required by the regulatory body. Criteria for clearance of material from facilities are set out in schedule I of GSR Part 3 [20].

8.83. Periodic estimates of the impact on the public of radioactive discharges from the nuclear fuel cycle R&D facility (i.e. based on estimated dose to the representative person(s)) should be made using data on effluent releases and standard models agreed with the regulatory body. An environmental monitoring programme is required (see para. 9.108 of SSR-4 [1]), and the results of this programme should be used to verify the impact of discharges (and any unplanned releases) on the public and on the surrounding area to identify any trends and to assess public exposure.

EMERGENCY PREPAREDNESS AND RESPONSE FOR A NUCLEAR FUEL CYCLE R&D FACILITY

8.84. General requirements for emergency preparedness and response are established in GSR Part 7 [18]. Supporting recommendations on emergency arrangements are provided in GS-G-2.1 [19] and in IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [40]. Requirements for emergency preparedness and response at nuclear fuel cycle facilities are established in Requirement 72 and paras 9.120–9.132 of SSR-4 [1].

8.85. The emergency arrangements established at a nuclear fuel cycle R&D facility should consider the layout of the site, which may be composed of a large number of buildings and facilities.

8.86. As part of emergency preparedness, arrangements are required to be developed for coordination between the operating organization and the local, regional and national emergency response organizations, as appropriate (see para. 3.1 and Requirement 22 of GSR Part 7 [18]). These arrangements are required to be tested periodically to ensure that emergency response functions are performed effectively during a nuclear or radiological emergency (see Requirement 25 of GSR Part 7 [18] and para. 9.130 of SSR-4 [1]).

8.87. Suitable, reliable and diverse means of communication are required to be established with local authorities and response organizations (see para. 5.43 of GSR Part 7 [18]).

8.88. Requirement 10 of GSR Part 7 [18] states:

> “**The government shall ensure that arrangements are in place to provide the public who are affected or are potentially affected by a nuclear or radiological emergency with information that is necessary for their protection, to warn them promptly and to instruct them on actions to be taken.**”

8.89. The emergency arrangements for a nuclear fuel cycle R&D facility are required to be periodically reviewed and updated (see para. 9.131 of SSR-4 [1]). In performing this review, any lessons 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.

8.90. When establishing procedures for access control during emergencies at a nuclear fuel cycle R&D facility, when there is a necessity for rapid access and egress of personnel, safety and security specialists should cooperate closely. Both safety and nuclear security objectives should be sought during emergencies to the extent possible, in accordance with regulatory requirements.

FEEDBACK ON OPERATING EXPERIENCE AT A NUCLEAR FUEL CYCLE R&D FACILITY

8.91. Requirements for feedback on operating experience at a nuclear fuel cycle facility are established in Requirement 73 and paras 9.133–9.137 of SSR-4 [1]. Recommendations on programmes for operating experience feedback are provided in SSG-50 [15].

8.92. The programme for feedback on operating experience at a nuclear fuel cycle R&D facility is required to cover experience and lessons learned from events (including low level events) and accidents at the facility as well as from other nuclear fuel cycle facilities worldwide (see para. 9.133 of SSR-4 [1]). Lessons from relevant events at other (i.e. non-nuclear) facilities should also be considered. This programme should include the evaluation of trends in operational disturbances, trends in malfunctions, near misses and other incidents that have occurred at the nuclear fuel cycle R&D facility and, if applicable, at other nuclear installations. The programme is required to include a reporting system and consideration of technical, human and organizational factors (see paras 9.134 and 9.135 of SSR-4 [1]).

8.93. Useful information on the causes and consequences of many of the most important anomalies and accidents that have been observed in nuclear fuel cycle R&D facilities and other nuclear fuel cycle facilities is provided in the Fuel Incident Notification and Analysis System (FINAS) database[12].

[12] http://finas.iaea.org

9. PREPARATION FOR DECOMMISSIONING OF NUCLEAR FUEL CYCLE R&D FACILITIES

9.1. General requirements for the decommissioning of facilities are established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [41]. Requirements for preparation for the decommissioning of a nuclear fuel cycle facility are established in Requirement 74 and paras 10.1–10.13 of SSR-4 [1]. The operating organization of a nuclear fuel cycle R&D facility is required to allocate adequate financial resources for safe decommissioning where these are not provided by the government (see para. 4.2(e) of SSR-4 [1]).

9.2. The decommissioning plan is required to be periodically reviewed and updated throughout the lifetime of the nuclear fuel cycle R&D facility (see paras 7.5 and 7.6 of GSR Part 6 [41] and paras 10.1, 10.2 and 10.9 of SSR-4 [1]) to take into account new information and emerging technologies. The aim should be to ensure the following:

(a) The (updated) decommissioning plan is realistic and can be performed safely.
(b) Updated provisions are made for adequate decommissioning resources and their availability, when needed.
(c) The anticipated radioactive waste remains compatible with available (or planned) temporary storage capacities and disposal facilities, including any transport and treatment.

9.3. Requirements for design features to facilitate decommissioning are established in Requirement 33 and para. 6.119 of SSR-4 [1]. The following measures should be applied at the design, construction and operation stages of a nuclear fuel cycle R&D facility to facilitate its eventual decommissioning:

(a) Identification of reasonably achievable changes to the facility design to facilitate or accelerate decommissioning;
(b) Specific design measures to prevent contamination from penetrating structural materials (e.g. the installation of pond liners);
(c) Engineered controls and administrative controls to prevent the spread of contamination during operation of the facility;
(d) Consideration of the implications for decommissioning resulting from modifications and experiments in the facility, when they are proposed;
(e) 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 (see also para. 7.6 of SSR-4 [1]).

9.4. The radiological hazards associated with preparation for the decommissioning of a nuclear fuel cycle R&D facility depend on the type of work performed. Either this work should already be addressed by the existing decommissioning plan for the facility or for individual experiments, or the plan should be subject to 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:

(a) In high activity equipment, beta and gamma surface contamination might exist that needs prior decontamination by chemical or mechanical means (e.g. chemical rinses, sand blasting, use of specialized tools). The objective should be to reduce contamination levels to allow direct access to the equipment. If dose rates remain high after decontamination, remote handling should be used.
(b) In equipment containing alpha emitters in solution, surface contamination may need to be removed by rinsing with chemicals other than those used during operation.
(c) In equipment containing powdered alpha emitters, deposits of powder could remain, and the use of appropriate personal protective equipment should be considered.

9.5. The preparatory steps for the decommissioning of a nuclear fuel cycle R&D facility should include the following:

(a) Preparation of safety assessments and method statements, as required by the regulatory body for the licensing of the decommissioning process;
(b) Post-operational clean-out to remove all bulk quantities of radioactive material and other hazardous materials, followed by their segregation and their storage, disposal or transfer to an authorized waste management facility, as necessary;
(c) Identification of contaminated parts of buildings and equipment and identification of the radionuclides present;
(d) Characterization of the types and levels of contamination;
(e) Decontamination of the facility to reach the levels required by the regulatory body for final decommissioning.

9.6. For any period of shutdown (planned or unplanned) prior to decommissioning, the implications for the safety of the nuclear fuel cycle R&D facility are required to be assessed and managed (see para. 10.9 of SSR-4 [1]). Safety measures should be implemented to maintain the nuclear fuel cycle R&D facility in a safe and stable state, including measures to prevent criticality and the spread of contamination and fire and to maintain appropriate radiological monitoring. The need to revise the safety assessment for the facility in its shutdown state should be considered. The application of knowledge management methods to retain the knowledge and experience of operating personnel in a durable and retrievable form should also be considered. Wherever practicable, hazardous and corrosive materials should be removed from process equipment to safe storage locations before the nuclear fuel cycle R&D facility is placed into a prolonged shutdown state.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSR-4, IAEA, Vienna (2017).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, IAEA, Vienna (2009).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Criticality Safety in the Handling of Fissile Material, IAEA Safety Standards Series No. SSG-27 (Rev. 1), IAEA, Vienna (2022).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Governmental, Legal and Regulatory Framework for Safety, IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), IAEA, Vienna (2016).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-6 (Rev. 1), IAEA, Vienna (2023).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, IAEA, Vienna (2011), https://doi.org/10.61092/iaea.ko2c-dc4q

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 27-G, IAEA, Vienna (2018).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Security During the Lifetime of a Nuclear Facility, IAEA Nuclear Security Series No. 35-G, IAEA, Vienna (2019).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016), https://doi.org/10.61092/iaea.cq1k-j5z3

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006). (A revision of this publication is in preparation.)

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership, Management and Culture for Safety in Radioactive Waste Management, IAEA Safety Standards Series No. GSG-16, IAEA, Vienna (2022).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.4, IAEA, Vienna (2008).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Survey and Site Selection for Nuclear Installations, IAEA Safety Standards Series No. SSG-35, IAEA, Vienna (2015).

[18] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, INTERPOL, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, PREPARATORY COMMISSION FOR THE COMPREHENSIVE NUCLEAR-TEST-BAN TREATY ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, WORLD METEOROLOGICAL ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015), https://doi.org/10.61092/iaea.3dbe-055p

[19] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007). (A revision of this publication is in preparation.)

[20] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014), https://doi.org/10.61092/iaea.u2pu-60vm

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Conversion Facilities and Uranium Enrichment Facilities, IAEA Safety Standards Series No. SSG-5 (Rev. 1), IAEA, Vienna (2023).

[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-7 (Rev. 1), IAEA, Vienna (2023).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Reprocessing Facilities, IAEA Safety Standards Series No. SSG-42 (Rev. 1), IAEA, Vienna (in press),
https://doi.org/10.61092/iaea.3egz-z7fy

[24] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Nuclear Energy — Fissile Materials — Principles of Criticality Safety in Storing, Handling and Processing, ISO 1709, ISO, Geneva (2018).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).
[26] INTERNATIONAL ATOMIC ENERGY AGENCY, WORLD METEOROLOGICAL ORGANIZATION, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-18, IAEA, Vienna (2011).
[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Volcanic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-21, IAEA, Vienna (2012).
[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).
[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).
[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (2023).
[31] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Prospective Radiological Environmental Impact Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSG-10, IAEA, Vienna (2018).
[32] AMERICAN CONFERENCE OF GOVERNMENTAL INDUSTRIAL HYGIENISTS, 2024 Threshold Limit Values (TLVs) and Biological Exposure Indices (BEIs), ACGIH, Cincinnati (2024).
[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Analysis and Licensing Documentation for Nuclear Fuel Cycle Facilities, Safety Reports Series No. 102, IAEA, Vienna (2020).
[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Reassessment for Nuclear Fuel Cycle Facilities in Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, IAEA Safety Reports Series No. 90, IAEA, Vienna (2016).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSG-3, IAEA, Vienna (2013).
[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Classification of Radioactive Waste, IAEA Safety Standards Series No. GSG-1, IAEA, Vienna (2009).
[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-41, IAEA, Vienna (2016).
[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Construction for Nuclear Installations, IAEA Safety Standards Series No. SSG-38, IAEA, Vienna (2015).
[39] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA, Vienna (2018).

[40] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-2, IAEA, Vienna (2011).

[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6, IAEA, Vienna (2014).

Annex I

PROCESS ROUTE IN A NUCLEAR FUEL CYCLE R&D FACILITY: LABORATORY SCALE (CASE 1)

I–1. Case 1 nuclear fuel cycle research and development (R&D) facilities involve small scale experiments, analyses and fundamental research studies conducted on the chemical, physical, mechanical and/or 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. Figure I–1 shows a typical process route in such a facility.

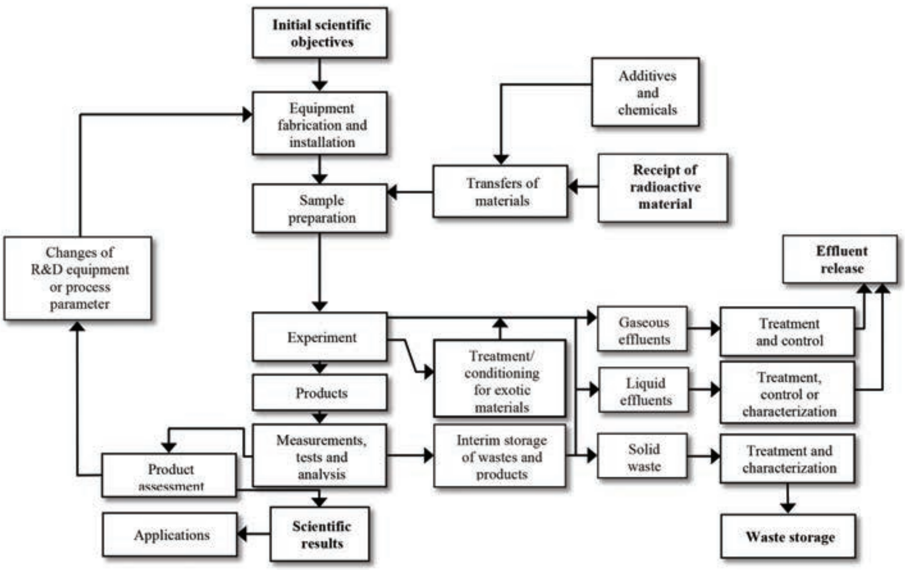

FIG. I–1. General processes in a nuclear fuel cycle R&D facility operating at laboratory scale (Case 1).

Annex II

PROCESS ROUTE IN A NUCLEAR FUEL CYCLE R&D FACILITY: PILOT SCALE (CASE 2)

II–1. Case 2 nuclear fuel cycle research and development (R&D) facilities involve R&D on processes and equipment envisaged for use on a pilot scale. Figure II–1 shows a typical process route in such a facility.

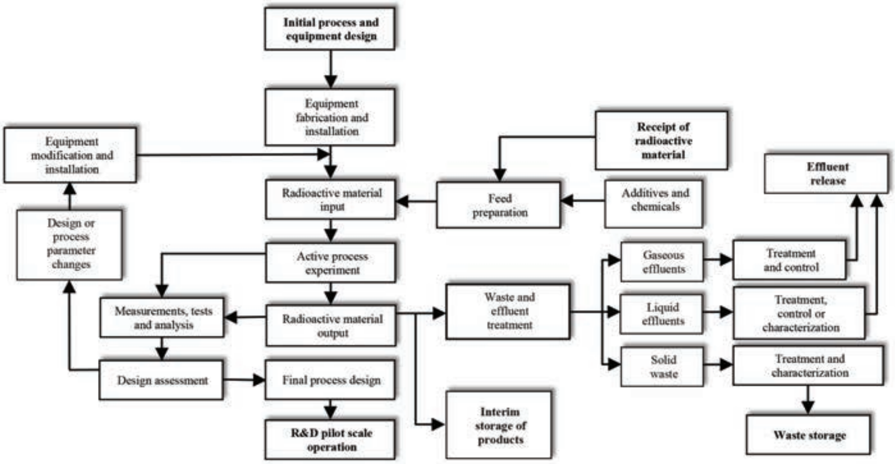

FIG. II–1. General processes in a nuclear fuel cycle R&D facility operating on a pilot scale (Case 2).

Annex III

EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES

III–1. The main safety functions of a nuclear fuel cycle research and development (R&D) facility are as follows:

(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.

With regard to these safety functions, Table III–1 lists the structures, systems and components important to safety at a nuclear fuel cycle R&D facility and gives examples of operational limits and conditions and/or means of mitigation in the event that these safety functions are challenged.

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
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 in accordance with the safety case and set procedures Accessibility and visibility to allow for periodic inspection and maintenance
	Equipment ensuring mass and concentration control	Criticality accident	1	Quality of the design and construction with diverse and robust control of key parameters Installation in accordance with the safety case and set procedures with realistic commissioning tests

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Equipment fabrication and installation (cont.)	Building, fume hoods, gloveboxes, hot cells, interim storage, ventilation system, filters	Contamination Loss of integrity	2	Quality of the design and construction Use of fail-safe designs where possible Installation in accordance with the safety case and set procedures Realistic commissioning tests Measurement points for airflow and pressure Accessibility and 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 for radiation protection Validation of the shielding during commissioning
Receipt of radioactive material	Transportation means	Degradation of criticality safety margin	1 (fissile material only)	Transport regulations and procedures Verification by the recipient in accordance with operational limits and conditions

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Receipt of radioactive material (cont.)	Measurement devices for isotopic and chemical composition	Violation of acceptance criteria Unexpected or exotic material (see para. 2.3(e))	1, 2 and 3	Suitably qualified and experienced personnel Non-destructive examination 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 regulations and procedures On-site transportation rules Authorized personnel Surface contamination tests, brake tests
	Transport 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 Surface contamination tests, pressure tests

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Receipt of radioactive material (cont.)	Shielding of transport container	Insufficient shielding	3	Transport 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 (non-radiological 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. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
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, nuclear fuel cycle 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} [a]	1	Procedures, criticality control measures, moderator limits, etc. Calibration of structures, systems and components
	Criticality accident alarm system	Unavailability of alarm	1	Procedures for the transfer of fissile materials and personnel access and egress in the case of a criticality accident
	Fume hoods, hot cells or gloveboxes	Breach of containment	2	Maintenance and periodic testing Permissible pressure

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Sample/feed preparation (cont)	Fume hoods, hot cells or shielded gloveboxes	Insufficient shielding	3	Maintenance and periodic checks for purposes of radiation protection
Performance of experiments	Calibrated equipment 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)
	Fume hoods, hot cells or gloveboxes Pressure monitoring and recording	Breach of containment	2	Effective isolation procedures Maintenance and periodic testing

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Performance of experiments (cont.)	Emergency power supply	Loss of power	3	System dependent procedures (e.g. for low battery voltage) Maintenance and periodic testing
	Fire protection system	Uncontrolled fire Accumulations of flammable materials, blocked exits	2	Evaluation of 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 or in chemical or physical form (e.g. precipitation) Maintenance and periodic testing of equipment

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Products (cont.)	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 conditions Implementation of conservative procedures Checks for purposes of radiation protection, surface contamination tests, measurements of pool water activity Placement of the nuclear fuel cycle R&D facility in a safe state Maintenance and periodic testing
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	Checks of the consistency of the material (i.e. with that assumed in the safety case) Operational limits and conditions Calibration, regular inspections Maintenance and periodic testing

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Application	None	Hazard transferred to a third party (customer of the facility)	1, 2 and 3	Quality assurance, applied to work conducted by the nuclear fuel cycle R&D facility with some transfer of knowledge and safety information to the user: — Product identified (labelled) and capable of being safely handled — Documentation and training of third parties and customers — Checks on export packages prior to use Transference of responsibility for the subsequent safety of the product and its application from the nuclear fuel cycle R&D facility to the user or third party

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
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
	Scrubbers, HEPA filters, connections and casings	Excessive contact dose rate on filter casing Deposition of radioactive particulates in ventilation system	3	Periodic checks for the purposes of radiation protection
Liquid effluents	Ion exchange resins and extraction	Abnormal presence of fissile material	1	Periodic testing by gamma and neutron counting Accountability Contamination tests Criticality controls

TABLE III–1. EXAMPLES OF STRUCTURES, SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY AND POSSIBLE CHALLENGES TO SAFETY FUNCTIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES (cont.)

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged (para. III–1)	Examples of operational limits and conditions and/or means of mitigation
Liquid effluents (cont.)	Connections, equipment for retention of filtering medium or resin (e.g. prevention of backflow)	Presence of leak	2	Measurements, periodic testing 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 operating personnel	3	Good planning and periodic checks for the purposes of radiation protection, as defined by procedures and regulatory limits
	Containers	Excessive contact dose rate on containers Breach of containment	2	Measurements (e.g. surface contamination tests) and periodic testing as defined by procedures and regulatory limits
	Shielding on containers	Exposure from packaging and increased risk to nuclear fuel cycle R&D facility operators	3	Checks for the purposes of radiation protection, records of radioactive materials and discharges

[a] k_{eff} is the ratio between the number of fissions in two succeeding generations (later to earlier) of the chain reaction.

Annex IV

EXAMPLES OF OPERATIONAL LIMITS AND CONDITIONS FOR NUCLEAR FUEL CYCLE R&D FACILITIES

IV–1. Operational limits and conditions are a set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel for the safe operation of a facility. Table IV–1 gives examples of some of the operations in a nuclear fuel cycle research and development facility and the corresponding operational limits and conditions that might be considered.

TABLE IV–1. EXAMPLES OF OPERATIONAL LIMITS AND CONDITIONS FOR A NUCLEAR FUEL CYCLE RESEARCH AND DEVELOPMENT FACILITY

Area or operation	Example operational limit or condition
Radiation protection in hot cells or shielded gloveboxes	No more than 100 mL of radioactive product, or 1 TBq ^{131}I 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 hydrogen:uranium atomic ratio is not to exceed 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

Abe, H.	Japan Atomic Energy Agency, Japan
Blanchard, A.L.	Conuar S.A., Argentina
Cols, H.	International Atomic Energy Agency
Khotylev, V.	Canadian Nuclear Safety Commission, Canada
Montalti, P.	Institute for Radiation Protection and Nuclear Safety, France
Rodin, A.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Rovny, J.	International Atomic Energy Agency
Shaw, P.	International Atomic Energy Agency
Shokr, A.	International Atomic Energy Agency
Siurano-Perez, O.	Nuclear Regulatory Commission, United States of America
Valiveti, L.N.	International Atomic Energy Agency
Wright, R.	International Atomic Energy Agency

IAEA
International Atomic Energy Agency

CONTACT IAEA PUBLISHING

Feedback on IAEA publications may be given via the on-line form available at:
www.iaea.org/publications/feedback

This form may also be used to report safety issues or environmental queries concerning IAEA publications.

Alternatively, contact IAEA Publishing:

Publishing Section
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530
Email: sales.publications@iaea.org
www.iaea.org/publications

Priced and unpriced IAEA publications may be ordered directly from the IAEA.

ORDERING LOCALLY

Priced IAEA publications may be purchased from regional distributors and from major local booksellers.

25-00201E-T

Safety through international standards

ISBN 978-92-0-102025-3

International Atomic Energy Agency
Vienna, 2025
www.iaea.org