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Safety of Nuclear Fuel Cycle Research and Development Facilities

(Revision of SSG-43)

DRAFT SPECIFIC SAFETY GUIDE

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 receive, handle, process and store various nuclear materials including uranium, other actinides and fission products, and activated materials in multiple physical forms such as powders, liquids and gases. These can present diverse hazards such as: nuclear and radiological hazards, toxic hazards from biologically active materials or chemicals (e.g. hydrofluoric acid, uranium hexafluoride or ammonia), or 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 toxic materials in such facilities can be significant. Consequently, all the relevant safety requirements for the management of nuclear fuel cycle facilities and activities, such as learning from experience, inspection and maintenance, apply to such R&D facilities. The relevant safety requirements for specific types of facility also apply to nuclear fuel cycle R&D facilities where similar operations are performed.

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

1.6. This Safety Guide supersedes IAEA Safety Standards Series No. 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 R&D facilities, regulatory bodies, designers, and other relevant organizations.

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Standards Series No. SSG-42 Safety of Nuclear Fuel Cycle Research and Development Facilities, Vienna, 2017

SCOPE

1.9. The safety requirements applicable to 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 the 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:

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

This Safety Guide also applies to the 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. The scope of this Safety Guide is limited to the safety of nuclear fuel cycle R&D facilities and the protection of workers, the public and the environment. This Safety Guide does not deal with 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 the requirements established in SSR-4 [1] (see paras 6.94–6.99 and 9.102–9.108), and 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 publication supplement more detailed recommendations provided in IAEA Safety Standards Series No. SSG-27, Criticality Safety in the Handling of Fissile Material [3].

1.14. The implementation of safety requirements on the governmental, legal and regulatory framework and related to the 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, 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.

[2] Often called also ‘deconversion’

27-G, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5) [7]. However, this Safety Guide includes recommendations on managing interfaces between safety, nuclear security and the State system for nuclear material accounting and control.

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 deals with safety in the design stage of a nuclear fuel cycle R&D facility: it provides recommendations on the safety analysis for operational states and accident conditions and presents the safety aspects of radioactive waste management in the R&D facility and other design considerations. Section 6 provides recommendations on safety in the construction stage of a nuclear fuel cycle R&D facility. Section 7 provides recommendations on safety in the commissioning stage. Section 8 deals with the safety in the operation of a nuclear fuel cycle R&D facility: it provides recommendations on the management of operation, maintenance and periodic testing, control of modifications, criticality control, radiation protection, 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 gives 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 materials are present in different forms with diverse physical and chemical characteristics. The main hazards are potential nuclear criticality, loss of confinement, radiation exposure (both internal exposure and external exposure), fire, chemical and explosive 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 R&D facility in a safe state during and immediately after an event include the following:

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

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

(a) The radiological consequences caused by the release of radioactive materials under accident conditions can be significant.
(b) Fissile material (if present) has the potential to achieve criticality under certain conditions. The subcriticality of a system depends on many parameters, including the fissile mass, concentration, volume, density, geometry and isotopic composition. Subcriticality is also affected by the presence of other materials, such as neutron absorbers, moderators and reflectors; see SSG-27 [3].
(c) When irradiated fuel is used, the radiation levels and the risk of internal exposure and external exposure are significantly increased.
(d) The chemical toxicity of material used in nuclear fuel cycle R&D facilities has to be considered (e.g. uranium hexafluoride, which if released, reacts with the moisture in the air to form hydrogen fluoride and soluble uranyl fluoride). Therefore, the safety analysis of such an R&D facility should also address impacts resulting from these chemicals and their potential mixing (e.g. in liquid effluent streams).
(e) The presence of products, sub-products or waste arising from R&D programmes on exotic nuclear materials, such as those listed below, which should be included in safety assessments:

    (i) Non-standard mixed oxide or uranium dioxide fuel fabrication, or new fuel matrices, e.g. carbides, nitrides, metallic forms;
    (ii) Isotopes with particular constraints for disposal, e.g. long half-life transuranic isotopes (such as curium), fission products (such as ^{99}Tc) and activated materials such as trace materials in cladding;
    (iii) Materials without an agreed national disposal route, e.g. graphite and aluminium in waste;
    (iv) Uranium with enrichment levels higher than 5%;
    (v) Materials in the thorium fuel cycle that contain high-energy gamma emitters such as ^{232}U.

2.4. Nuclear fuel cycle R&D facilities range from small scale academic research facilities to large nuclear pilot plants. 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 documented management system that integrates the safety, health, environmental, security, quality, human-and-organizational-factors, societal and economic elements of the operating organization is required to be implemented by the operating organization in accordance with Requirement 4 of SSR-4 [1]. The integrated management system should be established early in the lifetime of an R&D facility, to ensure that safety measures are specified, implemented, monitored, audited, documented and periodically reviewed throughout the lifetime of the facility or the duration of the activity.

3.2. Requirements for the management system are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [8]. Associated recommendations are provided in IAEA Safety Standards Series Nos GS-G-3.1, Application of the Management System for Facilities and Activities [9], GS-G-3.5, The Management System for Nuclear Installations [10], GSG-16, Leadership, Management System and Culture for Safety in Radioactive Waste Management [11], and TS-G-1.4, The Management System for the Safe Transport of Radioactive Material [12].

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 [10]. 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 the facility involve different groups and interface with other areas such as those relating to nuclear security and to the State system for nuclear material accounting and control. Activities with such interfaces should be identified in the management system, coordinated, planned and conducted to ensure effective communication and clear assignment of responsibilities. Communications regarding safety and security should ensure that confidentiality of information is maintained. This includes the system of nuclear material accounting and control, for which information security should be coordinated in a manner ensuring that subcriticality is not compromised. Potential conflicts between the transparency of information related to safety matters and protection of the information for security reasons are required to be addressed: see para. 4.10 of GSR Part 2 [10].

3.4. In determining how the requirements of the management system for 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 [8].

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 [8]. This should also include all aspects of criticality safety. Special consideration should be given to all processes covered by the management system associated with handling plutonium, including transition to hot commissioning or assigning 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. In general:

(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 safety strategy 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 provides 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 safety, including criticality safety, rests with the operating organization: see Requirement 2 of SSR-4 [1]. As required by para. 3.1 of GSR Part 2 [8], the senior management of an R&D facility is required to demonstrate leadership for and commitment to safety. In accordance with para. 4.11 of GSR Part 2 [8], the management system for an R&D facility is required to clearly specify the following:

(a) A description of the organizational structure;

(b) Functional responsibilities;
(c) Levels of authority.

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 [10]. The documentation should also cover other management measures, including planning, scheduling and resource allocation (see para. 9.9 of SSR-4 [1]).

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

> "the 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 is required to ensure that safety assessments and analyses are conducted, documented and updated: see Requirement 5 of SSR-4 [1]. Detailed requirements for safety assessment are established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [13]..

3.11. In accordance with para. 4.2 (d) of SSR-4 [1], the operating organization is required to audit all safety related matters on a regular basis. This includes the examination of arrangements for emergency preparedness and response at the R&D facility, such as emergency communications, evacuation routes and signage. Checks should be performed by the nuclear criticality safety staff who performed the safety assessments 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 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 R&D facility: see Requirement 9 of GSR Part 2 [8]. They are also required to ensure that suitable training is conducted: see para. 4.23 of GSR Part 2 [10]. The management of the operating organization should undertake the following:

(a) Prepare and issue specifications and procedures on safety related activities and operations;
(b) Support the performance of safety assessments of modifications;
(c) Having 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.**" In accordance with paras 9.39–9.47 of SSR-4 [1], the operating organization is required to ensure that these personnel receive training and refresher training at suitable intervals, appropriate to their level of responsibility. In particular, personnel involved in activities with fissile material (both uranium and plutonium), radioactive material including waste and 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 [8] states:

> "**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.**"

In accordance with paras 4.33–4.36 of GSR Part 2 [8], the management system for a nuclear fuel cycle R&D facility is required to include arrangements for procurement.

3.15. In accordance with para. 4.16(b) of SSR-4 [1], the operating organization is required to ensure that suppliers of items and resources important to safety have an effective management system. To meet these requirements, the operating organization should conduct audits of the management systems of the suppliers.

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

3.16. 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 parameters at the nuclear fuel cycle R&D facility that are intended to be controlled and the criteria that should be fulfilled.

3.17. The management system of an R&D facility should include management for criticality safety. Further recommendations on the management system for criticality safety are provided in SSG-27 [3].

3.18. Any proposed modification to an existing nuclear fuel cycle R&D facility, or a proposal for introduction of new activities, are required to be assessed for their implications on existing safety measures and appropriately approved prior to implementation: see para. 9.56 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 para. 9.57(h) and 9.59 of SSR-4 [1]. The facility or activity documentation is required to be updated to reflect modifications: see paras 9.57 (f) and (g) of SSR-4 [1]). The 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

3.19. Requirement 13 of GSR Part 2 [8] states:

> "**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.**"

3.20. The audits performed by the operating organization (see para. 3.11), as well as proper control of modifications to facilities and activities (see para. 3.18) are particularly important for ensuring subcriticality. The results of audits are required to be evaluated by the operating organization and corrective actions to be taken where necessary: see para. 4.2(d) of SSR-4 [1].

3.21. 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, 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 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.

3.22. 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 [14].

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

3.23. In accordance with Requirement 5 of SSR-4 [1], the safety of a nuclear fuel cycle R&D facility is required to be assessed in the safety analysis and verified by periodic safety reviews. The operating organization should ensure that these periodic safety reviews of the facility form an integral part of the organization's management system.

3.24. 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 experts in relevant areas including human factors, criticality safety as well as radiation protection. Such experts should be available to the facility at all times during operation.

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 [15] 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 [16].

4.2. The site evaluation process for a nuclear fuel cycle R&D facility will depend on a large number of variables. Since the earliest stage of planning of 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, the relevant hazard evaluated and the design basis for the facility 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 normal operation and accident conditions is required to be evaluated: see Requirement 12 of SSR-1 [17].

4.3. The scope of the site evaluation for a nuclear fuel cycle R&D facility is established in Requirement 3 of SSR-1 [15] and Requirement 11 and paras 5.1–5.14 of SSR-4 [1] and should also reflect the specific hazards listed 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 a part of a larger site for which criteria for site evaluation 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 [17] and the associated recommendations provided in IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [18].

4.5. SSR-1 [15] and section 5 of SSR-4 [1] establish the requirements for site evaluation for new facilities and for existing facilities and the use of 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 and 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 R&D facility:

(a) The appropriate monitoring and systematic evaluation of site characteristics;
(b) The incorporation of periodic, ongoing evaluation of the site parameters for natural processes and phenomena and human induced events in the design basis for the facility;
(c) The identification 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 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 anticipated future changes to site characteristics and of features 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 [15]. Also, in accordance with Requirement 25 and paras 6.1–6.7 of SSR-1 [15], the dispersion in air and water of radioactive material released from the R&D facility are 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 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 (Requirement 75 of SSR-4 [1]). The selection of a site should take into account both safety and security aspects and should be facilitated by experts from both safety and security.

4.8. The site characteristics are required to be reviewed periodically for their adequacy and persistent 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 safety reassessment are required to 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.**”

It is likely that all these safety functions could be applicable to Case 2 R&D facilities (see para. 1.10). This is much less likely for Case 1 facilities. 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, provide the main safety functions above.

5.2. Requirements on the confinement of radioactive material are established in Requirement 35 and paras 6.157–6.159 of SSR-4 [1]. In normal operation, internal exposure should be avoided by design, including 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 [19].

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

5.4. Requirements for protection against external exposure in 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 an 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 should be paid to provisions for maintenance: see Requirements 26 and 65 of SSR-4 [1].

5.5. Requirements on maintaining subcriticality are established in Requirement 38 and paras 6.138–6.156 of SSR-4 [1]. Recommendations on the design of a R&D facility to ensure subcriticality are provided in section 3 of SSG-27 [3].

5.6. The design of nuclear fuel cycle R&D facilities should give consideration to the handling of various types of radioactive material. Owing to the nature of the work done in such facilities, there are often design and engineering provisions for flexibility and adaptation to anticipate future uses, including the dismantling and reconfiguration of parts of the facility. These provisions should be designed to achieve the following:

(a) To enhance safety;
(b) To take into account the potential for ageing and degradation of items important to safety;
(c) To be operated to ensure safety is maintained over the lifetime of the facility;
(d) To not be used for handling new types of radioactive material without a modification proposal or safety assessment.

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 conservative methodology, and for which releases of radioactive material are kept within acceptable limits [1]. All estimates of source terms should include allowance for the ingrowth of radioactive decay products (such as ^{241}Am) over the lifetime of the facility.

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 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 and external hazards selected in the site evaluation phase and associated to the design basis of R&D facilities. These hazards may 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 of SSR-4 [1].

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

Structures, systems and components important to safety

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

> "The design of the nuclear fuel cycle facility…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 of this Safety Guide presents examples of representative safety functions and their associated SSCs.

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

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

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

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 typical openings between different confinement zones (e.g. doors, penetrations).

5.13. 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).

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

5.15. 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.

5.16. 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.

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

5.18. Airborne contamination (from liquids or dispersible solids) is required to be prevented or the level kept as low as reasonably practicable: see Requirement 34 and para. 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 during normal operation 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.

5.19. 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 (ISO) and relevant national requirements.

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

5.21. To reduce risks relating to transfer operations involving radioactive material, the number of transfer operations should be minimized in the design of the facility. To reduce the complexity of transfer operations, 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

5.22. 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 5.12–5.21) and on parameters that contribute to internal exposure (see paras 5.30–5.34) and external exposure (see paras 5.35–5.40).

5.23. 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 minimize exposure during maintenance of nuclear fuel cycle facilities are established in Requirement 26 of SSR-4 [1]. Examples of such provisions in an R&D facility include connection junctions at containment boundaries and easily cleanable surfaces.

5.24. The design of a nuclear fuel cycle facility is required to ensure that the accumulation of radioactive material (e.g. in process equipment, fume hoods, gloveboxes, hot cells, and secondary systems such as ventilation ductwork) 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.

5.25. 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].

5.26. Requirements for the designation of controlled areas and supervised areas are established in paras 3.88–3.92 of GSR Part 3 [19]. 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 requirements for the of areas apply 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.

5.27. Radiation protection in nuclear fuel cycle R&D facilities 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 an R&D facility.

5.28. 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 air (for beta/gamma and alpha activity) for access and evacuation purposes;
(b) Mobile area monitors (for gamma and neutron radiation) and mobile air samplers (for beta/gamma and alpha activity), for evacuation purposes during maintenance;
(c) Personal dosimeters consistent with the type(s) of radiation present in the R&D facility.

5.29. 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 Requirement 25 and paras 6.100–6.104 of SSR-4 [1], and Requirements 14 and 32 of GSR Part 3 [19]).

Protection of personnel against internal exposure

5.30. The static barriers (at least one is required between radioactive material and working areas: see para. 5.12 of this Safety Guide) normally protect personnel from internal exposure and external exposure (see paras 6.123–6.125 of SSR-4 [1]). The design of such barriers should be specified to ensure their integrity and effectiveness and, where appropriate, to facilitate maintenance. Their design specifications should include, for example: 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 achieve a high standard of confinement.

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

5.32. 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]. These should provide an immediate alarm on detection of airborne contamination with a low threshold. The system design and the location of monitoring points should be chosen with account taken of the following factors:

(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 temporary controlled areas, e.g. for maintenance.

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

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

Protection of personnel against external exposure

5.35. The aim of protection against external radiation exposure is to maintain doses below the limits established in schedule III of GSR Part 3 [19], and to optimize protection and safety (see paras 2.7 and 6.6 of SSR-4 [1]) through a combination of source removal, reduction, distance, shielding and administrative controls. Provision of shielding should also be considered in storage areas. Application of the requirement for the optimization of occupational exposure should also take into account maintenance personnel t.

5.36. In areas containing high levels of beta/gamma activity (such as 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, including deposited radionuclides. In areas containing medium or low levels of activity (such as a teaching laboratory), a combination of shielding and administrative controls should be utilized 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 is practical.

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 (such as 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 nuclear criticality at a nuclear fuel cycle R&D facility

5.38. 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 [3].

5.39. Prevention of nuclear criticality is an important topic with various aspects to be considered during the design and operation of an R&D facility. 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. This should be achieved by determining the effective multiplication factor (k_{eff}), which mainly depends on the mass, the geometry, the distribution and the nuclear properties of the fissionable material and all other materials with which it is associated. The calculated value of k_{eff} (including all uncertainties and biases) should then be compared with the value specified by the design limit (which should be set in accordance with paras 2.4–2.7 of SSG-27 [3]) and actions should be taken to maintain the value of k_{eff} under this limit.

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. The 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 materials are 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 mass in an area should be independent of all other factors. A number of such areas may coexist independently in a single facility with suitable interface controls.

5.43. For Case 2 R&D facilities, recommendations provided in facility-specific Safety Guides (IAEA Safety Standards Series Nos SSG-5, Safety of Conversion Facilities and Uranium Enrichment Facilities [20], SSG-6 [5], SSG-7, Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities [21], and SSG-42, Safety of Nuclear Fuel Reprocessing Facilities [22]) should be applied. When the recommendation for a specific pilot facility type are not applicable, the recommendations for the prevention of criticality in SSG-27 [3] should be followed. Some examples of the parameters that should be controlled in nuclear fuel cycle R&D facilities to prevent criticality include the following:

(a) Mass: mass margins[3] should be based on a representative material with the lowest critical mass. The margin should not be less than 100% of the normal value in operation (unless the likelihood of double batching is demonstrated to be sufficiently remote), or a mass margin equal to the physical mass that can be accumulated.
(b) Geometry or shape: The analysis should give consideration to the layout of the facility, the dimensions and locations of pipes, vessels and other laboratory equipment. For example, control by geometry could be used in the design of furnaces and dissolvers.
(c) Density and forms of materials: The analysis should consider the range of densities for different forms of materials (e.g. powder, pellets or rods) used in an R&D facility.
(d) Concentration and density in analytical laboratories and in liquid effluent units: The analysis should consider the range of fissile material in solution as well as any potential precipitates (e.g. recovery of Pu in waste streams).

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

(e) Moderation: The analysis should consider a range of moderation to determine the most reactive conditions that could occur. Water, oil and similar hydrogenous substances are common moderators that are present in R&D facilities, or may be present under accident conditions (e.g. water from firefighting). The possibility of non-homogenous distributions of moderators with fissile material should be considered (e.g. organic binders and porosity enhancing agents used in the pelletizing process).
(f) Moisture content in powders: The analysis should consider the range of moisture content for the powders used in an R&D facility.
(g) Reflection: The most conservative margin of those resulting from different assumptions should be retained, such as: (i) a hypothetical thickness of water around the processing unit; and (ii) consideration of the actual neutron reflection effect due to, for example, the presence of personnel, organic materials, shielding materials, concrete or steel of the containment in or around the processing unit.
(h) Neutron absorbers: If claims are made for neutron absorbers in the safety analysis, their effectiveness should be verified depending on the relevant operating conditions. Neutron absorbers such as cadmium and boron may be used in R&D facilities and the safety analysis should address their effect as neutron absorbers; however, ignoring their effects would still yield conservative results. The use of mobile or easily displaced or removed solid absorbers should be avoided.
(i) Neutron interaction: Consideration should be given to neutron interaction between fissile material in all locations in the R&D facility and all potential locations that may be involved. Specific consideration should be given to the layout of the R&D facility and any possible changes. Physical locators are preferred to floor markings as a means of indicating or ensuring the placement of equipment with potential neutron interactions.
(j) Fissile content: For any fissile material (e.g. fresh or irradiated fuel), the maximum fissile content (e.g. level of enrichment) in any part of the facility should be used in all assessments unless the extreme improbability of having this isotopic composition in a particular area of the facility is demonstrated in accordance with the double contingency principle.

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 should meet Requirement 66 and paras 9.83–9.85 of SSR-4 [1] for criticality control at all times. The design of the R&D facility, including any support systems, should provide the necessary equipment for accounting 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, 6.172–6.173 of SSR-4 [1],. Information regarding the need to install criticality accident alarm systems can be found in Ref. [23]. Where such systems are installed, the R&D facility designed 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[4] should be defined in

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

accordance with the layout of the facility, the process at hand, the national safety regulations and the criticality safety analysis.

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 defence in depth requirements in paras 6.19 – 6.27 of SSR-4 [1]. For example, consideration should be given to the provision of remote mitigation devices, for example, devices to empty a vessel containing the solution initiating the event or to absorb the neutron flux.

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

5.48. In accordance with para. 6.60 of SSR-4 [1], postulated initiating events from the list of internal hazards and external hazards for nuclear fuel cycle R&D facilities are required to be identified for detailed further analysis.

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 and paras 6.43–6.6.48 of SSR-4 [1].

Fire and explosion

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

5.51. In an 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, moderation or the control system.

5.70. An analysis of fire and explosion hazards is required to be conducted for R&D facilities to meet the requirements established in Requirement 22 and paras 6.77–6.79 of SSR-4 [1]. Fire hazard analysis involves the identification of the causes of fires, assessment of the potential consequences of a fire and, where appropriate, estimation of the frequency or probability of occurrence of fires. Fire hazard analysis should identify potential causes of fires, i.e. any fuels or oxidizing agents present. The potential consequences of fires should be assessed with, where appropriate, an estimation of the frequency or probability of the occurrence. The analysis should also assess the inventory of radioactive materials, ignition sources and combustible materials nearby, and should determine the adequacy of measures for fire protection. Computer modelling of fires may sometimes 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.

5.53. An important aspect of the fire hazard analysis for an R&D facility is the identification of areas of the facility that require 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 and resins and reactive chemicals, or involving mechanical treatment of pyrophoric metals or alloys (e.g. cuttings, shavings);
(d) Areas with high fire loads, for example, waste storage areas;
(e) Waste treatment areas, especially if incineration is used;
(f) Rooms housing safety related items, i.e. items such as air filtering systems and electrical switch rooms, whose degradation might have radiological consequences or consequences that are unacceptable in terms of criticality;
(g) Process control rooms and supplementary control rooms, where appropriate;
(h) Evacuation routes.

Fire prevention, detection and mitigation

5.54. 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."

5.55. 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 9.109–9115 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 is stored from process areas.
(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) Compartmentalization of buildings and ventilation ducts as far as possible to prevent spreading of fires. The higher the fire risk, the greater the number of fire compartments a building should have. Utility lines penetrating fire compartment boundaries (e.g. electricity, gas or process lines) should be designed to ensure that fire does not spread.
(f) Suppression or limitation of the number of possible ignition sources such as open flames or electrical sparks, and their segregation from combustible material.
(g) Insulation of hot or heated surfaces.
(h) Placing fire detection systems inside rooms where radioactive material is handled. Provision of detectors inside cells, gloveboxes and ventilation ducts should also be considered.
(i) Consistency of the fire extinguishing media with the requirements of other safety analyses, especially with the requirements for criticality control (see Requirement 38 and para. 6.146 of SSR-4 [1]).
(j) Avoiding 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 to the integrity of the gas supply where inert gas is used as a fire suppressant.

5.56. The design of ventilation systems in a nuclear fuel 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. Such dampers should close automatically on receipt of a signal from the fire detection system, or by means of fusible links. Spark arrestors should be used to protect filters if necessary. The operational performance of the ventilation system should be specified.

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

Explosions

5.58. 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).

5.59. To prevent a release of radioactive material resulting from 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 and abnormal situations (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 situations leading to explosion;
(d) The use of inert atmospheres;
(e) Controlling levels of humidity.
(f) Effective airlocks should be provided between flammable atmospheres and other areas.

Handling errors

5.60. The 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 radioactive or other materials might result in the degradation of criticality controls, confinement, shielding, or in a degradation of defence in depth. A nuclear fuel cycle R&D facility should be designed to:

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

(b) Limit 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) Minimize the failure frequency of mechanical handling systems (e.g. cranes, carts) by appropriate design, including 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 the definition of appropriate administrative controls (see paras 9.36 and 9.37 of SSR-4 [1]).

Equipment failures

5.61. Paragraphs 6.80–6.89 of SSR-4 [1] establish requirements to address equipment failure among the initiating events considered in the design of a nuclear fuel cycle R&D facility. Thus, an 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 structures, systems and components important to safety is required to be assessed and consideration given (in accordance with a graded approach) to the design or procurement of items that fail to a safe state. Where no fail-safe state can be defined, the functionality of structures, systems and components important to safety is required to be maintained (e.g. by redundancy, separation, diversity and independence, as necessary).

5.62. Failure due to fatigue or chemical corrosion or lack of mechanical strength should be considered in the design of containment systems.

5.63. To prevent failure of equipment containing hazardous materials, effective programmes for maintenance, periodic testing and inspection should be established at the design stage (see also paras 5.148 – 5.150).

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

Loss of services

5.65. A nuclear fuel cycle R&D facility should be designed to cope with loss of services that might have consequences for 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 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].

5.66. To meet 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[5]. 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 structures, systems and components important to safety: see para. 6.187 of SSR-4 [1]. For an R&D facility, this includes the following:

(a) Criticality accident detection and alarm systems;
(b) Ventilation fans and monitoring systems for the confinement of radioactive material;

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

(c) Heat removal systems;
(d) Emergency control systems;
(e) Fire detection and alarm systems;
(f) Monitoring systems for radiation protection;
(g) Instrumentation and control associated with the above items;
(h) Adequate lighting (see also para. 6.182 of SSR-4 [1]).

5.67. The loss of general supplies such as gas for actuators of the instrumentation and control, water for process equipment and ventilation systems, heating, breathing air and compressed air might also have consequences for safety. Examples of suitable measures to be addressed in the design of a nuclear fuel cycle R&D facility to ensure safety include the following:

(a) In accordance with the safety assessment, the design of supply systems 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 for all supplies and considered in the design.
(c) For loss of air supply to pneumatically actuated valves, in accordance with the safety analysis, valves should be used that are designed to be fail-safe, as far as practicable.
(d) For loss of water or heating, adequate backup capacity or a redundant supply should be provided;
(e) For 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.

5.68. Consideration should be given to the loss and excess of process media or additives that might have safety consequences. Examples include the following:

(a) The loss or excess of process gas supplies, for example, hydrogen, nitrogen, helium and argon;
(b) Overpressure in gloveboxes that 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.

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

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

Leaks and spills

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

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

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

5.74. Spillage might occur from cans, drums and 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

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

5.76. For areas where fissile material is present, a criticality assessment should be undertaken to determine the risk 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 conditions unless the criticality assessment has taken into account the presence and potential leakage of water.

5.77. 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 analyses 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.

5.78. The potential hydraulic pressure and upthrust on large vessels, ducting and containment structures in the event of flooding should be considered in the design.

Chemical hazards

5.79. The requirements relating to the management of chemical hazards in a nuclear fuel cycle R&D facility are established in Requirement 42 and para. 6.168 of SSR-4 [1]. A number of chemical processes can be affected by radiolysis, potentially generating secondary hazards. Irradiation of organic or hydrated substances by radioactive material can lead to gas generation, especially of hydrogen. 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 enclosing 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

5.80. 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 provisions of SSR-1 [15]. Detailed recommendations on external hazards are provided in IAEA Safety Standards Series Nos SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [24], SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [25], SSG-21, Volcanic Hazards in Site Evaluation of Nuclear Installations [26], SSG-67, Seismic Design for Nuclear Installations [27] and SSG-68, Design of Nuclear Installations Against External Events Excluding Earthquakes [28].

Earthquakes

5.81. 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 [15]) using a graded approach. Recommendations on this assessment are provided in SSG-9 (Rev. 1) [24] and SSG-67 [27]. 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 and flooding);
(e) The effect on criticality safety functions such as geometry, moderation, absorption and reflection of the following:
    (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).

5.82. In accordance with Requirement 14 and para. 6.49 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 to ensure that such an event will not impair the function of control rooms (where provided), 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

5.83. 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, forests, pipelines, and road, rail or sea routes used for the transport of flammable material such as gas or oil, and volcanic hazards.

5.84. To demonstrate that the risks associated with such external hazards are below acceptable levels, the operating organization should first identify all potential sources of hazards and then estimate the associated event sequences that might affect 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. Toxic and asphyxiant hazards should also be assessed 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. The operating organization is required to consider potentially hazardous installations and transport operations for hazardous material in the vicinity of the R&D facility: see paras 5.36 and 5.37 of SSR-1 [17]. In the case of explosions, risks should be assessed for compliance with overpressure criteria. To evaluate the possible effects of flammable liquids, volcanic ashes, falling objects (such as chimneys), air shock waves and missiles resulting from explosions, their possible distance from the facility and hence their potential for causing physical damage should be assessed.

Extreme meteorological phenomena

5.85. An 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 [17]. This should 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 safe shutdown of 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) Keeping the groundwater level within acceptable limits during flooding;
(e) Events consequential to extreme meteorological conditions.

Tornadoes

5.86. Measures for the protection of the 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 specific national regulations do not exist, the design should adhere to international good practices.

5.87. High winds are capable of lifting and propelling large, heavy objects (e.g. automobiles or telegraph poles). The possibility of impacts of such missiles are required to be taken into consideration in the design stage for the facility: see para. 5.14 of SSR-1 [17]. This should include a 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

5.88. The potential duration of extreme low or high temperatures is required to be taken into account in the design: see para. 5.11 of SSR-1 [17]. For a nuclear fuel cycle R&D facility the aim should be to prevent unacceptable effects of the following:

(a) The freezing of cooling circuits (including cooling towers and outdoor actuators);
(b) The loss of efficiency of cooling circuits (hot weather);
(c) Adverse effects on a building’s venting, heating and cooling systems, to avoid poor working conditions and excess humidity in the buildings and adverse effects on structures, systems and components important to safety.

Administrative controls to limit or mitigate the consequences of extreme temperatures should only be relied upon if the operators have the necessary information, sufficient time to respond and the necessary equipment, e.g. portable air-conditioning.

5.89. If limits for humidity and/or temperature are specified in a building or a compartment, the air-conditioning system should be designed to perform efficiently also under extreme hot or wet weather conditions. Structural components of buildings (as static containment) should also be designed for extreme temperature and humidity and its associated thermal stress effects such as shrinkage in concrete.

Snowfall and ice storms

5.90. The occurrence of snowfall and ice storms and their effects are required to be taken into account in the design of the facility and the safety analysis: see paras 5.11 and 5.27 of SSR-1 [17]. Snow and ice are generally taken into account as an additional load on the roofs of buildings. Snow can also block the inlets of ventilation systems and the outlets of drains. The flooding resulting from snow or ice accumulation and infiltration and the possibility that it could damage equipment important to safety (e.g. electrical systems) should be considered. The neutron reflecting effect, or the interspersed moderation effect of the snow should be considered. The effect of ice on wall loadings should also be considered where this is a possibility.

Flooding

5.91. For any flood events such as extreme rainfall (for an inland site) or storm surge (for a coastal site) attention should be focused on structures, systems and components 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 (remain undamaged and static) to the dynamic effects of flooding and all glovebox penetrations should be above any potential flood levels. Electrical systems, instrumentation and control systems, emergency power systems (batteries and power generation systems) and control rooms should be protected by design.

5.92. For 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. 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 (of natural and human induced origin)

5.93. Measures for the protection of the facility against inundation events (dam burst, flash flood, storm surge, tidal wave, seiche, tsunami), including both static effects (floods) and dynamic effects (run-up and draw-down), will depend on the data collected during site evaluation for the area in which the reprocessing facility is located. The design of buildings, electrical systems and instrumentation and control systems should comply with specific national regulations for these hazards, including the recommendations provided in paras 5.91 and 5.92 of this Safety Guide. Particular attention should be given to the rapid onset of these events, the probable lack of warning and their potential for causing widespread damage, disruption of utility supplies and common cause failures both within the reprocessing facility and at other facilities on the site, locally and potentially regionally, depending on the magnitude of the event.

Accidental aircraft crash hazards

5.94. In accordance with the risk identified in the site evaluation (see Section 4), the R&D facility is required to be designed to withstand the design basis impact: see para. 5.7(e) of SSR-4 [1] and para. 5.35 of SSR-1 [17].

5.95. For evaluating the consequences of impacts or the adequacy of the design to resist aircraft or secondary missile 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 nuclear fuel cycle R&D facility. Such scenarios require knowledge of such factors as the possible angle of impact, velocity or the potential for fire and explosion due to the aviation fuel load. In general, fire cannot be ruled out following an aircraft crash. Therefore, specific requirements for fire protection and for emergency preparedness and response should be established and implemented as necessary.

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

5.96. 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. Monitoring and control is 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 proper actions can be undertaken in accordance with operating procedures, emergency procedures or accident management guidelines, as appropriate, for all facility states.

5.97. Instrumentation and control systems are required to be provided for criticality safety, and for hot cells, gloveboxes and hoods: see paras 6.172–6.174 of SSR-4 [1].

5.98. 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. 6.169 and 6.170 of SSR-4 [1].

5.99. Appropriate information should be made available to operating personnel for monitoring the effects of automatic actions. The layout of instrumentation and the manner of presentation of information should provide the 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 and that could affect safety.

5.100. Control systems should be provided to ensure compliance with regulatory limits, for example, on discharges (see para. 5.101). Where appropriate, provision should be made for the automatic measurement and recording of parameters that are important to safety, and manual periodic testing should be used to complement automated continuous testing of conditions.

Instrumentation and control systems at a nuclear fuel cycle R&D facility

5.101. Safety related instrumentation and control systems for a nuclear fuel cycle R&D facility include 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, 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 and/or toxic chemical material should be equipped with provisions for fire detection and fire extinguishing;
    (ii) Gas detectors should be used in areas where a leakage of gases (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:
    (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 to ensure the early detection of any dispersion of airborne radioactive material;
    (iv) Devices for detecting surface contamination, installed or located close to relevant working areas and also close to the exits from these areas.
(f) Control of liquid discharges and gaseous effluents:
    (i) Systems to monitor and control liquid discharges from nuclear fuel cycle R&D facilities. 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 continuous monitoring of discharges.
(g) Monitoring and control of airflows and air quality:
    (i) Systems to ensure that the airflows in all areas of the nuclear fuel cycle R&D facility are flowing in the correct directions, i.e. from less contaminated to more contaminated areas.
    (ii) In work areas, the temperature, humidity and pollutants should be controlled to ensure worker comfort and hygiene.
    (iii) In some cases, local ventilation should be used, for example, in rooms housing backup batteries.

Control rooms

5.102. 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 Case 2 nuclear fuel cycle R&D facilities, 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 together and monitored. Such controls should be located in parts of the R&D facility where risks to operating personnel can be minimized. Particular consideration should be given to identifying events, both internal and external to the control rooms, that might pose a direct threat to the control room operators and to the operation of control rooms. Ergonomic principles are required to be applied in the design of the control rooms and the design of control room displays and systems: see para. 6.108 of SSR-4 [1].

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

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

5.104. 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 to 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.

5.105. All 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 and accident conditions should be identified. The aim should be to facilitate the necessary actions of operating personnel activities and ensure that safety functions and the structures, systems and components 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.

5.106. Experts in human factors engineering and experienced operating personnel should be involved from the earliest stages of design. Areas that should be considered 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 and no more;
    (ii) The working environment, e.g. good accessibility to, and adequate space around, equipment, good lighting, including emergency lighting, and suitable finishes to surfaces to 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) Choice of location and 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) Task design and job organization, particularly during maintenance work, when automated control systems may be disabled.
(e) Minimization of the need to use personal protective equipment.
(f) Operational experience feedback relevant to human factors.

5.107. 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, account should be taken of the potential for conventional industrial hazards 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 changing. Sharp edges and corners on equipment and fittings and associated tools should be avoided to minimize risks of glove damage.
(d) Training of operators on procedures to be followed for normal and abnormal conditions (see para. 9.48 of SSR-4 [1]).

SAFETY ANALYSIS FOR A NUCLEAR FUEL CYCLE R&D FACILITY

5.108. Requirement 14 of GSR Part 4 (Rev. 1) [13] states:

> "**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 include the analysis of the variety of hazards for the whole facility (see Section 2) and all the activities performed within the facility.

5.109. 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 structures, systems and components important to safety that might be affected by the postulated initiating events identified: see para. 4.20 of GSR Part 4 (Rev. 1) [15].

5.110. For nuclear fuel cycle R&D facilities, the safety analysis should be performed iteratively with the development of the design with the objectives of achieving the following:

(a) That doses to workers and the public during operational states do not exceed dose limits and are as low as reasonably practicable, in accordance with Requirement 9 of SSR-4 [1];
(b) That the 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) The development of appropriate operational limits and conditions.

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

5.111. 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 cumulative annual working time at each workplace for both normal work activities and maintenance;
(c) Conservative assumptions about the efficiency of shielding.

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

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

5.114. The calculated doses should be compared with actual doses 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.

5.115. The calculation of dose to the public should include all the exposure routes originating from the facility, i.e. external exposure through direct or indirect radiation, and internal exposure through intakes of radioactive material (e.g. received 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 [29].

5.116. 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

5.117. The acceptance criteria associated with the safety analysis for accident conditions should be defined in accordance with Requirement 16 of GSR Part 4 (Rev. 1) [13], and with respect to any national regulations.

5.118. To estimate the on-site and off-site consequences of an accident, the wide range of physical processes that could lead to a release of radioactive material to the environment need to be considered and bounding cases[6] encompassing the worst consequences should be determined.

5.119. 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 and hydrogeological site conditions) and 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.

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

(d) Identification and analysis 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, in accordance with reasonable scenarios.
(e) Quantification of the consequences for site personnel and the representative person(s) identified in the safety assessment.
(f) Specification of the structures, systems and components important to safety that may be credited to reduce the likelihood of, and/or to mitigate the consequences of accidents. These structures, systems and components that are credited in the safety assessment and are required to be qualified to perform their functions reliably in accident conditions: see paras 4.30 and 4.36 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.

5.120. 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 transferring of energy that might be released from the facility.

5.121. 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.

5.122. Further recommendations on the assessment of potential radiological impact to the public are provided in GSG-10 [29]. Guidelines for assessing the acute and chronic toxic effects of chemicals used in R&D facilities are provided Ref. [30].

Analysis of design extension conditions

5.123. The safety analysis 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."

5.124. 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 structures, systems, and components important to safety or that could challenge the fulfilment of the main safety functions. The list of postulated initiating events provided in Appendix of SSR-4 [1], including combinations of these events, should be used as well as events with additional failures.

5.125. 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.

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

5.127. Examples of design extension conditions that are applicable to nuclear fuel cycle R&D facilities are listed in Ref. [31].

5.128. Analysis of design extension conditions should also demonstrate that the R&D facility can be brought into the state where the confinement function and sub-criticality 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 [32], GSG-1, Classification of Radioactive Waste [33], SSG-41, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities [34] and GSG-16 [11].

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 the 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 [3]. Examples of such waste at nuclear fuel cycle R&D facilities 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 to ensure the integrity of the facility and the waste containers, taking into account low probability events, should be taken, 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 can be 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 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 (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 should be made to allow the efficiency of these systems to be monitored.

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 dose rates (neutron, beta/gamma) depending on the isotopic composition of the material processed. Therefore, consideration should be given at the design stage to the need for shielding for both neutron and gamma radiation.

5.137. Effective gamma and neutron 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) Prevention of the spread of contamination when maintaining or replacing equipment (e.g. motors and drives can be located outside gloveboxes).
(d) The design should aid 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. Waste accumulation should be avoided.
(e) Removal of shielding material. Shielding on gloveboxes is often provided for normal process operations and may need to be removed for maintenance access. Consideration should be given to removing radioactive material before removing any shielding.

(f) The facility design should minimize sharp edges and the need for sharp equipment in gloveboxes to minimize the potential to cause wounds that could become contaminated.
(g) The design of replaceable parts should facilitate segregation and handling of mixed and hazardous waste.
(h) Surveillance and monitoring requirements for ageing and degradation.

Decontamination and dismantling

5.141. Floor, wall and ceiling surfaces in a nuclear fuel cycle R&D facility, particularly in wet chemical areas, are required to be selected 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 include the application of coverings or coatings to such 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 eventual decontamination (e.g. by stripping of paint or coatings).

EMERGENCY PREPAREDNESS AND RESPONSE

5.142. The Government is required to ensure that a hazard assessment is performed in accordance with Requirement 4 of GSR Part 7 [17]. 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 [18].

5.143. 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]. 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 potential hazards assessed at 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, but are not limited to, criticality accidents and nuclear or radiological emergencies coincident with external hazards affecting the infrastructure in the vicinity of the R&D facility (e.g. widespread fires, earthquakes and tsunamis).

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

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

5.146. For Case 2 R&D facilities, the hazards listed in the IAEA Safety Guides related to the corresponding type of nuclear fuel cycle facilities, for example in SSG-5 [20], SSG-6 [5], SSG-7 [21] and SSG-42 [22], 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 alternative arrangements, e.g. emergency control panels, should be provided for the safety functions identified by this analysis.

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 [17]).

AGEING MANAGEMENT AT A NUCLEAR FUEL CYCLE R&D FACILITY

5.149. The design of a nuclear fuel cycle facility is required take into account the effects of ageing on systems, structures and components 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 systems, structures and components important to safety. This should include the detection of the effects of ageing (static containment deterioration, corrosion) and allow the maintenance or replacement of such items, if needed.

5.151. An ageing management programme is required to by the operating organization: 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 construction of a nuclear fuel cycle R&D 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 [35].

6.2. For a complex nuclear fuel cycle R&D facility (e.g. a Case 2 facility), regulatory authorization should be sought in several stages. Each stage may have a hold point at which approval by the regulatory body may be necessary before the subsequent stage may be commenced, as described in para. 7.2 of SSR-4 [1]. Frequent visits by the regulatory body to the construction site should be used to provide feedback of information to the construction contractor to prevent future operational problems.

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 sub-standard 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 in the construction of nuclear fuel cycle R&D facilities used for fundamental research (i.e. Case 1 facilities). This enables equipment to be tested and proven at the manufacturer's premises before installation in the R&D 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) Construction personnel receiving unnecessary exposure 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.

Preventative 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:

(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.

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 8.1–8.23 and 8.27 of SSR-4 [1].

7.2. Paragraph 8.12 of SSR-4 [1] requires the commissioning phase to be divided into stages; this requirement is also applicable to Case 1 and Case 2 nuclear fuel cycle R&D facilities. For such facilities, this typically involves three stages, which are described below.

STAGE 1: COLD COMMISSIONING (‘INACTIVE COMMISSIONING’)

7.3. At this stage, the facility’s systems are tested in the absence of radioactive material. The facility is tested systematically, as individual items of equipment and as systems in their entirety (see para. 8.9 of SSR-4 [1]). As it is relatively easy to take corrective actions at this point, as much verification and testing as possible should be performed in this stage.

7.4. In this stage, operating personnel should take the opportunity to further develop and finalize the operational documentation and to learn the details of the systems. Such operational documentation should include 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. Leaktightness and the stability of control systems are best tested at this stage.

STAGE 2: WARM COMMISSIONING (‘TRACE ACTIVE COMMISSIONING’)

7.5. As appropriate, natural or depleted uranium should be used[7] in this stage, 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.

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

7.7. Prior to hot commissioning, the emergency arrangements (on-site and off-site, as appropriate) need to be established, including procedures, training, sufficient numbers of trained personnel, emergency drills and exercises.

STAGE 3: HOT COMMISSIONING (‘ACTIVE COMMISSIONING’ OR ‘HOT PROCESSING COMMISSIONING’)

7.8. This 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.

7.9. The licence to operate the nuclear fuel cycle R&D facility is generally issued by the regulatory body to the operating organization just before this stage. The regulatory body should define hold points and witness points commensurate with the complexity and potential hazard of the facility, to ensure proper inspection during commissioning. The purpose of these hold points should be principally to verify compliance with regulatory requirements and authorization conditions.

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

7.11. The safety committee of the 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 hazard associated with a nuclear fuel cycle R&D facility described in Section 2 should be taken into account in meeting the safety requirements for operation established in section 9 of SSR-4 [1].

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

8.2. Safety should be coordinated between the operational functions and the research functions of the nuclear fuel cycle R&D facility. The safety committee should provide an interface between operations and research; however, this should not be used as a substitute for everyday communication and cooperation on safety between these functions, which should also be documented. Responsibilities that should be coordinated carefully include the management of radioactive material, the monitoring of experiments and the management of radioactive waste. The safety committee of the R&D facility should include representatives of operations, safety and research functions.

8.3. Research programmes should comply with the existing safety case or be considered as a modification. Research involves flexibility in the materials and processes used and the safety case should 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 related 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 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 [9].

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 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.

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 R&D facility personnel operating gloveboxes and hot cells (see also para. 9.48 of SSR-4 [1]), including the actions to be taken in response to anticipated operational occurrences (e.g. a punctured glove in a glovebox or a loss of ventilation in a hot cell).

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

8.10. Requirement 57 and paras 9.27–9.37 of SSR-4 [1] establish requirements for operational limits and conditions be developed for a nuclear fuel cycle facility. Operating personnel 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 structures, systems and components relevant to defining operational limits and conditions for each process area are presented in Annex III.

8.11. In order to ensure that under normal circumstances, the R&D facility operates well within its operational limits and conditions, a set of limits on operating parameters are required to be defined by the operating organization (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 maximize the safety margin while minimizing breaches of the sub-limits.

8.12. The authority to make operating decisions should be assigned to suitable levels of management, depending on the operational limits and conditions, the operational sub-limits and the potential safety implications of the decision. The management system should specify the authority and responsibilities at each management level. If a sub-limit or 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 shift staff or day staff should be trained and authorized to make the necessary decisions, and take the necessary actions, in accordance with these procedures.

8.13. Any non-compliance with operational parameters should be adequately investigated by the operating organization and the lessons learned should be applied to prevent a recurrence. As required by national regulations, the regulatory body should be notified in a timely manner of such non-compliances and any immediate actions taken and should be kept informed of the subsequent investigations and their outcome.

8.14. Operating procedures should be prepared that list all the operational limits and conditions for the nuclear fuel cycle R&D facility. Annex IV gives examples of operational limits and conditions applicable to Case 1 facilities and Case 2 facilities.

8.15. Limits 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) Specified limits on concentrations, geometry and moderators in solutions containing fissile materials;
(c) Specified inventory limits of radioactive material and isotopic compositions in gloveboxes or interim storage areas;
(d) Maximum heat loads specified for locations such as hot cells or gloveboxes;
(e) Maximum quantities of additives at different steps in R&D facility processes;
(f) Specified limits on combustible material in gloveboxes and hot cells;
(g) Specified limits for flammable atmospheres in enclosed equipment, for example, for hydrogen in a furnace.

8.16. 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: se 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 (operating experience feedback).

8.17. The operating organization should establish operating procedures to ensure safety during limited operation of the R&D facility, especially where this is followed by a long period of shutdown. Training programmes should reflect such procedures.

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

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

8.20. 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, conditions and other key safety information, is known, recorded and accessible at all times. Operating procedures should also be established for the ventilation system in fire conditions.

8.21. The management of the 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 R&D facility should participate in such meetings.

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

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

8.23. When carrying out maintenance in an 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 R&D facility should not be placed in an unsafe or unanalysed condition in order to perform periodic testing or routine maintenance.

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

(a) The development of s suitable maintenance programme that includes all processes used in work control, for example, handover and handing back of approved documents, means of communication and visits to job sites, changes to the planned scope of work, suspension of work and ensuring safe access.
(b) Equipment isolation, for example, de-energizing and disconnecting electrical cabling, hot or pressurized piping, and draining, venting and purging of equipment.
(c) Testing and monitoring, for example, checks of workplace and tools before commencing work, monitoring during maintenance and checks for re-commissioning, and communications.
(d) Safety precautions for the work, for example, specifications ensuring the availability and use of personal protective equipment.
(e) Continued monitoring systems for control of criticality and radiation protection.

(f) Reinstallation of equipment, for example, reassembly, reconnection of pipes and cables, testing, cleaning the job site and monitoring should be performed after maintenance and before re-commissioning.

8.25. A programme of periodic inspections of the nuclear fuel cycle R&D facility is required to be established and implemented: see Requirement 65 of SSR-4 [1]. As 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. There should be routine programmes of inspection and maintenance designed to prevent the spread of contamination or a release of hazardous material. These programmes should include, for example:

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

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

8.27. Regular verification of the availability of materials necessary for maintenance should be conducted. For continuity of safe operations of a nuclear fuel cycle R&D facility, a programme for the provision of spare parts for items important to safety, including radiation monitoring equipment, should be established and implemented.

8.28. 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 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 such externally supplied or located equipment is properly calibrated at all times. Where necessary, traceability to national or international standards should be provided.

8.29. 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.

AGEING MANAGEMENT FOR NUCLEAR FUEL CYCLE R&D FACILITIES

8.30. 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 an R&D facility should take into account 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 structures, systems and components ageing, rather than a reactive approach responding to the failure of structures, systems and components;
(d) Ensuring optimal operation of structures, systems and components 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, sense of ownership and awareness, and understanding of the basic concepts of ageing management;
(g) Ensuring availability and use of correct operating procedures, tools and materials, and of a sufficient number of qualified personnel for a given task;
(h) Collecting feedback of operating experience to learn from relevant ageing related events.

8.31. The aging management programme should also consider the non-technical aspects of ageing.

8.32. 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 the operating personnel, such as the following:

(a) Monitoring of deterioration;
(b) Regular visual inspections of structures, systems and components 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 R&D FACILITY

8.33. R&D facilities are normally established in such a way that they can be utilized for a variety of different R&D programmes. It may nevertheless be necessary to modify the facility and its safety case if a new programme of work or item of equipment not covered by the existing authorization is to be implemented or installed. Where this involves a large increase in the scale of operations, the operating organization should plan the increase in stages where possible, in order to permit the gathering of feedback and the validation of each stage.

8.34. 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 an 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.

8.35. In accordance with the safety significance of the modification, and in accordance with regulatory requirements, modifications should be assessed by the operating organization and then submitted to the regulatory body for authorization (or, if appropriate, by registration: see para. 3.8 of GSR Part 3 [19]) before the modifications are implemented. The reassessment of the safety of the facility and the formal authorization by the regulatory body, as required by para. 3.10 of SSR-4 [1], should consider, in particular, the need to assess human factors, e.g. the human–machine interface, alarm systems, procedures and the qualification or requalification of personnel.

8.36. The operating organization is required to prepare procedures and provide training to ensure that the relevant personnel have the necessary competence and authority to ensure that modification projects are carefully controlled: see paras 9.57(e) and 9.58 of SSR-4 [1]. The safety of modifications should be assessed for potential hazards during installation, commissioning and operation.

8.37. Proposed modifications 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 considered particularly important if the modification could have an effect on criticality safety.

8.38. The depth of the safety arguments 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].

8.39. The safety committee is required to review the proposed modifications: see para. 4.31(d) of SSR-4 [1]. Suitable records should be kept of their decisions and recommendations.

8.40. The modification should also specify which documentation and training will need to be updated because of the modification (e.g. training plans, specifications, safety assessment, notes, drawings, engineering flow diagrams, process instrumentation diagrams and operating procedures). Procedures for the control of documentation are required to be implemented to ensure that relevant documents are updated to reflect the planned modification: see para. 9.57 of SSR-4 [1]. Personnel involved in making the modification are required to be suitably trained and qualified: see para. 9.57(f) of SSR-4 [1].

8.41. 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 to ensure 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 may be declared fully operational again.

8.42. Modifications of the design, layout or procedures of a nuclear fuel cycle R&D facility might adversely affect nuclear security. Therefore, in addition to a review of the implications for safety, the possible effects on nuclear security are required to be evaluated before approval and implementation of the modification to verify that safety measures and security measures do not compromise each other: see Requirement 75 of SSR-4 [1].

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

8.44. The modification control documentation (see para. 9.57(f) of SSR-4 [1]) should be retained at the nuclear fuel cycle R&D facility in accordance with regulatory requirements.

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

8.45. Requirements for criticality safety in the operation of a nuclear fuel cycle R&D facility are established in Requirement 66 and paras 9.83–9.85 and 9.89 of SSR-4 [1]. Recommendations on criticality safety in all facilities and activities are provided in SSG-27 [3].

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

(a) Unexpected changes in conditions that could increase the risk of a criticality accident, for example, unplanned accumulation of fissile material (e.g. in gloveboxes or ventilation ducts) or hydrogenated materials;
(b) Unexpected accumulation of water due, for example, to fire suppression sprays or leaks from water pipes;
(c) Management of moderating materials, particularly hydrogenated materials such as those used for decontamination of gloveboxes and leakages of oils from gear boxes;

(d) Management of the transfer of fissile material (procedures, mass measurement, systems and records) where mass control is used;
(e) Reliable methods for detecting the onset of unsafe conditions with respect to criticality control;
(f) Emergency drills and/or exercises (see paras 8.83–8.88);
(g) Periodic calibration or testing of criticality control and monitoring systems (e.g. material movement control, balances and scales).

8.47. 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 is any uncertainty about the characteristics of fissile material, conservative values are required to be used for parameters such as fissile material content and isotopic composition: see para. 7.52 of SSR-4 [1]. This is especially important when managing cell floor or glovebox sweepings and similar waste material.

8.48. Additional criticality safety measures may be necessary for activities such as maintenance work. Fissile material, including waste and residues arising from experiments or pilot processes, decontamination, and maintenance activities is required to be accumulated 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

8.49. The requirements for radiation protection in 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 Part 3 of GSR Part 3 [19]; recommendations on the implementation of GSR Part 3 [19] requirements for the protection of workers are provided in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [36].

8.50. The operating organization of a nuclear fuel cycle R&D facility should have a policy to optimize protection and safety with a systematic approach, and is required to ensure 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].

8.51. In a nuclear fuel cycle R&D facility, the exposure pathways (for both workers and members of the public) include intakes (inhalation or ingestion of particulates, aerosols and gases) and external exposure. Paragraphs 9.38–9.43 of SSR-4 [1] require the establishment of an appropriate radiation protection programme to fulfil the operating organisation’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.

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

8.53. Radiation protection personnel (i.e. radiation protection manager, radiation protection officer and their representatives) should be part of the decision-making process associated with the optimization of protection and safety (e.g. for the early detection and mitigation of hot spots) and proper housekeeping (e.g. waste segregation, packaging and removal).

8.54. Intrusive maintenance 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 the dose, including:
    (i) Identification of specific risks associated with the activities;
    (ii) The use of additional shielding, remote devices or mock-ups, as appropriate;
    (iii) Definition of specific procedures within the work permit (e.g. the use of respiratory protective equipment, protective clothing, and time limitations).
(c) Measurement of the doses received during the activities.
(d) Implementation of feedback to identify possible improvements.

8.55. During 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 extent of workplace monitoring should be sufficient to achieve low levels of airborne activity and contamination in the facility, taking into account the 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, as described in para. 5.26 of this Safety Guide. In addition, to further identify the risk involved in a task, facility areas should be classified into radiation and contamination zones. The boundaries between such zones should be regularly checked and adjusted to match current conditions.
(d) Access to areas designated as controlled areas due to the presence of contamination should be avoided by R&D facility personnel with skin wounds.
(e) Radiation and contamination zones should be demarcated with appropriate warning signs.
    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 radiological protection personnel. Personal protective equipment should be maintained in good condition and be regularly inspected.
(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 and rebalanced if necessary, 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-radiation-related hazards (e.g. oxygen deficiency, heat stress) with particular attention paid to the risks and benefits of the use of personnel protective equipment, especially for air-fed systems.
(k) Personnel and equipment should be checked for contamination and should be decontaminated, if necessary, prior to crossing boundaries between contamination zones.

8.56. 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.

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

8.58. On completion of maintenance work, areas should be decontaminated and air sample and smear 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.

8.59. There should be careful preparation before entry into hot cells or gloveboxes that 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 operations necessitate appropriate authorizations, depending on local rules (see para. 3.94 of GSR Part 3 [19]) and industrial safety requirements for confined space entries.

8.60. Periodic estimates of the impact on the public (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.

8.61. 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.

8.62. 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) Performance of routine 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 occupation of controlled areas and limiting the working time near radiation sources, and maximising 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. by the use of finger dosimeters).

8.65. Performance standards set for air purification systems should specify performance levels at which filters or scrubber media should be changed. After filter changes, tests should be 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 the assessment of occupational exposure is necessary (see Requirement 25 of GSR Part 3 [19]), this should be based on individual dosimeters, as described in paras 5.28(c) and 5.101(e)(i) and 8.64 of this Safety Guide. The assessment of internal exposures, where necessary, may be based on the collection of air sampling data. Where necessary, in vivo (whole body) monitoring and biological sampling (for example, nose blows, faecal and urine samples) should also be available (for routine monitoring and/or accident conditions, as appropriate) as complementary measures to monitor internal exposure.

8.68. Further recommendations on occupational radiation protection and the assessment of internal exposure and external exposure are provided in GSG-7 [36].

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-radiation-related hazards that may 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 organics, pyrophoric metals, hydrogen, ammonium nitrate and 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 do not exceed the threshold at which containment would be lost, as determined by the safety analysis. A fire hazards analysis should be conducted at periodic intervals to incorporate changes that might affect the likelihood of a fire. Computer modelling may be used to support the fire hazards analysis.

8.72. Personnel should be informed about the chemical hazards that exist. 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]) in order to adequately identify and respond to the problems that might lead to chemical accidents.

8.73. As required by national regulations, a health surveillance programme should be set up for routinely monitoring the health of nuclear fuel cycle R&D facility personnel who might be exposed to harmful chemicals. The surveillance programme should address short term effects (acute exposure) and long term effects (chronic exposure).

8.74. The exposure of personnel to chemical hazards should be assessed using a method similar to that for the assessment of radiation exposure and should be based upon 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 appropriate regulatory authority. The acceptance levels of exposure for various chemical hazards can be found in Ref. [30].

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

8.75. The requirements relating to the management of radioactive waste and effluents in operation are established in Requirement 68 and paras 9.102–9.108 of SSR-4 [1].

8.76. All operating personnel should be trained in the waste management hierarchy (eliminate, reduce, reuse, recycle and dispose: see para. 6.17 of SSR-4 [1]), 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 (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 a consideration 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 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 by the reuse, recycling and decontamination of materials). Trends in the generation of radioactive waste 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 an R&D facility is required to be characterized: see paras 6.94 and 9.103 of SSR-4 [1]. This should include a determination of its physical, chemical and radiological properties to allow its subsequent optimum 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 existing temporary storage or disposal routes, as appropriate. Particular care should be taken to segregate waste containing fissile material and 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 in order to avoid uncontrolled or unexpected reactions.

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

8.82. Before the clearance of equipment for recycling or for disposal, it should be decontaminated to the level required by the regulatory body. Criteria for clearance of material from facilities are set out in Schedule I of GSR Part 3 [19].

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

8.83. General requirements for emergency preparedness and response are established in GSR Part 7 [17], and supporting recommendations on emergency arrangements are provided in GS-G-2.1 [18] and in IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [37]. 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].

8.84. The emergency arrangements established at a nuclear fuel cycle R&D facility should consider the layout of the site (i.e. the site may be composed of a large number of buildings and facilities).

8.85. As part of emergency preparedness, arrangements are required to be developed for the local, regional and national emergency response organizations: see para. 3.1 and Requirement 22 of GSR Part 7 [19]. 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 [17] and para. 9.130 of SSR-4 [1].

8.86. Clear communication protocols are required to be established with local authorities and response organizations: see para. 5.43 of GSR Part 7 [19].

8.87. The emergency arrangements 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.

8.88. For establishing access control procedures during emergencies, when there is a necessity for rapid access and egress of personnel, safety and security specialists should cooperate closely. Both safety and security objectives should be sought for during emergencies as much as possible, in accordance with regulatory requirements.

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

8.89. Requirements on feedback of operating experience are established in Requirement 73 and paras 9.133–9.137 of SSR-4 [1]. Further recommendations on a programme for operating experience feedback are provided in SSG-50 [14].

8.90. The programme for the feedback of operational experience at a nuclear fuel cycle R&D facility should cover experience and lessons learnt 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 R&D facility and, as far as applicable, at other nuclear installations. The programme is required to include a consideration of technical, organizational and human factors: see para. 9.134 of SSR-4 [1].

8.91. Useful information on the causes and consequences of many of the most important anomalies and accidents that have been observed in R&D facilities and other nuclear fuel cycle facilities is provided in Ref. [38].

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 [39]. Requirements for the preparation for decommissioning of a nuclear fuel cycle R&D facility are established in Requirement 74 and paras 10.1–10.13 of SSR-4 [1].

9.2. The developed decommissioning plan and the safety assessment are required to be periodically reviewed and updated throughout the lifetime of the R&D facility: see paras 7.5 and 7.6 of GSR Part 6 [39] and paras 10.1 and 10.2 of SSR-4 [1]. This review should take into account new information and emerging technologies to ensure that:

(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 radioactive waste anticipated 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) Design measures to prevent contamination from penetrating structural materials, such as pond liners;
(b) Engineered controls and administrative controls to prevent the spread of contamination;
(c) Consideration of the implications for decommissioning resulting from modifications and experiments in the facility, when they are proposed;
(d) Identification of reasonably achievable changes to the facility design to facilitate or accelerate decommissioning;
(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 operating organization is required to ensure adequate financial resources for safe decommissioning: see para. 4.2(e) of SSR-4 [1].

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

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

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

(a) Preparation of risk assessments and method statements for the licensing of the decommissioning process.
(b) Post-operational clean-out to remove all bulk quantities of radioactive material and other hazardous materials;
(c) Identification of contaminated parts of buildings and equipment, and radionuclides;
(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, or the lowest reasonably achievable level of residual contamination.

9.7. For any period between a planned or unplanned shutdown and prior to decommissioning starting, safety measures are required to be implemented to maintain the nuclear fuel cycle R&D facility in a safe and stable state, including measures to prevent criticality, spread of contamination and fire, and to maintain appropriate radiological monitoring: see para. 10.9 of SSR-4 [1]. The need to revise the safety assessment for the facility in its shutdown state is also required to 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 R&D facility is placed into a prolonged shutdown state.

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Annex I

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

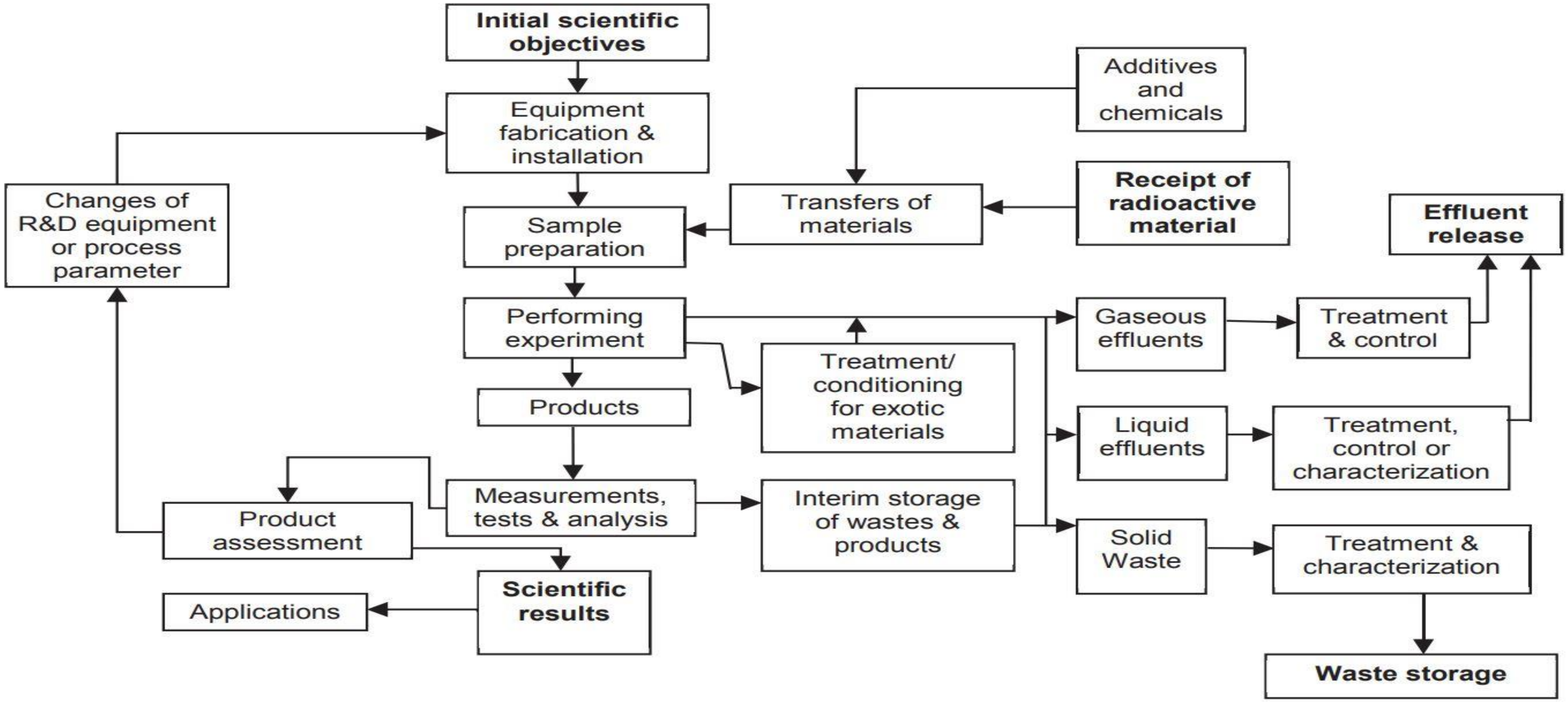

FIG. I-1. Diagram showing the general processes in an R&D facility operating at laboratory scale (Case 1)

Annex II

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

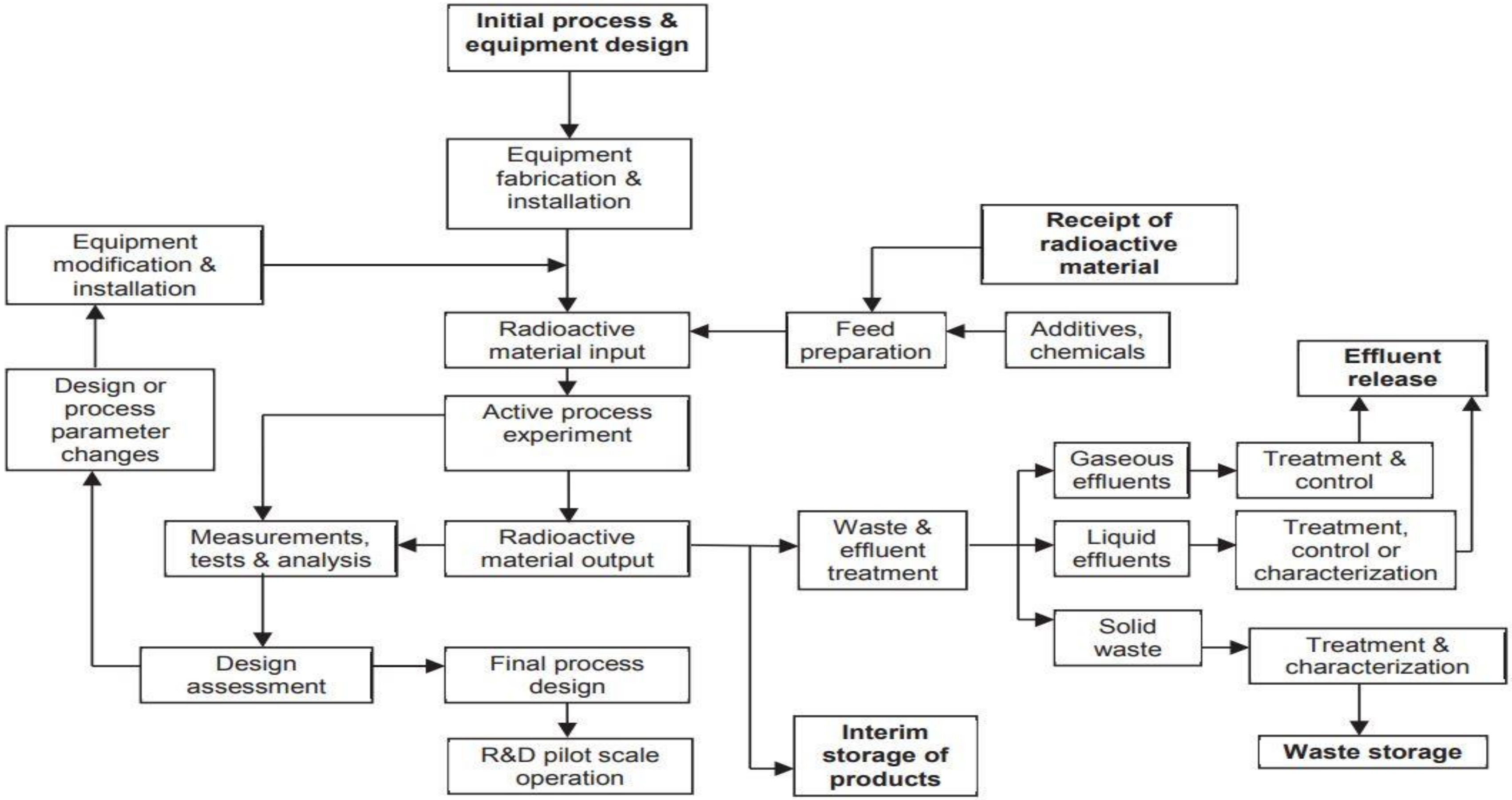

FIG. I-2. Diagram showing the general processes in an R&D facility operating at a pilot scale (Case 2)

Annex III

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

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

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

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Initial scientific objectives			1, 2 and 3	Application of Safety Principles Nos 4–9[8] Safety assessment of programmes and activities

[8] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards SeriesNo. SF-1, IAEA, Vienna (2006).

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

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

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

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

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

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

[a] Rules for the safe transport of radioactive materials and samples at the facility are defined by the operator or IAEA safety standards for transport[9] may be applied in a graded manner.

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

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

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

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

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

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

Process area	Structures, systems and components important to safety	Events	Safety function initially challenged	Operational limits and conditions, other means of mitigation and comments
Performance of experiments	Calibrated equipment Diverse equipment ensuring mass, geometry, moderation control Reflectors Neutron absorbers Detection and alarm systems	Non-acceptable k_{eff} Double batching Inadvertent accumulation of fissile material	1	Operational limits and conditions where necessary Independent double check by suitably qualified and experienced persons especially for mass and concentration of fissile materials Stringent implementation of quality assurance including maintenance and periodic inspection, e.g. of reflectors Questioning attitude
	Fume hoods, hot cells or gloveboxes Pressure monitoring/ recording	Breach of containment	2	Effective isolation procedures Maintenance and periodic testing
	Emergency power supply	Loss of power	3	System dependent procedures e.g. for low battery voltage Maintenance and periodic testing

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

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

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

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

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

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

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

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

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

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

Annex IV

EXAMPLES OF OPERATIONAL LIMITS AND CONDITIONS

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

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