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Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors

DS 509F

DRAFT SAFETY GUIDE

A revision of NS-G-4.6

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Requirements for the safety of research reactors, with particular emphasis on their design and operation, are established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [1]. Requirements on radiation protection are established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [2]. Requirements on radioactive waste management are established in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [3].

1.2. This Safety Guide provides recommendations on the radiation protection and radioactive waste management in the design and operation of research reactors.

1.3. This Safety Guide was developed in parallel with seven other Safety Guides on the safety of research reactors, as follows:

- IAEA Safety Standards Series No. DS509A, Commissioning of Research Reactors [4];
- IAEA Safety Standards Series No. DS509B, Maintenance, Periodic Testing and Inspection of Research Reactors [5];
- IAEA Safety Standards Series No. DS509C, Core Management and Fuel Handling for Research Reactors [6];
- IAEA Safety Standards Series No. DS509D, Operational Limits and Conditions and Operating Procedures for Research Reactors [7];
- IAEA Safety Standards Series No. DS509E, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [8];
- IAEA Safety Standards Series No. SSG-10 (Rev. 1), Ageing Management for Research Reactors [9];
- IAEA Safety Standards Series No. SSG-37 (Rev. 1), Instrumentation and Control Systems and Software Important to Safety for Research Reactors [10].

1.4. Additional recommendations on the safety of research reactors are provided in IAEA Safety Standards Series Nos SSG-20 (Rev. 1), Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report [11] and SSG-24 (Rev. 1), Safety in the Utilization and Modification of Research Reactors [12].

1.5. The terms used in this Safety Guide are to be understood as defined and explained in the IAEA Safety Glossary [13].

1.6. This Safety Guide supersedes IAEA Safety Standards Series No. NS-G-4.6, Radiation Protection

and Radioactive Waste Management in the Design and Operation of Research Reactors[1].

OBJECTIVE

1.7. The objective of this Safety Guide is to provide recommendations on radiation protection and radioactive waste management in the design and operation of research reactor facilities, to meet the relevant requirements of SSR-3 [1], in particular Requirements 8, 15, 34, 57, 59, 84 and 85.

1.8. The recommendations provided in this Safety Guide are aimed at operating organizations of research reactors, regulatory bodies and other organizations involved in a research reactor project.

SCOPE

1.9. This Safety Guide is primarily intended for use for heterogeneous, thermal spectrum research reactors having a power rating of up to several tens of megawatts. For research reactors of higher power, specialized reactors (e.g. fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops) additional guidance may be needed. For such research reactors, the recommendations provided in IAEA Safety Standards Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants [14] might be more suitable. Homogeneous reactors and accelerator driven systems are out of the scope of this publication.

1.10. Research reactors with a low hazard potential having a power rating of up to several tens of kilowatts and critical assemblies and subcritical assemblies might need less comprehensive radiation protection and waste management programmes. While all recommendations in this Safety Guide are to be considered, some might not be applicable to such research reactors, critical assemblies and subcritical assemblies (see paras 2.15– 2.17 and Requirement 12 of SSR-3 [1], and IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [15]).

1.11. In this Safety Guide, subcritical assemblies will be mentioned separately only if a specific recommendation is not relevant for, or is applicable only to, subcritical assemblies.

1.12. Owing to their specialized nature, certain topics are not dealt with in detail in this Safety Guide, such as environmental monitoring, disposal of radioactive waste, management of spent fuel and off-site emergency response. These topics are covered in other IAEA publications [3, 16, 17, 18, 19]; this Safety Guide should be used in conjunction with these publications.

1.13. Except for design approaches and planning for decommissioning (see Section 4), this Safety Guide does not provide recommendations on radiation protection and radioactive waste management in the decommissioning of research reactors.

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors Commissioning of Research Reactors, IAEA Safety Standards Series No. NS-G-4.6, IAEA, Vienna (2008).

STRUCTURE

1.14. This Safety Guide consists of ten sections and three annexes. Section 2 presents the applicable requirements, objectives and concepts for radiation protection and radioactive waste management. Section 3 provides recommendations on the radiological hazards associated with different types of research reactor and with sources of radiation in the research reactor. Section 4 provides recommendations on design aspects for radiation protection and radioactive waste management, including design approaches for operation, decommissioning and managing accident conditions. Section 5 provides recommendations on the control of the radiological hazards identified in Section 3. Section 6 provides recommendations on radioactive waste management in the operation of research reactors. Section 7 provides recommendations on monitoring of facilities and personnel, and on radioactive releases associated with research reactors. Section 8 provides recommendations on instrumentation for radiation protection purposes. Section 9 provides recommendations on organizational aspects of radiation protection at research reactors. Section 10 provides recommendations on the radiological aspects of emergencies at research reactors.

1.15. Annex I provides examples of radiation sources in research reactors to be considered in the programmes for radiation protection and radioactive waste management. Annex II provides an overview of personal protective clothing and equipment, and Annex III gives some considerations with respect to the design of a collection tank or holding tank for liquid radioactive waste.

2. REQUIREMENTS, OBJECTIVES AND CONCEPTS FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT FOR A RESEARCH REACTOR

FUNDAMENTAL SAFETY OBJECTIVE

2.1. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [20] states that "**The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.**" To meet this objective for a research reactor, a comprehensive safety assessment is required to be performed (see Requirement 5 of SSR-3 [1]) to identify all sources of radiation and to evaluate the radiation doses that could be received by workers and the public, as well as potential effects on the environment. This safety assessment should examine the following:

(a) All planned normal modes of operation of the research reactor;

(b) The performance of the research reactor during and following anticipated operational occurrences;

(c) The effects of design basis accidents;

(d) Event sequences that might lead to design extension conditions.

2.2. The safety assessment is used to demonstrate the robustness of the engineering design of the research reactor in withstanding postulated initiating events and accidents and the effectiveness of the safety systems and of safety related items. The safety assessment should also be used as a basis for determining the necessary arrangements for emergency preparedness and response.

2.3. Paragraph 2.1 of SF-1 [20] states:

> "measures have to be taken:
>
> (a) To control the radiation exposure of people and the release of radioactive material to the environment;
>
> (b) To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
>
> (c) To mitigate the consequences of such events if they were to occur."

Such measures could include engineered safety features, on-site accident management procedures established by the operating organization and, possibly, off-site protective measures established by appropriate authorities.

DOSE LIMITS,DOSE CONSTRAINTS AND AUTHORIZED LIMITS ON DISCHARGES FOR A RESEARCH REACTOR

2.4. The design of a research reactor is required to ensure that dose limits established by the regulatory body (in accordance with Requirement 12 of GSR Part 3 [2]) for workers and the public will not be exceeded in operational states (normal operation and anticipated operational occurrences) for the entire lifetime of the research reactor: see Requirement 8 of SSR-3 [1].

2.5. As part of the implementation of the principle of optimization of protection and safety (see paras 2.8–2.10), the operating organization is required to establish dose constraints for controlling occupational exposure and public exposure: see para. 7.112 of SSR-3 [1]. Paragraph 1.22 of GSR Part 3 [2] states that "Dose constraints are set separately for each source under control and they serve as boundary conditions in defining the range of options for the purposes of optimization of protection and safety." For public exposure, dose constraints ensure that the sum of doses from planned operations for all sources under control remains within the dose limit [13].

2.6. A dose constraints should be set for workers who do not enter supervised areas or controlled areas, to ensure that they receive the same level of protection as members of the public (see para. 3.78 of GSR Part 3 [2]). With regard to dose constraints for public exposure, para. 3.36 of IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and Environment [21] states:

> "The dose constraint for a particular source is intended to ensure that the sum of the doses from

planned operations for all sources that may contribute to the exposure of the representative person remains within the dose limit."

Studies should be performed prior to the operation of a research reactor to identify the representative person and the main pathways of exposure for such a person.

2.7. Authorized limits for discharges from a research reactor are determined on the basis of the dose constraint for public exposure and the exposure pathways associated with the representative person. These limits are typically annual limits on the activity of specific radionuclides in liquid and gaseous effluents, which allow for increased release rates over short time periods, thus providing operational flexibility.

APPLICATION OF THE PRINCIPLE OF OPTIMIZATION

2.8. Optimization is a process for ensuring that exposures are as low as reasonably achievable. In applying this process at a research reactor, the following economic and social factors should be taken into account:

(a) The radiation exposure of workers and the public should be reduced by the implementation of radiation protection measures such that further expenditure for radiation protection measures in design, construction and operation would not be warranted by the associated reduction in radiation exposure;

(b) The design of a research reactor should aim to avoid any major disparities in the individual doses received by different types of worker.

2.9. In general, the application of the principle of optimization implies that a choice is to be made from a range of possible protective measures. Feasible options should be identified, the parameters to serve as criteria for comparison and their appropriate values should be determined, and the options should be evaluated and compared. The optimization principle should also be applied to design features, and procedures whose purpose is to prevent, or to mitigate the consequences of, accidents at the facility that could lead to radiation exposure of workers or the public.

2.10. The fundamental role of optimization in the design of a research reactor and its components is to provide a basis for decisions on which engineered provisions for controlling radiation exposures are practicable. This is frequently a matter of judgement, based on past experience. In most cases, the principle of optimization should be applied to achieve a balance between the need for dose reduction, the need to ensure safe and reliable research reactor operation, and the costs involved. A qualitative approach based on expert judgement and on the utilization of the best available and proven technology may be sufficient to make decisions on the optimum level of protection that can be achieved. At the design stage of a research reactor, or for a major modification or at the decommissioning stage, where significant expenditure is involved, the use of a more structured approach may be appropriate and techniques that aid decision making may be useful.

DESIGN TARGET DOSES FOR OPERATIONAL STATES FOR A RESEARCH REACTOR

2.11. In order to ensure that the design is such that protection is optimized, 'design target doses' should be set in terms of both individual doses and collective doses to workers and to members of the public. The term 'target dose' is used throughout this Safety Guide in relation to both individual doses and collective doses. The setting of design target doses in terms of individual doses to workers and to members of the public should be consistent with the concept of dose constraints, as discussed in paras 1.22 and 1.23 of GSR Part 3 [2]. The design target doses should be set with reference to the dose constraints, and with consideration given to technical and more cost effective alternatives for the design. Design target doses are not limits; they are useful design tools in the optimization process. Provided that it can be justified, design target doses may be exceeded. However, compliance with a design target dose does not, in itself, demonstrate that the design satisfies the optimization principle. Therefore, if a further reduction in dose below a design target dose can reasonably be achieved, it should be implemented.

2.12. To focus the design effort on those aspects that contribute most to the collective dose and individual doses to the workers, it is useful to set design target doses in terms of the collective dose to the groups of workers who are expected to receive the highest doses, such as maintenance workers or radiation protection personnel. It is also useful to set design target doses in terms of the collective dose for each category of work, such as the maintenance of major components, in-service inspections, refuelling and waste management. Design target dose should be verified during hot commissioning (see DS509A [4]) of the research reactor.

DESIGN TARGET DOSES FOR ACCIDENT CONDITIONS FOR A RESEARCH REACTOR

2.13. The adequacy of the design provisions for the protection of workers and the public against postulated accident conditions should be judged by comparing predicted doses against design target doses for such accidents. In general, the higher the probability of the postulated accident, the lower the specified design target dose should be. Consequently, the operating organization and/or designer may set different design target doses for postulated accidents with different probabilities of occurrence. In addition, the regulatory body may define design target doses by specifying frequency criteria for all accidents in specified dose bands. For design basis accidents, it would be expected that there will be either no or only minor radiological consequences outside the site boundary, depending on regulatory requirements. The regulatory body should specify what constitutes minor radiological consequences. Typically, these would correspond to levels of exposure for which there would be no need for any off-site protective actions.

2.14. It is beneficial to address design basis accidents and design extension conditions separately and to set design target doses for each of them. In addition to providing assurance to the regulatory body,

design target doses may be set to address the concerns of members of the public.

3. RESEARCH REACTORS AND ASSOCIATED RADIOLOGICAL HAZARDS

3.1. Research reactors are a diverse group of facilities that can be classified in many ways (e.g. as research reactors, training reactors and prototype reactors, critical and subcritical assemblies, by type of moderator and coolant or by purpose of utilization). The radiological hazards and the methods of control vary depending on the potential hazard of the research reactor.

3.2. Not all of the types of radiation source mentioned in this section are relevant for all types of research reactor, nor will the importance of each source be equal for all types of research reactor or their utilization. A careful evaluation of the radiological hazards, their magnitude and their impacts for the specific type of research reactor should be performed to ensure the proper implementation of adequate programmes for radiation protection and radioactive waste management.

RADIOLOGICAL HAZARDS ASSOCIATED WITH DIFFERENT REACTOR TYPES AND UTILIZATION

3.3. The radiological hazard associated with a research reactor will depend predominantly on the following aspects:

(a) The power level of the research reactor and its radionuclide inventory;

(b) The type of research reactor;

(c) The transient (power excursion) characteristics of the research reactor;

(d) The irradiation facilities and experimental devices in and around the reactor core, in particular, beam tubes and any pneumatic ‘rabbit’ systems used to transfer irradiation samples to and from the core.

Power level of the research reactor and radionuclide inventory

3.4. For research reactors with very low power levels, including critical and subcritical assemblies, the low neutron fluxes generally result in insignificant levels of activation products. Systems for the retention or reduction of gaseous releases, as well as the continuous monitoring of releases, might not therefore be necessary. As the amounts of liquid and solid activation products are also low, there may be little or no need for water purification circuits or for the shielding of solid activated components, devices or pipes. The fission product inventory of the core of such reactors is generally low, and shielding is usually only necessary to protect operating personnel from direct radiation from the fuel elements.

3.5. For reactors with power levels above 10 kW, the need for engineered safety features for the retention of gaseous, liquid or solid activation products and for appropriate shielding against the radiation from them becomes gradually more important. Similarly, additional measures should be taken to control releases of radioactive material in operational states and in accident conditions. The need for programmes for radiation monitoring and waste management also increases.

3.6. For reactors with power levels above 1 MW, depending on the design of the core and the fuel elements, fuel damage, including damage from residual decay heat becomes a possibility and special design features are necessary to preserve the integrity of the fuel (e.g. in the event of a loss of coolant accident). In such cases, special consideration should be given to operational aspects (see Appendix II of SSR-3 [1]), as well as to staff training, to ensure an appropriate response to such events. This should also include the capability to adequately monitor any possible accidental releases.

Type of research reactor

3.7. The type of reactor significantly influences the nature and scope of the programmes for radiation protection. While most research reactors of low and medium power levels will have similar characteristics, some designs, in particular at higher power levels, should have particular radiation protection features and programmes. This applies, for instance, to heavy water moderated reactors, for which special attention should be paid to the production of ^{3}H (tritium), including the means of monitoring for this radionuclide. Another example is light water reactors, for which gamma radiation dose rates from ^{41}Ar and ^{16}N should be taken into consideration.

Transient (power excursion) characteristics of the research reactor

3.8. Power excursions (transients) can lead to comparatively high power levels and consequent high levels of direct radiation if the core is not well shielded. Power excursions may be of particular importance for research reactors of low power level or critical and subcritical assemblies, which in normal operating conditions do not need significant shielding. In the design of the irradiation facilities (e.g. beam tubes), the direct radiation caused by transients should be considered.

3.9. If power excursions are part of the design for normal operation, the systems and equipment for both monitoring and radiation protection should be able to withstand high power levels. Power excursions give rise not only to intense direct radiation fields, but also to fuel degradation and the release of fission products from the fuel if it is not designed to cope with such excursions.

3.10. Inherent safety features with a strong negative temperature coefficient may be included in the design so that transients cannot lead to core degradation. Such features are often used in research reactors designed for transient operation (e.g. pulsed research reactors).

Irradiation facilities and experimental devices

3.11. Irradiation facilities and experimental devices use direct radiation from the core, irradiated fuel

or from activated sources, each of which might pose a radiation hazard to personnel. For such facilities and devices, special design features should be implemented to provide for radiation monitoring and radiation protection. This applies specifically to beam tubes and thermal columns, even for research reactors of low power levels.

3.12. Irradiation loops or rigs can present a significant radiological hazard owing to the increased risk of a release of radioactive material caused by high pressures and temperatures. Issues relating to the possible melting of fissile materials, which are usually present in loops or rigs, should be considered in the planning of the radiation protection programme for the research reactor.

3.13. Special precautions should be taken in irradiating materials, including materials in cold neutron sources, that can readily decompose or change state or whose chemical reactivity could be enhanced, causing an overpressure or the release of gases that could be flammable and/or explosive.

3.14. The effect on core reactivity of material placed in irradiation facilities should be determined prior to its irradiation, especially when the irradiated material contains fissile material or fertile material.

RADIATION SOURCES AT A RESEARCH REACTOR

3.15. In the radiation protection programme for any research reactor, account should be taken of all sealed radioactive sources and unsealed radioactive sources that may be present in the facility. A full listing of such sources and their forms, locations and levels of activity during normal operation, as well as during anticipated operational occurrences, provides the basis for shielding calculations, zoning (area designation), the design of ventilation systems, surveillance planning, dose assessment, radioactive waste management and the determination of effluent discharges [10]. Spent fuel presents a radiological hazard from direct radiation and from the release of radioactive material in the event of cladding failures. Consideration should be given, therefore, to the handling, storage, transport and subsequent disposal of spent fuel. Specific recommendations are provided in SSG-15 (Rev. 1) [18].

3.16. The following are examples of possible radiation sources in a research reactor:

(a) Fuel in the reactor core;

(b) Spent fuel and core components stored in the reactor building;

(c) Transuranic nuclides, fission products and activation and corrosion products in the spent fuel pool or the coolant system(s);

(d) Equipment, systems and piping containing activation products;

(e) Solid and liquid radioactive waste and material arising from the treatment of radioactive waste;

(f) Gaseous radioactive material;

(g) Experimental facilities with the potential to generate activated material or other radioactive material;

(h) Tools and devices for the storage and handling of radioactive material, including sample activation and/or irradiation facilities, in-core experiments, beam ports and hot cells;

(i) Material that has been irradiated in the reactor;

(j) Neutron sources and calibration sources;

(k) Neutron detectors (particularly fission chambers and self-powered neutron detectors);

(l) Components of purification systems, such as filters and ion exchange columns.

3.17. Examples of possible radiation sources in a research reactor to be taken into consideration in the programmes for radiation protection and radioactive waste management at a research reactor are provided in Annex I.

4. RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT IN THE DESIGN OF A RESEARCH REACTOR

APPROACH TO DESIGN FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT

Human resources

4.1. The designer of a research reactor should be fully aware of the measures for operational radiation protection and for radioactive waste management that need to be incorporated into the design.

4.2. If necessary, the designer should invite experts from relevant operating organizations and maintenance organizations to participate in the design of a new research reactor and in making design modifications to an existing facility, to ensure that requirements for radiation protection and radioactive waste management are met. In addition, relevant operating experience should be transferred to the designer. In this way, the interrelationship between design and operation can be properly addressed.

4.3. Specialists in radiation protection and radioactive waste management should be closely involved in the design process. Use should be made of advice from these specialists in the design process, in particular in relation to the following:

(a) Their expertise in the production, handling and transport of radioactive material at a research reactor and the release of radioactive material to the environment;

(b) Their ability to provide safety related inputs, such as evaluation of sources of radiation and the associated radiation exposures, using available software and data from relevant operating experience;

(c) Their familiarity with regulatory requirements and guidance and with best practices.

4.4. As part of the application of the optimization principle at the design stage, the management for the research reactor project should establish a system of shared knowledge and common objectives and attitudes to ensure that the management of occupational exposure and public exposure benefits from the cooperation of all personnel who are involved in the project.

Safety culture

4.5. Requirements on fostering a strong safety culture are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [22], and associated recommendations are provided in IAEA Safety Standards Series Nos GS-G-3.5, The Management System for Nuclear Installations [23] and GSG-16, Leadership, Management and Culture for Safety in Radioactive Waste Management [24].

4.6. A strong safety culture associated with the application of the optimization principle should be established by ensuring that all participants in the project are aware of the requirements for radiation protection and of the direct and indirect effects of their activities or functions on the protection of workers, the public and the environment. To implement this approach, the designer should foster an appropriate safety culture in which the importance of radiation protection and of the safety of radioactive waste management at each stage of the design is recognized.

4.7. A strong safety culture in the application of the optimization principle should be established on the basis of the following:

(a) A strong commitment to safety by managers;.

(b) Knowledge of the practices that result in exposure of site personnel and of members of the public;

(c) A comprehensive training programme;

(d) Management of the relationship between operation, maintenance and design;

(e) Familiarity with the main factors, including exposure pathways, that influence individual doses and collective doses;

(f) Familiarity with available software to assist in optimization of the design;

(g) Recognition that specialists in radiation protection should be consulted whenever necessary to ensure that aspects of the design that affect radiation protection are properly evaluated and taken into account.

Organizational aspects

4.8. The need to achieve an adequate level of radiation protection affects a wide range of issues associated with the design of a research reactor. Hence, there is a need to ensure that all design decisions that might affect exposure to radiation are consistent with the requirements for radiation protection. The design process should be planned in such a manner that implementation of these requirements is an integral part of the project and is not on the critical path does not affect its scheduling and completion.

4.9. The whole design process is required to be performed under an appropriate management system established in accordance with Requirements 6–8 of GSR Part 2 [22]. The operating organization should establish a process as part of the management system to ensure that the designers implement the necessary measures for radiation protection and waste safety at all stages of the design process.

4.10. In the organization of the design of a research reactor, it should be ensured that the following are implemented:

(a) Radiation protection specialists and waste safety specialists are consulted at the early stages of the design when options for the major aspects of the design are being evaluated;

(b) The design incorporates proven engineering practices that operating experience has shown to be effective in optimizing protection, and that deviations from such practices are only acceptable when a net benefit has been demonstrated;

(c) All decisions that have a major impact on radiation safety and waste safety should be reviewed by radiation protection specialists and waste safety specialists, respectively;

(d) There is an appropriate review mechanism to resolve any design or operational issues that might be in conflict with requirements for radiation protection and radioactive waste management.

(e) The design incorporates measures to ensure that the activation of structures, systems and components will not prevent their removal or replacement over the lifetime of the research reactor.

DESIGN STRATEGY FOR RADIATION PROTECTION AT A RESEARCH REACTOR

4.11. As described in paras 2.10–2.13, the design target doses set at the start of the design process should include the following:

(a) Design target doses in terms of the annual collective doses and annual individual doses to site personnel;

(b) Design target doses in terms of the annual individual dose to members of the public.

These design target doses are interrelated. For example, any enhancement of the waste treatment systems to reduce the release of radioactive material to the environment may necessitate additional work being performed by site personnel, with a consequent increase in their exposures. In providing means for reducing radioactive releases, the implications in terms of the exposure should be assessed to optimize the protection of site personnel.

4.12. In setting the design target doses, account should be taken, where practicable, of experience at similar research reactors where there has been good operating experience in terms of radiation protection. Account should be taken of any differences in the design and in operation between other facilities and the research reactor that is being designed. Such differences might include differences in the power level, the materials used, the type of fuel, the burnup and the arrangements for conducting

experiments.

Design for radiation protection of site personnel

4.13. To ensure the protection of site personnel, the design of the research reactor should include the following:

(a) A strategy for controlling exposure of site personnel should be developed so that important aspects are considered early in the design. In most types of research reactors, design provisions in two areas may have major impacts in reducing exposures. The first area is preventive and corrective maintenance. Appropriate design provisions should be made for inspection and maintenance. Hence, a proven design with high reliability and less need for maintenance, especially corrective maintenance, would be the preferred option. The second area is design features that minimize the production and buildup of radionuclides, which will reduce radiation levels and contamination levels throughout the facility. In contrast, design features that increase shielding or ventilation may offer less benefit.

(b) General design criteria for radiation protection should be developed and documented. These include the principles on which the layout of the facility will be based, taking into account the interface between safety and nuclear security, provisions for accident conditions, and restrictions on the use of particular materials in the design of the reactor.

(c) A logical layout for the research reactor should be developed and zones (designated areas: see Requirement 24 of GSR Part 3 [2]) should be defined on the basis of predicted dose rates and contamination levels, access requirements and other design requirements, such as the need to separate components that perform safety functions (see para. 6.27 and Requirements 26 and 27 of SSR-3 [1]). The predicted dose rates may be calculated by using source terms that provide the basis for the radiation protection aspects of the design, or they may be determined on the basis of operating experience at similar research reactors (i.e. provided that there are no significant differences in the relevant parameters for design and operation). The designation of radiological zones (designated areas) within the research reactor needs to be consistent with regulatory requirements and with Requirement 24 of GSR Part 3 [2].

(d) The maintenance programme and operations programme should be specified, preferably on the basis of well established concepts for optimizing exposure. The collective doses and individual doses associated with these programmes should be evaluated. Use should be made of relevant operating experience, where available, particularly for work that is difficult to predict precisely, such as corrective maintenance. In such cases, the use of mock-ups would assist in the reduction of uncertainties.

(e) At each stage of the design, the doses that are evaluated should be compared with the design target doses and dose constraints. Where there are options for the design, optimization studies should be performed. These are particularly important in cases where it is predicted that the design

target doses will be exceeded.

4.14. The design provisions to facilitate long term operation of a research reactor should include the following features:

(a) Design features that enable in-service inspection to facilitate the replacement of ageing structures, systems and components. Structures and components that are relatively inaccessible should be identified, and in each case their design life should be specified and clearly documented;

(b) Measures for the prevention of corrosion or other deterioration of the containment or means of confinement and of other items important to safety, including the use of protective coatings and/or cathodic protection;

(c) Provisions for the installation and removal of irradiated sample coupons, where possible, to facilitate periodic inspection.

4.15. An auditable record should be kept of all the decisions that are made in the course of the design process and the reasons for those decisions, so that each aspect of the design that affects exposure to radiation is adequately justified. This record should be part of the management system for the design.

4.16. An outline of the decommissioning plan should be developed during the design stage to help verify that the design includes the necessary features to control exposures during decommissioning. In many cases, these features are the same as those for operational states, but some additional special features may be necessary to facilitate decommissioning. If these additional features are significant, the design should ensure that protection is optimized for operational states and for decommissioning.

4.17. A review of design features for the purpose of facilitating decommissioning is required to be performed at the design stage for the research reactor: see Requirement 33 of SSR-3 [1]. In general, design features that are conducive to maintenance and inspection during the operating lifetime of the reactor will also facilitate decommissioning. In accordance with para. 6.92 of SSR-3 [1] and the requirements established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [25], to facilitate decommissioning, the following are required to be considered:

(a) Careful selection of materials to minimize the following:

  (i) Activation of materials;

  (ii) Generation of radioactive waste;

  (iii) The spread of activated corrosion products;

  (iv) The effort needed for the decontamination of surfaces;

  (v) The use of potentially hazardous substances (e.g. oils, flammable materials, chemically hazardous materials, fibrous insulations).

(b) Optimization of the design, layout and access routes of the facility to facilitate the following:

(i) The removal of large components;

(ii) Easy detachment and removal (by remote means) of significantly activated components;

(iii) Future installation of equipment for decontamination and waste handling;

(iv) Decontamination or removal of embedded components, such as pipes and drains;

(v) The storage of waste;

(vi) Control of radioactive material within the installation.

4.18. The design should facilitate compliance with the design target doses for site personnel using all or some of the following measures:

(a) Reduction of dose rates in working areas by implementing the following:

(i) Source reduction (e.g. by material selection, decontamination, control of corrosion, control of water chemistry, filtration and purification);

(ii) Improvement of shielding;

(iii) Increasing the distance between workers and sources (e.g. by means of provisions for remote handling);

(iv) Ensuring good ventilation.

(b) Reduction of occupancy times in areas with significant dose rates by:

(i) Specifying high reliability equipment to ensure very low failure rates;

(ii) Ensuring ease of maintenance or ease of removal of equipment;

(iii) Simplifying operating procedures;

(iv) Ensuring ease of access and good lighting.

Design for radiation protection of members of the public

4.19. Design target doses should be set at the start of the design process for the annual individual doses to members of the public. Developments in the area surrounding the site and likely future population distributions should be taken into account, as necessary.

4.20. Compliance with the design target doses should be achieved by considering any site specific features that could affect doses to members of the public. Such features should be identified at an early stage of the design process and should be taken into account in the design. This should include the identification of the representative person and the main exposure pathways.

Design strategy for radiation protection in the commissioning of a research reactor

4.21. The measures that are included in the design to provide an optimized level of radiation protection in operational states are usually adequate for radiation protection in the commissioning stage. Special

care should be taken, however, during power and overpower tests. During these tests, comprehensive measurements, tests and verifications for radiation protection purposes should be performed for ensuring the adequacy of the design (see SSG-24 (Rev. 1) [11]).

DESIGN FEATURES FOR RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

4.22. In accordance with Requirement 15 of SSR-3 [1] and Requirements 8 and 17 of GSR-Part 5 [3], the design is required to incorporate features to control the generation of radioactive waste at the research reactor and to facilitate the safe handling and storage of this waste, the removal and transport of waste and its disposal, and to control effluent discharges. Subcritical assemblies and lower power research reactors typically do not generate significant quantities of radioactive waste and design provisions for radioactive waste management in these facilities should be applied using a graded approach.

4.23. Requirement 59 of SSR-3 [1] states:

> "**The design of a research reactor facility and its associated experimental facilities shall include provisions to enhance safety in waste management and to minimize the generation of radioactive waste. Systems shall be provided for treating solid, liquid and gaseous radioactive waste to keep the amounts and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges.**"

Provisions for the interim storage of waste in transit and for the removal of waste should also be considered.

4.24. Where necessary, provision should be made for an interim means of confinement in areas where radioactive effluent or radioactive waste is stored prior to its treatment and discharge or disposal.

4.25. The design should ensure adequate flexibility in terms of the facilities for handling radioactive waste, including radioactive waste that might be generated in abnormal situations (e.g. faulty containers) and radioactive waste of non-standard physical or chemical composition, and for handling major components from modifications and experiments.

4.26. In accordance with paras 6.27(a) and 6.92(a) and Requirement 59 of SSR-3 [1], the design is required to minimize the generation of radioactive waste in all operational stages in the lifetime of the facility, including decommissioning. Design features are required to be implemented to ensure the amounts and concentrations of radioactive releases as low as reasonably achievable and that discharges remain below authorized limits: see Requirement 59 of SSR-3 [1]. Design considerations for waste minimization should be compatible with the safety analysis and with relevant dose criteria and the need to ensure that exposures are as low as reasonably achievable (see also para. 4.11). The measures implemented in research reactors include the following:

(a) Selection of materials that are less prone to activation (e.g. use of plastic for the pneumatic

'rabbit' system irradiation target carriers) or materials that generate activation products that decay quickly (e.g. use of aluminium components near the core);

(b) Making allowance for the thermal expansion and contraction of pool water in a manner that avoids or minimizes overflow to liquid retention tanks;

(c) Minimization or evacuation of air spaces near neutron sources to reduce the production of radionuclides (e.g. ^{41}Ar).

4.27. In accordance with paras 6.27, 6.101 and 6.202 of SSR-3 [1], the following features to facilitate the management of radioactive waste generated during operation of the research reactor need to be considered in the design:

(a) Provisions that reduce the quantity of the radioactive waste generated and transferred within the facility and that minimize the activity in effluents released to the environment.

(b) Provisions for the isolation of radioactive waste from site personnel and the public, with access control to radiological zones (designated areas) for radiation protection purposes. Controlled areas and supervised areas are required to be designated in accordance with the potential for radioactive contamination and radiation exposure: see paras 3.88–3.92 of GSR Part 3 [2]. Recommendations on the designation of areas are provided in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [26].

(c) Provisions for local detection, collection and treatment of liquid leakages and spills before they are discharged as effluents.

(d) Provisions for the decontamination of personnel and equipment including mobile devices.

(e) Provisions for handling the radioactive waste arising from decontamination activities.

4.28. The extent to which the containment or the means of confinement is vented in operational states to prevent the buildup of radioactive gases should be addressed in the design.

4.29. The varying nature of the radioactive waste generated during the lifetime of a research reactor, should be taken into account in the design of the shielding, the means of confinement and the isolation features associated with facilities, equipment and components for waste management. Safety assessment of radioactive waste management activities and facilities in accordance with the recommendations provided in IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [27] could be used for the optimisation of these design features.

Management of gaseous effluents at a research reactor

4.30. The measures for the management of gaseous effluents to be considered in the design of a research reactor include the following:

(a) Provision for radioactive gases to be channelled through proper ducting to a common release

point;

(b) Provisions for the proper selection of process gases and decay systems (e.g. the use of delay tanks for ^{16}N) to minimize releases of gaseous radioactive material;

(c) Provision of appropriate filters to clean gaseous effluent to minimize the discharges while also minimizing the production of secondary waste (see also para. 6.211 of SSR-3 [1]);

(d) Provision of means for the discharge of gases, such as a suitable stack.

(e) Provision of means and methods for sampling and monitoring discharges of gaseous effluents in accordance with para. 7.117 of SSR-3 [1].

Management of liquid effluents at a research reactor

4.31. The measures for the management of liquid effluents to be considered in the design of a research reactor include the following:

(a) Collection of radioactive liquid effluents to a common point such as a holding tank (general design considerations for collection or holding tanks are discussed in Annex III);

(b) Where necessary, provisions to address the potential for reconcentration downstream of some discharged radionuclides, especially in relation to liquid radioactive waste with low levels of activity (such as liquids from pit tanks, delay tanks and holding tanks that might be suitable for discharge through a low level effluent system directly to sewers), and the means and methods to monitor such discharges;

(c) Provisions for the management and control of liquid radioactive waste with higher levels of activity, such as waste that might arise from planned major shutdowns of some types of research reactor;

(d) Provisions for decay systems (e.g. use of retention tanks) to minimize releases of radioactive substances;

(e) Provisions for sampling and monitoring holding tanks prior to the release of liquid content, preferably at the point of release;

(f) Provisions for the treatment of liquid effluents (e.g. using ion exchange resins), either for reuse or because the activity levels are too high for their management as waste or release to the environment.

(g) Provisions for filtration in liquid waste collection lines to prevent the release of solids.

Management of solid radioactive waste at a research reactor

4.32. In accordance with the requirements established in GSR Part 5 [3] and the recommendations provided in GSG-16 [24], the measures for the management of solid radioactive waste (including waste generated by experiments) to be considered in the design of a research reactor include the following:

(a) Provisions for segregating waste by type (amount, form, volume, isotopic composition and activity concentration);

(b) Provisions for the packaging, handling and storage of solid low level radioactive waste, such as contaminated cleaning equipment, protective clothing, paper and tools, which might be accumulated and treated in on-site storage and treatment facilities;

(c) The packaging, handling and storage of solid intermediate level radioactive waste, such as waste arising from ion exchange resins, ventilation filters and charcoal beds, which might be accumulated and treated in on-site storage and treatment facilities;

(d) The packaging, handling and storage of solid high level radioactive waste such as replaceable core components;

(e) Areas and tools for handling and loading waste;

(f) Equipment and tools for radiation protection purposes;

(g) Provisions for compacting waste, if needed;

(h) Provisions, as necessary, for storing resins and for dehydrating liquid waste;

(i) Space for storing waste until its transport from the site;

(j) Provisions for ensuring that any solid radioactive material discharged in liquid effluents is within authorized limits.

LIFETIME MANAGEMENT AND DECOMMISSIONING OF A RESEARCH REACTOR

4.33. Requirement 33 of SSR-3 [1] states that "**Decommissioning of a research reactor facility shall be considered in the design for the research reactor and its experimental facilities**."

4.34. The expected lifetime of the research reactor should be stated in the design. Issues that could affect the possible extension of the lifetime of the facility and its eventual decommissioning should be identified at the design stage. The design should include features for the storage and/or processing of radioactive waste generated throughout the lifetime of the research reactor. General requirements on the decommissioning of facilities are established in GSR Part 6 [25].

DESIGN APPROACHES FOR DEALING WITH ACCIDENT CONDITIONS AT A RESEARCH REACTOR

4.35. The design of a research reactor is required to protect people and the environment against the possible radiological consequences of accidents, by reducing the likelihood that accidents will occur (prevention of accidents) and by mitigating the consequences associated with accidents if they do occur: see paras 2.8 and 6.13 of SSR-3 [1].

4.36. The design objectives for accident conditions are to limit to acceptable levels the risks to the

public from potential releases of radioactive material from the research reactor and the risks to site personnel from such releases and from direct radiation exposure. These design objectives are to be achieved through high quality design and special features, such as protection systems and safety systems, which are incorporated into the design of the research reactor in accordance with the requirements established in section 6 of SSR-3 [1]. The aim is to ensure the reliable control of reactivity and, for reactors with non-negligible thermal power levels and radioactive inventories, to ensure the removal of residual heat and the confinement of radioactive material.

4.37. The achievement of the design objectives should be confirmed by means of a safety analysis conducted in accordance with Requirement 41 of SSR-3 [1]. Deterministic safety analyses and the associated dose assessments, together with, as appropriate, complementary probabilistic safety analyses, should be performed using conservative assumptions for the analyses of design basis accidents, and using realistic or best estimate assumptions for the analyses of design extension conditions.

4.38. To achieve the design objectives, the design should include provisions and procedures (e.g. shutdown systems, systems for residual heat removal, actions on receipt of alarms) to enable operating personnel to take appropriate actions in accident conditions.

4.39. The measures used for radiation protection in operational states should also be employed to ensure adequate protection for site personnel and the public in accident conditions.

4.40. The design of the structures, systems and components (see paras 4.41–4.49) for radiation protection in accident conditions should be developed through consultation with experts in radiation protection, reactor operations, design and accident analyses and regulatory requirements. There should be regular interaction among these experts throughout the design process to produce a design that provides a level of radiation protection under accident conditions that is acceptable to the operating organization and the regulatory body. The design should also be such as to ensure that effective arrangements for accident management can be implemented.

DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

4.41. The design of structures, systems and components for research reactors should take into account experience that has been gained in reducing radiation exposure at operating research reactors.

4.42. The following features for reducing radiation exposure should be incorporated into the research reactor design:

(a) Work areas with high radiation levels containing components that need regular maintenance should be shielded from the radiation emitted by nearby active systems and components;

(b) Non-radioactive components that do not need to be mounted close to active components should

be installed outside zones (controlled areas) with high radiation levels;

(c) Components to be used in zones (controlled areas) with high radiation levels should be designed to be easily removable;

(d) Techniques for sampling radioactive liquids while ensuring that protection is optimized should be provided;

(e) Sedimentation of radioactive sludge in piping and containers should be prevented, and necessary measures to prevent this should be provided.

4.43. The design should provide for structures, systems and components of high reliability that involve the minimum surveillance, maintenance, testing and calibration (see para. 4.13(a)).

4.44. The exposure of site personnel should be reduced by minimizing the amount of radioactive material that might accumulate in reactor components. Thus, traps and rough surfaces where radioactive material can accumulate should be avoided, as far as practicable.

4.45. Components and areas of buildings that could become contaminated are required to be designed to facilitate decontamination: see para 6.97 of SSR-3 [1]. This includes using suitable coatings on walls and floors, smooth surfaces, avoiding angles and pockets where radioactive material can collect, and providing means of isolation, flushing and drainage for circuits that contain radioactive liquid.

4.46. Pipelines and tanks containing radioactive fluids should not be placed near clean piping or tanks, and they should be properly shielded or should be located at a distance from items that need maintenance. Sufficient space should be left between pipelines or tanks and walls for making inspections as well as for repairs and modifications.

4.47. The uncontrolled buildup of radioactive particles should be prevented by means of chemistry control and appropriate design for fluid flow, including the use of piping with a smooth and even inner surface.

4.48. Pipelines should be designed in such a way that the venting and drainage lines are minimized. Drainage should lead to a sump or a closed system. Pipelines should be designed to avoid causing fluid to collect anywhere.

4.49. In the design of pipelines, welded seams requiring inspection should be avoided to the extent practicable, and any such welded seams should be easily accessible for inspection and repair.

DESIGN FEATURES TO FACILITATE DECONTAMINATION AT A RESEARCH REACTOR

4.50. In accordance with Requirement 34 of SSR-3 [1], the need for decontamination should be considered at the design stage of a research reactor. In particular, para. 6.101 of SSR-3 [1] states that "Provision shall be made for appropriate decontamination facilities for both personnel and equipment

and for handling the radioactive waste arising from decontamination activities."

4.51. When decontamination facilities are being planned, all components expected to come into contact with primary coolant or waste material should be considered potential items for decontamination.

4.52. Special consideration is necessary for areas where leakage or spills of contaminated liquid might occur. These areas should be designed to control of the spread of contamination and to facilitate decontamination.

4.53. Decontamination facilities should be provided for removing radioactive material from the surfaces of transport packages (e.g. containing irradiated fuel elements or radioactive waste) before shipment, as well as from components that may need to be repaired and from tools and equipment.

DESIGN OF SHIELDING AT A RESEARCH REACTOR

4.54. Paragraph 6.98 of SSR-3 states:

> "The design shall include the shielding required not only for the reactor but also for experimental devices and associated facilities (e.g. beam tubes, particle guides or facilities for neutron radiography or boron neutron capture therapy) and provision shall be made for installing the necessary shielding associated with the future utilization of the reactor and other radiation sources."

4.55. In designing shielding for a specific radiation source, a target dose rate should be set on the basis of the expected frequency and duration of occupancy of the area. When setting this target dose rate, any uncertainties associated with determining the output of the radiation source should be taken into account.

4.56. In establishing specifications for shielding, the buildup of radionuclides over the lifetime of the research reactor should be taken into account.

4.57. The choice of shielding materials should be made on the basis of the nature of the radiation (e.g. whether neutron radiation and gamma radiation, or gamma radiation only), the shielding properties of materials (e.g. their degree of scattering, absorption, production of secondary radiation, activation), their mechanical and other properties (e.g. stability and compatibility with other materials, structural characteristics, seismic behaviour), and space and weight limitations.

4.58. Losses in the efficiency of shielding might occur as a result of environmental conditions. The design of shielding should take into account effects due to interactions of neutron radiation and gamma radiation with the shielding materials (e.g. the burnup of radionuclides that have high neutron absorption cross-sections, radiolysis and embrittlement), effects due to reactions with other materials (e.g. erosion or corrosion), and effects due to temperature (e.g. the removal of hydrogen and/or water from concrete).

4.59. Penetrations through shielding (e.g. pipelines, ducts) should be appropriately designed to avoid direct streaming of radiation (e.g. use of multiple bends in pipes). Shielding composed of several blocks should be designed to prevent neutron streaming through gaps between these blocks (i.e. avoiding a

'line-of-sight' to the neutron source). The effects of 'sky shine' and 'ground shine' should be considered when designing shielding.

4.60. The generation of heat in the shielding material should be taken into account and, if necessary, an appropriate cooling system should be included in the design of the shielding.

4.61. A combination of materials may be necessary to obtain an optimum design of shielding for the core or for other sources of neutrons. A material such as iron or steel, with a high inelastic scattering cross-section, should be used to reduce the energy of high energy neutrons. A material such as water or concrete, containing elements of low atomic number, reduces the energies of neutrons for which the cross-sections are below the threshold for nuclear inelastic scattering by the shielding material(s).

4.62. In areas where temporary additional shielding may be necessary in operational states of the research reactor, the weight of additional shielding and the provisions necessary for transporting and installing it should be taken into account in the design.

DESIGN OF VENTILATION SYSTEMS AT A RESEARCH REACTOR

4.63. Provision of a dedicated active ventilation system should be considered and, if necessary, this system should maintain appropriate clean conditions in working areas within the controlled area, in accordance with para. 6.99 of SSR-3 [1].

4.64. For the purposes of radiation protection, the primary objective of providing a ventilation system should be to control the contamination of the working environment by airborne radionuclides and to reduce the need to wear respiratory protection. Design provisions should be made for ventilation in the event of an accident.

4.65. The spread of airborne contamination and the amount of radioactive material released to the environment is required to be controlled by providing features such as air cleaning filters (see para. 6.99 of SSR-3 [1]) and by maintaining appropriate pressure differentials between areas.

4.66. In designing a ventilation system to control airborne contamination, account should be taken of the following:

(a) Mechanisms of thermal and mechanical mixing;

(b) Exhausting of air from areas of potential contamination at points near the source of contamination;

(c) Preventing the contamination of adjacent facilities;

(d) Use of air exchange rates that are commensurate with the potential for contamination of the area;

(e) Arranging the airflow to minimize the potential for resuspension of contamination;

(f) Ensuring that the discharge point for exhaust air is not close to an intake point of the ventilation system;

(g) The provisions of redundancy, independency and diversity in the system.

4.67. The airflow in the ventilation system should be such that the pressure in an area with lower levels of airborne contamination is higher than the pressure in an area with potentially higher levels of contamination. Thus, the airflow in the ventilation system should be directed from regions with lower levels of airborne contamination to areas with higher levels, and air should be extracted from the latter. If this is not feasible, separate ventilation systems should be provided for the two areas.

5. RADIATION PROTECTION IN THE OPERATION OF A RESEARCH REACTOR

5.1. A radiation dose might be received from external exposure (i.e. from direct radiation, including immersion in a gas) and from internal exposure (i.e. from ingestion, inhalation or absorption through the skin or through wounds). The goals of radiation protection are to ensure the effective control of external exposure and internal exposure of workers and to the public, and of releases to the environment, to ensure conformance with all regulatory requirements and to enable further optimization of operational practices.

5.2. Requirement 84 of SSR-3 [1] states that "**The operating organization for a research reactor facility shall establish and implement a radiation protection programme**." The basis of the radiation protection programme is a set of measures (e.g. design features, administrative controls, adequate training) aimed at reducing the radiation hazard and optimizing protection and safety. The measures include restrictions specified in operating procedures in combination with an extensive monitoring programme for radiation and contamination. Recommendations on the contents of a radiation protection programme are provided in GSG-7 [26]. Recommendations on the control of discharges and the calculation of public exposure resulting from authorized releases are provided in IAEA Safety Standards Series Nos GSG-9, Regulatory Control of Radioactive Discharges to the Environment [16] and GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [28].

5.3. A high standard of housekeeping is required at a research reactor: see Requirement 76 of SSR-3 [1]. Control of contamination and of radiation exposure is greatly facilitated in a clean and tidy area. Good housekeeping should be encouraged by ensuring that operating personnel and experimenters arrange for the cleanup and restoration of an area when they leave or at the completion of the work or the experiment, and that they identify and promptly report any equipment that is degraded. The operating organization should ensure that appropriate corrective actions are taken to maintain a high standard of material conditions at the facility.

5.4. At most research reactors, areas accessible to the general public should be sufficiently far away from radiation sources to give assurance that direct external radiation doses to members of the public

are negligible under normal operating conditions. Public exposure would generally only occur as a consequence of the routine discharge of radioactive effluents from the research reactor. In estimating public exposure, relevant exposure pathways should be considered (e.g, through the food chain or from ground deposition).

5.5. Radioactive releases from research reactors might arise in gaseous, liquid and solid forms. Control and monitoring of each of these three types of release should be performed to ensure that the doses from these releases are below the dose constraints for public exposure. With all types of release, the operating organization is required to minimize both the radioactivity content and the volume: see paras 6.12 and 7.116 of SSR-3 [1]. Recommendations on the management of radioactive waste during the operation of a research reactor are provided in Section 6 of this Safety Guide. Recommendations on assessing the impact of authorized releases on the environment are provided in GSG-10 [28].

DOSE LIMITS

5.6. Paragraph 7.107 of SSR-3 [1] states:

> "The radiation protection programme shall ensure that for all operational states and accident conditions, doses due to exposure to ionizing radiation at the research reactor facility or doses due to any planned releases of radioactive material from the facility are kept below authorized limits and are as low as reasonably achievable."

In meeting this requirement, it needs to be ensured that the exposure of individuals from the normal operation of the research reactor is such that neither the total effective dose nor the total equivalent dose to relevant organs and tissues exceeds any relevant dose limit set by national regulations (see also Schedule III of GSR Part 3 [2]).

OPTIMIZATION PRINCIPLE AND DOSE CONSTRAINTS

5.7. The operating organization should take appropriate measures to ensure that protection and safety is optimized so that the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures are kept as low as reasonably achievable, economic and social factors being taken into account [13]. For each radiation source, the level of radiation protection provided should be optimized so that both individual doses and collective doses (for both normal exposure and potential exposure) are kept as low as reasonably achievable.

5.8. Optimization of protection and safety associated with any particular source should be subject to dose constraints, as described in Section 2 of this Safety Guide. A dose constraint is a source related quantity used to restrict the range of options considered in the process of optimization. A dose constraint is not a limit on doses but rather a ceiling on the values of individual dose that should be considered acceptable in the optimization process. It is used prospectively for the planning and execution of tasks as well as for design purposes.

5.9. To apply the optimization principle, the individual doses for the various options that are under consideration should be predicted in the planning stage and compared with the appropriate dose constraint. Any options that are predicted to produce doses that would exceed the constraint should be rejected, or else should be revised to bring the predicted doses within the constraint.

5.10. The government or the regulatory body is responsible for establishing or approving of dose constraints: see para. 3.22(c) of GSR Part 3 [2]. In practice, the regulatory body should generally encourage the operating organization to develop dose constraints, which are then subject to regulatory approval.

5.11. The dose constraint should be used to ensure that the dose limit for public exposure is not exceeded: see paras 2.4–2.7.

ADMINISTRATIVE CONTROL LEVELS

5.12. Administrative control levels are values of measured quantities (e.g. dose rates, activity, activity concentrations) above which some specified action or decision should be taken. These levels (which include recording levels, investigation levels and operational intervention levels) are an important part of the radiation protection programme. The administrative control levels should be set by the operating organization and should comply with regulatory requirements.

5.13. A recording level is a level of dose, exposure or intake specified by the regulatory body at or above which the values of dose, exposure or intake received by workers are to be entered in their individual exposure records [13]. An investigation level is the value of a quantity such as effective dose, intake or contamination per unit area or unit volume at or above which an investigation would be conducted [13]. Recommendations on the use of recording levels and investigation levels are provided in GSG-7 [26].

5.14. Operational intervention levels are used to protect members of the public in the event of an emergency. Operational intervention levels are typically expressed in terms of dose rates or of activity of radioactive material released, time integrated air activity concentrations, ground or surface concentrations, or activity concentrations of radionuclides in environmental, food or water samples [13]. The values of operational intervention levels included in the emergency plan for a research reactor should be used as initial criteria for implementing protective actions, but they may be modified to take into account the prevailing circumstances and their likely evolution.

ON-SITE RADIATION PROTECTION AND DOSE CONTROL AT A RESEARCH REACTOR

Radiation protection through source control

Control by material selection

5.15. A large reduction in exposure can be achieved at the design stage of a research reactor and during

the design of individual experiments. In particular, care should be taken in the choice of materials that are likely to be activated, with account taken of neutron activation cross-sections, half-lives and corrosion resistance. Impurities in standard materials should be carefully investigated as part of minimizing doses to operating personnel and experimenters.

Control by quality of reactor coolant

5.16. The design of a research reactor is required to include provisions for the removal radioactive substances, including activated corrosion products and fission products: see para. 6.162 of SSR-3 [1]. The reactor coolant should be kept as free from impurities as possible to minimize dose rates from exposures due to coolant activation and to minimize problems that might arise due to contamination. Good coolant chemistry should be maintained to reduce corrosion and thereby minimize the activation of corrosion products. Items that have been in contact with the reactor coolant should be monitored for contamination. Care should be taken to ensure that reactor coolant does not leak and give rise to a spread of contamination.

5.17. Cleanup systems such as filters and ion exchange columns should be used to remove contaminants from reactor coolant. These systems can themselves become significant sources of radiation, and the design should provide features to optimize protection. For reactors moderated or cooled by heavy water, the release of such material would also generate a hazard from the inhalation or absorption of tritiated water in vapour form.

Control of beam ports and beam tubes

5.18. Beam ports and similar devices can be significant sources of radiation and, therefore, should be well designed, controlled and monitored. Beam tubes are required to be adequately shielded: see para. 6.98 of SSR-3 [1]. Beam ports that are not in use should be properly closed and shielded.

Control by minimizing the production of gaseous material

5.19. Noble gases produced in research reactors consist of activation products such as ^{41}Ar (in the case of irradiation of air) and isotopes of krypton and xenon as fission products. Production of radioactive gases can be minimized, for example, by the following:

(a) Reducing the ingress of air near the core;

(b) Sealing air spaces to let the radioactive material decay, or purging the spaces with a gas such as carbon dioxide or helium (or a combination of these), or evacuating the spaces;

(c) Removing and sealing failed fuel and defective devices containing radioactive gases.

5.20. Filters are usually installed in ventilation systems to retain particulate radioactive material within the building and reduce releases to the environment and public exposure. The installation of filters close to the source of radiation can help to prevent the buildup of radioactive material in ducts and thereby reduce any possible leakage of contaminated air into the workplace, and prevent the ducts from

becoming an external source of radiation.

Control of irradiated fuel

5.21. Irradiated fuel is a significant source of radiation and should be handled and stored in such a way that the corrosion and/or failure of cladding, melting of fuel and the possibility of unintentional criticality are avoided. Specific recommendations on the storage of spent fuel are provided in SSG-15 (Rev. 1) [18].

Control by minimization of waste

5.22. Radioactive waste produced by the research reactor is also a source of radiation and is required to be minimized to the extent possible: see paras 6.12 and 7.116 of SSR-3 [1]. Radioactive waste should be segregated, contained and shielded, and placed in designated collection points prior to its predisposal management and disposal.

Radiation protection through physical controls and the use of time and distance

5.23. Appropriate engineered controls are required to be included in the design, such as shielding, ventilation and decay systems: see para. 6.94 of SSR-3 [1]. Radiation protection should also be provided through the use of distance and time. Shielding should be used where there is a need to reduce the dose rates everywhere outside the shield, and ventilation should be used where there is a need to control the levels of airborne radioactive material. The use of time and distance should be considered as additional ways to reduce exposures. The ways in which these factors apply in the operation of research reactors are considered in the following section.

Shielding

5.24. Paragraph 6.98 of SSR-3 [1] states:

> "The design shall include the shielding required not only for the reactor but also for experimental devices and associated facilities (e.g. beam tubes, particle guides or facilities for neutron radiography or boron neutron capture therapy) and provision shall be made for installing the necessary shielding associated with the future utilization of the reactor and other radiation sources. Hazard assessments and shielding arrangements shall be given due consideration in relation to the use of beam tubes and other experimental devices."

5.25. The need for shielding should be considered for all sources of radiation at the research reactor (see paras 3.16 and Annex I), including the following:

(a) The reactor core;

(b) The reactor coolant system;

(c) Filters;

(d) Ion exchange columns;

(e) Argon vent pipes;

(f) Beam ports, beam tubes and beam targets;

(g) Irradiated samples and equipment;

(h) Storage tanks for radioactive gases or liquid effluents;

(i) Irradiated fuel.

5.26. In the design of shielding, consideration should be given to the fact that, in most cases, a mixed neutron and gamma radiation field is present. Neutrons and gamma radiation are attenuated by different materials and, because of neutron scattering, the order of the different shielding materials should also be considered.

Ventilation

5.27. Ventilation should be used as a means of supplying fresh air for breathing and for removing and/or diluting the radioactive contaminants present in the work environment. It is also the means for maintaining ambient temperature and humidity. Slight underpressure should be maintained to confine the contaminated air where necessary. Recommendations on the design of ventilation systems for a research reactor are provided in paras 4.63–4.67 of this Safety Guide. Depending on the design of the facility, ventilation should be placed above the reactor to control concentrations of ^{41}Ar and, in the event of fuel failure, fission product gases.

5.28. Ventilation systems with appropriate filtration are required to be provided for use in operational states and in accident conditions: see para. 6.99 of SSR-3 [1]. Depending on the design of the research reactor, the ventilation system should include a separate subsystem with charcoal filters to be used in accident conditions involving a radioactive release. In many research reactors, ventilation systems are essential for the fulfilment of the confinement function.

5.29. The efficiency of the ventilation systems should be routinely checked in terms of integrity, the number of air changes per hour provided in different areas, flow rates, velocities and pressures. The efficiency of filters should be checked and assessed periodically.

Distance

5.30. Site personnel should be made aware of the need to make effective use of distance between themselves and sources of radiation, and to stay in areas of lower dose rates whenever possible. The use of special tools may help increase the distance from the source. Distance control may be combined with shielding. For example, manipulators or robots can be used for actions remotely controlled from greater distances behind a shield.

Time

5.31. The time spent by persons in areas of high dose rates should always be minimized. In some cases (i.e. where the dose rates are high even after the use shielding and distance), this may be the only

practicable method of dose control. For tasks that have to be undertaken in areas of high dose rates, the action should be planned and, whenever possible, practised with a mock-up or without the source present. This enables the personnel involved to refine the procedure and to perform the task faster, and also provides an opportunity to anticipate problems before they arise.

Radioactive decay

5.32. Activated material, equipment and samples should be allowed to decay as long as possible before they are handled. This often involves careful planning and scheduling. In some cases, time for appropriate decay is ensured by means of delay systems (e.g. tanks).

Decontamination

5.33. Routine operations and experiments may involve the handling of contaminated components and equipment, and might result in the spread of surface contamination. Procedures, facilities and equipment are therefore required for the decontamination of skin, clothing, tools, surfaces in controlled areas and equipment: see para. 6.101 of SSR-3 [1] (see also paras 4.50–4.53 of this Safety Guide). Procedures should be established in which both the contaminant and the contaminated material are considered.

5.34. All items taken from radioactive systems for maintenance or research purposes should be decontaminated to below specified contamination limits before they are released from a controlled area.

5.35. Components and areas of buildings that could become contaminated are required to be designed to facilitate decontamination: see para. 6.97 of SSR-3 [1]. To facilitate efficient decontamination, surfaces that could become contaminated should be smooth and easily cleanable. This applies to surfaces of areas, fittings and furniture, as well as to equipment and experimental devices.

Personal protective equipment

5.36. Priority should be given to preventing, eliminating or controlling sources of radiation hazards. However, there can be situations in which residual sources of airborne and surface contamination continue to pose an exposure hazard in the research reactor. In such situations where individual contamination is possible, appropriate personal protective equipment, including protective clothing, is required to be provided by the operating organization: see para. 3.95(a) of GSR Part 3 [2].

5.37. Protective clothing and equipment should be commensurate with the radiation hazard, and may vary from a simple laboratory coat, gloves and overshoes to a full protective suit with air supply. For protecting the lens of the eye, plastic or safety glass goggles should be provided, where necessary. The personal protective equipment should be used in accordance with appropriate procedures, and training and retraining it its correct use should be provided. Further information is provided in Annex II.

5.38. Tasks requiring the use of specific personal protective equipment, such as breathing equipment, are required to be assigned only to workers who, on the basis of medical advice, are capable of safely sustaining the extra effort necessary: see para. 3.95(c) of GSR Part 3 [2].

5.39. Whenever the use of personal protective equipment is considered for any given task, consideration is required to be given to any additional exposure that could result owing to the additional time taken or because of its inconvenience: see para. 3.95(d) of GSR Part 3 [2].

Radiation protection through administrative controls

5.40. Administrative controls defined in local rules and operating procedures form an essential part of the radiation protection programme. Care should be taken to ensure that only reasonable and necessary measures are imposed. Compliance with administrative controls should be emphasized in the training programmes and should be enforced in appropriate ways.

5.41. In establishing administrative controls, the operating organization should take into account human factors. The need for administrative controls should be minimized by providing appropriate protective measures and safety measures, including engineered controls, and satisfactory working conditions.

5.42. A workplace monitoring programme is required to be established as part of the radiation protection programme: see para. 7.110(b) of SSR-3 [1]. This programme should be maintained and regularly reviewed, and may include monitoring of the following:

(a) Leaks of radioactive material (e.g. by means of a 'wipe test' or other means);

(b) Dose rates (neutron radiation and gamma radiation);

(c) Surface contamination;

(d) Airborne contamination.

Area classification and access control

5.43. Areas within the research reactor are required to be classified (see paras 5.44–5.50 of this Safety Guide) in accordance with Requirement 24 of GSR Part 3 [2]. The designation of areas should be indicated with appropriate warning signs that provide information on the nature of the hazard: see para. 5.48 of this Safety Guide.

5.44. To minimize the potential for ingestion of radioactive material, actions such as eating, drinking, smoking or applying cosmetics should be prohibited in areas in which there is contamination.

5.45. Paragraph 3.88 of GSR Part 3 [1] states that the operating organization:

> "shall designate as a controlled area any area in which specific protective measures or safety provisions are or could be required for:
>
> (a) Controlling exposures or preventing the spread of contamination in normal operation;
>
> (b) Preventing or limiting the likelihood and magnitude of exposures in anticipated operational occurrences and accident conditions."

5.46. In determining the boundaries of any controlled area, the magnitudes of the expected normal

exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the procedures for protection and safety should be taken into account. In research reactors, controlled areas are normally surrounded by supervised areas.

5.47. Paragraph 3.90 of GSR Part 3 [1] states (citations removed) that the operating organization:

(a) "Shall delineate controlled areas by physical means or, where this is not reasonably practicable, by some other suitable means.

(b) Shall, where a source is only intermittently brought into operation or energized, or is moved from place to place, delineate an appropriate controlled area by means that are appropriate under the prevailing circumstances and shall specify exposure times.

(c) Shall display the symbol recommended by the International Organization for Standardization and shall display instructions at access points to and at appropriate locations within controlled areas.

(d) Shall establish measures for protection and safety, including, as appropriate, physical measures to control the spread of contamination and local rules and procedures for controlled areas.

(e) Shall restrict access to controlled areas by means of administrative procedures such as the use of work permits, and by physical barriers, which could include locks or interlocks, the degree of restriction being commensurate with the likelihood and magnitude of exposures.

(f) Shall provide, as appropriate, at entrances to controlled areas:

(i) Personal protective equipment;

(ii) Equipment for individual monitoring and workplace monitoring;

(iii) Suitable storage for personal clothing.

(h) Shall provide, as appropriate, at exits from controlled areas:

(i) Equipment for monitoring for contamination of skin and clothing;

(ii) Equipment for monitoring for contamination of any objects or material being removed from the area;

(iii) Washing or showering facilities and other personal decontamination facilities;

(iv) Suitable storage for contaminated personal protective equipment.

(v) Shall periodically review conditions to assess whether there is any need to modify the measures for protection and safety or the boundaries of controlled areas;

(vi) Shall provide appropriate information, instruction and training for persons working in controlled areas."

5.48. In some areas, optimization of protection can be achieved only by limiting the time spent there or by using special protective equipment. The definition of different zones within a controlled area, on the basis of dose rates or levels of non-fixed contamination, should be considered. Some zones will necessitate special restrictions on access. Local rules (see para. 3.94 of GSR Part 3 [1]) or radiation work permits may be used for administrative control of access into such zones.

5.49. Warning signs that include appropriate information (such as information on radiation levels or contamination levels, the category of the zones, entry procedures or restrictions on access time, emergency procedures and contacts in an emergency) should be displayed at access points to controlled areas and, where appropriate, specific zones and other appropriate locations within controlled areas. Persons crossing a zone boundary should immediately be made aware (e.g. by the use of different wall colours) that they have entered another zone in which dose rates or contamination levels, and thus the working conditions, are different.

5.50. Paragraph 3.91 of GSR Part 3 [1] states that the operating organization:

> "shall designate as a supervised area any area not already designated as a controlled area but for which occupational exposure conditions need to be kept under review, even though specific measures for protection and safety are not normally needed."

5.51. Paragraph 3.90 of GSR Part 3 [1] states (citations removed) that the operating organization:

> "taking into account the nature, likelihood and magnitude of exposures or contamination in the supervised areas:
>
> (a) Shall delineate the supervised areas by appropriate means;
>
> (b) Shall display approved signs, as appropriate, at access points to supervised areas;
>
> (c) Shall periodically review conditions to assess whether there is any need for further measures for protection and safety or any need for changes to the boundaries of supervised areas."

Approval of work involving radiological hazards

5.52. A system of formal approvals for undertaking work involving radiological hazards should be implemented by the operating organization. The application for such an approval also enhances the awareness of personnel of the radiological hazard associated with the work. Applications should include a justification for the work. Appropriate criteria for such applications can be developed to avoid the need for a large number of formal approvals. The way to obtain an approval should be specified in a procedure that includes the roles and responsibilities.

5.53. *Work permits:* A system of work permits (see paras 3.94–3.96 of GSG-7 [26]) to perform certain tasks that involve radiological hazards or other hazards should be implemented, as appropriate, by the operating organization. Work permits should be subject to approval by appropriate personnel within the research reactor and may, in relation to controlling radiation exposure, include the following

information:

(a) Identification, location and description of the work;

(b) Identification of the responsible persons;

(c) Identification of workers to be involved in the work and their previous occupational exposure;

(d) Qualification of the workers;

(e) Data for exposure control (e.g. dose targets, maximum exposure time, actual doses received, time in and out);

(f) Radiological conditions, including any survey results;

(g) Arrangements for individual monitoring;

(h) Protective clothing and equipment;

(i) Special precautions or instructions;

(j) Actions on completion of the work;

(k) Provisions for restoration of the normal configuration;

(l) Signatures of workers stating that they have read and understood the necessary precautions;

(m) Approval signatures.

5.54. *Experiment approvals:* All new experiments are required to go through a process that includes review and approval by the safety committee and the regulatory body, as appropriate: see para. 4.27(c) and Requirement 83 of SSR-3 [1]. Further recommendations are provided in SSG-24 (Rev. 1) [12]. The review is required to include an assessment of the radiological hazards likely to arise during normal and abnormal operation of the experiment (including commissioning, decommissioning, predisposal management and disposal of waste and activated components): see paras 7.99–7.101 of SSR-3 [1].

5.55. *Routine irradiation approvals:* For routine operations, such as isotope production or irradiation of samples for activation analysis, the operating organization should develop a practical procedures, such as a simple irradiation form containing only the most important information. Such routine operations should be subject to approval by appropriate personnel within the research reactor. Non-routine irradiations should be treated as a new experiment (see para. 5.53).

5.56. *Modifications to the facility:* The safety implications of any modification to the facility are required to be examined: see Requirement 83 of SSR-3 [1]. Implementation of any modifications may need the review and approval of the safety committee and the regulatory body. Further recommendations are provided in SSG-24 (Rev. 1) [12].

5.57. *Procedures:* Procedures are required to be prepared and approved for all safety related activities and operations, and for radiation protection activities: see paras 7.45 and 7.58 of SSR-3 [1]. These

procedures (and any revisions) should be subject to review and approval, as appropriate (i.e. by the reactor manager, safety committee and regulatory body). The management system for the research reactor should include a process to ensure that current and valid procedures are in use. Further recommendations are provided in DS509D [7], GS-G-3.5 [23] and GSG-16 [24]).

OPTIMIZATION OF THE PROTECTION OF THE PUBLIC

5.58. A dose constraint for public exposure is required be established for a research reactor, as described in paras 2.4–2.6 of this Safety Guide. This dose constraint should be used by the operating organization as a starting point for the process of optimization of the protection of the public. The results of the optimization process should be the basis for the establishment of authorized discharge limits (see paras 5.13 and 5.14 of GSG-9 [16]).

Limitation of discharges

5.59. The regulatory body is required to establish or approve authorized limits on discharges for a research reactor: see para. 3.123 of GSR Part 3 [2]. Authorized limits should be set in terms of the discharges per unit time (i.e. weekly, monthly and annually, as appropriate).

5.60. The operating organization is required to ensure that the authorized limits on discharges from a research reactor are not be exceeded: see Requirement 59 and para. 6.211(f) of SSR-3 [1]. In addition, the reactor management[2] may set its own administrative control levels (see paras 5.11–5.13), derived on the basis of optimization and within the authorized limits. Exceeding a dose constraint should trigger actions by the operating organization (such as an investigation and administrative measures). Exceeding a dose constraint may also initiate actions by the regulatory body: however, exceeding a dose constraint should not represent a regulatory infraction, as would be the case if the dose limit were to be exceeded (see para. 5.17 of GSG-9 [16])

Record keeping for releases and waste

5.61. Releases of gaseous and liquid radioactive material are required to be recorded in accordance with Requirement 82 of SSR-3 [1]: these records should demonstrate compliance with authorized limits on discharges. Records are required to be kept of any radioactive waste that is generated: see para. 7.115 of SSR-3 [1]. Such records should include the planned duration and location of any temporary on-site storage of waste.

6. RADIOACTIVE WASTE MANAGEMENT IN THE OPERATION OF A RESEARCH REACTOR

[2] The reactor management comprises members of the operating organization to whom the responsibility and the authority for directing the operation of the research reactor have been assigned.

6.1. Gaseous, liquid and solid radioactive waste is generated in various types and amounts from the operation of research reactors. The nature and the amount of such waste depend on factors including the following:

(a) The type of research reactor;

(b) Specific design features;

(c) Operating procedures and practices;

(d) Irradiation of various targets, including those for radioisotope production, neutron activation analysis;

(e) The maintenance programme;

(f) Modifications to the facility and activities to extend the lifetime of the research reactor;

(g) Refuelling and operational occurrences;

(h) Operating history of the facility;

(i) Integrity of the fuel.

6.2. Requirement 85 of SSR-3 [1] states that "**The operating organization for a research reactor shall establish and implement a programme for the management of radioactive waste.**" In accordance with Requirement 12 of SSR-3 [1], a graded approach needs to be applied to this programme.

6.3. The programme for the management of radioactive waste at a research reactor needs to include provisions for the following:

(a) Keeping the generation of radioactive waste to the minimum practicable, in terms of both activity and volume, by using suitable technology;

(b) Possible reuse and recycling of materials;

(c) Appropriate classification and segregation of waste on the basis of its physical, chemical and radiological properties, and maintenance of an accurate inventory for each radioactive waste stream, with account taken of the available options for clearance, predisposal management and disposal;

(d) Collection, characterization and safe storage of radioactive waste;

(e) Adequate storage capacity for the radioactive waste expected to be generated;

(f) Ensuring that the radioactive waste can be inspected and retrieved at the end of the storage period;

(g) Pretreatment, treatment and conditioning of radioactive waste to ensure safe storage and disposal;

(h) Safe handling and transport of radioactive waste;

(i) Adequate control of discharges of effluents to the environment;

(j) Monitoring of effluent discharges and the environment;

(k) Maintaining facilities and equipment for the collection, processing and storage of waste to ensure safe and reliable operation;

(l) Monitoring the status of the means of confinement for the radioactive waste in the storage location;

(m) Monitoring for possible changes in the characteristics of radioactive waste by means of inspection and regular analysis, especially if storage is continued for extended periods;

(n) Initiating, as necessary, research and development activities to improve existing methods for processing radioactive waste or to develop new techniques and to ensure that suitable procedures are available for the retrieval of stored radioactive waste;

(o) Adopting and implementing corrective actions on the basis of the results of monitoring.

6.4. General requirements for the predisposal management of radioactive waste are established in GSR Part 5 [3], and specific recommendations on meeting these requirements are provided in SSG-40 [17]. Recommendations on safety assessment and development of a safety case for radioactive waste management activities and facilities are provided in GSG-3 [27].

MANAGEMENT OF THE GENERATION OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

Gaseous radioactive waste

6.5. The typical sources of gaseous radioactive waste generated during the operation of research reactors include the following:

(a) Gaseous radioactive elements or compounds from the pools, coolant systems, irradiation facilities and experimental devices;

(b) Airborne radioactive material produced in ancillary facilities, including fume cupboards and decontamination areas.

6.6. The generation of gaseous radioactive waste should be kept to the minimum practicable by adopting suitable measures, such as the following:

(a) The areas with potential for the generation of gaseous radioactive waste should have provisions for the renewal of air or the cover gas and for capturing the waste using suitable filters;

(b) The ventilation system of the controlled areas outside the containment or means of confinement should include high efficiency particulate air (HEPA) filters and charcoal beds or demisters prior to discharge to the stack;

(c) An atmosphere of an inert gas should be used for the transfer and cooling of irradiation targets;

(d) Levels of coolant impurities should be kept as low as practicable;

(e) Maintenance activities should be carefully planned to reduce the possibility of leakage of gaseous waste.

Liquid radioactive waste

6.7. The typical sources of liquid radioactive waste generated during the operation of research reactors include the following:

(a) Cooling water blowdown;

(b) Primary system drains (in the case of light water reactors);

(c) Liquid waste from the demineralized water plant;

(d) The drain of the ventilation water system (e.g. from condensation or from scrubbers or chillers within the system);

(e) Demineralized wastewater recovered from the drainage of large equipment in maintenance operations;

(f) Washbasin and shower liquids;

(g) Floor drain liquids;

(h) Liquids from laboratories (these can be radioactive or non-radioactive).

6.8. The generation of liquid radioactive waste at a research reactor should be kept to the minimum practicable by adopting suitable measures, such as the following:

(a) The proper selection of reactor materials, for example by avoiding materials containing cobalt;

(b) Reducing leakage from the various systems (see also para. 6.202 of SSR-3 [1]);

(c) Chemical adjustment of the coolant to minimize reactions with the reactor materials and to minimize deposits;

(d) Planning and performing maintenance work with due care and with particular emphasis on precautions to avoid the spread of contamination;

(e) Taking precautions to avoid contamination of equipment and building surfaces to reduce the need for decontamination;

(f) Optimizing decontamination procedures;

(g) Reducing the production of secondary waste by the appropriate selection of waste processing methods.

Solid radioactive waste

6.9. The typical sources of solid radioactive waste generated during the operation of research reactors

include the following:

(a) Irradiated target cans;

(b) Used irradiation rigs and reactor components (e.g. thermocouples);

(c) Neutron beam guide tubes;

(d) Used control rods;

(e) Waste arising from the pool service area;

(f) Ventilation system waste (charcoal filters, HEPA filters);

(g) Spent ion exchange resins;

(h) Cleaning materials and used personal protective items;

(i) Laboratory waste (e.g. gloves, tissue paper, disposable glassware);

(j) Contaminated items arising from maintenance and other works.

Some of the items listed above might not be applicable for subcritical assemblies.

6.10. The generation of solid radioactive waste at a research reactor should be kept to the minimum practicable by adopting suitable measures, such as the following:

(a) Careful planning and performance of maintenance work;

(b) Careful control of the packaging and handling of radioactive material;

(c) Avoiding the generation of secondary radioactive waste by, for example, placing restrictions on packaging and other unnecessary material being taken into controlled areas;

(d) Efficient operation of processing systems for gaseous and liquid radioactive waste;

(e) Effective procedures for the control of contamination and the use of effective methods of decontamination;

(f) Adopting good segregation practices, including clearance of materials, at the point of waste generation;

(g) Selection of materials that do not easily become activated (e.g. use of pure plastic for the target carriers used in pneumatic rabbit systems), and reuse and recycling of materials wherever practicable (e.g. use of titanium cans).

OPTIMIZING PROTECTION IN THE MANAGEMENT OF RADIOACTIVE WASTE

6.11. In accordance with Requirement 34 of SSR 3 [1], the operating organization is required to ensure that the exposure of workers and members of the public due to waste management operations are as low as reasonably achievable by optimizing the radiation protection arrangements for the handling, treatment, transport, storage and transfer of radioactive waste. This should be achieved by adopting

suitable measures, including the following:

(a) Identification of all credible exposure pathways (for workers and for the public) associated with activities and facilities for the management of radioactive waste;

(b) Following documented procedures for the handling, pretreatment, treatment, conditioning, transport, storage and disposal of all radioactive waste generated at the research reactor;

(c) Use of suitable protective equipment and shielding, and implementation of a radiation monitoring programme (see Section 7);

(d) Use of adequate ventilation for the management of gaseous effluents and the control of releases during operational states and in accident conditions;

(e) Clearly specifying limitations on discharges for each waste stream and authorized discharge point, derived using reasonably conservative methods and modelling;

(f) Following documented operating procedures for radiation monitoring, and assessing results to demonstrate that discharges are within specified limits;

(g) Use of a documented system for demonstrating and reporting compliance with discharge limits;

(h) Use of designated work areas with appropriate shielding for handling unsealed radioactive waste;

(i) Following documented procedures on the periodic measurement of radiation levels outside radioactive waste stores.

PACKAGING AND CONFINEMENT OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

6.12. In accordance with Requirement 10 of GSR Part 5 [15], radioactive waste arising from the operation of research reactors is required to be packaged to provide an appropriate degree of confinement, to minimize the potential for migration or dispersion of radionuclides and to limit the external dose rate associated with the package.

6.13. With regard to the packaging of radioactive waste at a research reactor, the potential for migration or dispersion of radionuclides should be minimized and the external dose rate should be limited by adopting suitable measures, such as the following:

(a) Providing appropriate areas and facilities for waste handling and packaging.

(b) Following appropriate documented procedures for the segregation, treatment and packaging of radioactive waste, in accordance with para. 7.118 of SSR-3 [1]. These procedures should take into account the physical and chemical form and the pyrophoric or other hazardous nature of the waste, and the method of storage and eventual disposal.

(c) Applying documented procedures to ensure that containers are clearly labelled with the radiation warning symbol, a description of the radioactive contents (i.e. the radionuclide and the form of

the waste), the activity when packaged, other hazards of the waste, the date of packaging and the name of the person who is to be contacted for further information or in the event of an abnormal occurrence;

(d) Applying documented procedures for performing dose rate measurements at the surface of each package and at specified distances from the surface of each package and for recording the results, to ensure compliance with waste acceptance criteria (see Requirement 12 of GSR Part 5 [3]);

(e) Use of spillage trays of adequate volume.

(f) Complying with national and international guidance for the packaging and transport of radioactive waste.

6.14. Radioactive waste varies in terms of both its activity content and its chemical composition, both of which are important factors in determining suitable packaging and disposal options that satisfy the operating organization of the research reactor, the operating organizations of disposal facilities and the regulatory body. The parameters of key importance for one particular option for waste disposal may be different from the most important parameters for another option. Account should be taken of the important parameters for waste packages as described in Section 4 of Ref. [29].

STORAGE OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

6.15. Requirement 11 of GSR Part 5 [3] states:

> "**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. Due account shall be taken of the expected period of storage, and, to the extent possible, passive safety features shall be applied. For long term storage in particular, measures shall be taken to prevent degradation of the waste containment.**"

Recommendations on the storage of radioactive waste are provided in IAEA Safety Standards Series No. WS-G-6.1, Storage of Radioactive Waste [30].

6.16. Where radioactive waste is stored at a research reactor, protection and safety (including criticality safety, as necessary) over the entire time periods involved are to be considered. In the case of heat-generating waste, the need for cooling can pose additional demands on the waste package, and should also be considered. During storage at a research reactor, waste packages should be able to remain within specifications (e.g. waste acceptance criteria) and, if necessary, readily overpacked when subsequently retrieved for transport and disposal. Some ageing of the package is probably inevitable; consequently, it should be ensured that there is no excessive surface corrosion or other means of deterioration of the stored packages. For packages closed with a lid and seal arrangement, the seal might be especially susceptible to ageing and, in extreme cases, remedial action may be necessary. The material used for packaging may show effects due to radiation exposure, which affect the strength of the material. Such effects, as well as the biodegradation of organic materials, might result in the generation of gases. If

necessary, passive ventilation systems should be used to relieve internal pressure. Passive ventilation systems should be designed to minimize the entry of moisture and the release of radioactive waste from the container.

6.17. Storage facilities should be adequately located, designed, constructed, operated, secured and maintained to allow for safe custody of the waste packages and for the protection of persons and the environment against the radiological hazards associated with radioactive waste.

6.18. The capacity of the storage facilities for radioactive waste is required to be adequate to accept the maximum operational holdings anticipated from the research reactor in normal operation and following anticipated operational occurrences: see para. 6.27(c) of SSR-3 [1].

6.19. The location, design, construction, operation and maintenance of storage facilities for radioactive waste at a research reactor should have the following characteristics:

(a) The storage facilities should be appropriately located to minimize the impact of potential hazards (both natural hazards and human-induced hazards);

(b) The storage facilities should be sited above the groundwater level;

(c) Durable fire retardant and heat resistant materials should be used for the construction of the storage facilities (see also para. 6.50 of SSR-3 [1]);

(d) A conservative design approach should be used to limit external dose rates in and around the storage facilities to help ensure that individual doses remain within the dose constraints;

(e) The storage facilities should be designed to control any spread of contamination from any gaseous or liquid radioactive releases that might occur;

(f) The surfaces inside the storage facilities should be constructed for ease of decontamination;

(g) Enough space should be provided in the storage facilities for the stacking, sorting and visual inspection of packages;

(h) Provision should be made for the maintenance of adequate environmental conditions (e.g. temperature and humidity) in the storage facilities to ensure the proper conservation of waste packages;

(i) An adequate monitoring system (for storage areas air and for off-gases) should be provided;

(j) Provision should be made for the cooling of heat-generating waste if there is a potential for such waste to be present;

(k) Suitable equipment for the handling of waste packages in an emergency should be provided;

(l) The maintenance programme for the storage facilities should be documented, with records kept of all maintenance activities;

(m) Appropriate radiation warning signs should be displayed on each entrance to the storage facilities;

(n) Provision should be made for a system to extract any gases or vapours originating from stored radioactive waste;

(o) Provision should be made to prevent the dispersion of radioactive material in the event of a fire in or around the storage facilities.

6.20. The packaging of radioactive waste should be suitable for the type of storage and for the foreseeable time frame of storage at the research reactor.

6.21. Paragraph 7.118 of SSR-3 [1] states that "Written procedures shall be followed for the handling, processing, transport and storage of radioactive waste." This should include procedures for the following:

(a) The storage of short lived radioactive waste to allow it to decay to a level of activity at which it can be cleared from regulatory control in accordance with Requirement 8 and Schedule I of GSR Part 3 [2].

(b) The use of storage containers for short lived radioactive waste, including the need for clear labelling with details of the waste (e.g. radionuclide(s), activity, date of storage, responsible person).

(c) The keeping of an accurate inventory of all waste packages and containers and their contents that are in storage.

(d) The periodic inspection and monitoring of stored radioactive waste, storage facilities and their environment.

6.22. The written procedures and systems for the handling, pretreatment, treatment, conditioning, transport, storage and disposal of radioactive waste should be periodically reviewed.

6.23. Radiation monitoring and a visual inspection should be performed whenever waste packages are handled or moved (placed into storage, retrieved or transported on or off the site) to verify the protection of operating personnel, to prevent the accidental spread of contamination and to provide an additional inventory check.

DOCUMENTATION RELATING TO THE MANAGEMENT OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

6.24. Paragraph 7.119 of SSR-3 [1] states:

> "An appropriate record shall be kept of the quantities, types and characteristics of the radioactive waste processed and stored on the reactor site or removed from the reactor site for the purpose of processing, storage or disposal."

This record should also describe the location of the waste, and the safety and security measures associated with the radioactive waste. Further recommendations on record keeping for radioactive waste management are provided in GSG-16 [24].

6.25. A register or database should be kept that includes the following information in relation to each radioactive waste package:

(a) The radionuclide(s) and activity, the type of waste and the physical and chemical characteristics;

(b) The chain of custody and responsibility (including details of acceptance, movement, storage, discharge and disposal);

(c) The waste matrix for immobilization (if appropriate);

(d) The methods of treatment and/or conditioning of the waste;

(e) The identification number of the package.

6.26. Records associated with the inspection and monitoring of stored radioactive waste and the associated environmental monitoring should be maintained in accordance with documented procedures.

6.27. Paragraph 7.117 of SSR-3 [1] states:

> "Releases of liquid and/or gaseous radioactive effluents to the environment shall be monitored and the results shall be recorded in order to verify compliance with the authorized limits. They shall also be reported periodically to the regulatory body or another competent authority in accordance with its requirements."

6.28. Documentation detailing compliance with regulatory requirements should be properly maintained and easily retrievable.

CLASSIFICATION, CHARACTERIZATION AND SEGREGATION OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

6.29. The classification of radioactive waste assists in the development of management strategies and in the implementation of the radioactive waste management programme at a research reactor. Segregation of waste with different properties will also be helpful at any stage between the arising of the raw waste and its conditioning, storage, transport and disposal. The characterization of waste may also be performed for safety assessment purposes. and for quality management of waste and waste packages during conditioning.

6.30. Waste characterization should be performed in the various steps of waste management. The characterization process should include measurements of physical and chemical properties and parameters, identification of radionuclides and measurement of the activity content. Such measurements are necessary for monitoring the history of the radioactive waste or waste packages through the stages of conditioning, storage and disposal, and for maintaining records for the future and confirming the effectiveness of processing.

6.31. In general, the waste generator is responsible for waste characterization, which is used as the basis for acceptance for ultimate disposal. Requirement 9 of GSR Part 5 [3] states:

"**At various steps in the predisposal management of radioactive waste, the radioactive waste shall be characterized and classified in accordance with requirements established or approved by the regulatory body.**"

6.32. In accordance with para. 7.118 of SSR-3 [1], written procedures are required to be followed for the characterization of radioactive waste and its segregation, and for assigning the waste to a particular class.

6.33. Recommendations on the purpose, methods and approaches to the classification of radioactive waste are provided in IAEA Safety Standards Series No. GSG-1, Classification of Radioactive Waste [31].

6.34. The programme for radioactive waste management at a research reactor should include arrangements for the segregation of radioactive waste. For the purposes of determining arrangements for the handling, treatment and storage of radioactive waste, consideration should be given to the following:

(a) The origin of the waste;

(b) Criticality safety;

(c) Radiological properties (e.g. half-life, activity and activity concentration of nuclides, dose rates);

(d) Other physical properties (e.g. size and mass, compactibility, solubility);

(e) Chemical properties (e.g. corrosion resistance, combustibility, gas generation properties);

(f) Biological properties (e.g. biological hazards);

(g) Intended methods of processing, storage and disposal.

6.35. The IAEA waste classification distinguishes materials that can be exempted or cleared from regulatory control, very low level waste, low level waste, intermediate level waste and high level waste, with a further class for very short lived waste [31]. Special care should be taken with the handling of waste containing alpha emitting radionuclides, which could arise from failed fuel. Waste containing inflammable, pyrophoric, corrosive or other hazardous materials should be handled with care. Mixing of such types of waste with other waste should be avoided.

6.36. Individual consideration should be given to the processing options for gaseous radioactive waste arising from pools, cooling systems, irradiation facilities and experimental devices, together with gaseous waste produced in auxiliary facilities, such as fume cupboards in active laboratories and decontamination areas.

6.37. Solid radioactive waste should be segregated in accordance with its nature and activity concentration. Waste such as sludge, cartridge filters, failed irradiated targets for radioisotope production, irradiated canning material, irradiated rigs, contaminated equipment and components, ventilation filters and other solid waste (e.g. paper, plastic, gloves) should be segregated in accordance

with the type of treatment and conditioning (e.g. compaction, incineration, immobilization) adopted for its processing.

PROCESSING OF RADIOACTIVE WASTE AT A RESEARCH REACTOR

6.38. The processing of radioactive waste at a research reactor may yield effluent that is suitable for authorized discharge, material suitable for some other authorized use, and/or material that meets the criteria for clearance from regulatory control (see Schedule I to GSR Part 3 [2]). Requirements for the different elements of waste processing — pretreatment, treatment and conditioning — are established in GSR Part 5 [3]. An assessment of the wastes that will be generated should be performed as part of the safety assessment for the research reactor.

6.39. The radioactive waste generated by the operation of the research reactor is required to be processed in accordance with written procedures: see para. 7.118 of SSR-3 [1].

6.40. Paragraph 4.13 of GSR Part 5 [3] states that "Waste has to be rendered into a safe and passive form for storage or disposal as soon as possible."

6.41. Waste packages resulting from the conditioning of radioactive waste should meet the waste acceptance criteria for the handling, transport, storage and disposal of radioactive waste

Gaseous radioactive waste

6.42. Gaseous radioactive waste arises when air becomes contaminated or when gases that are present or used in the facility become contaminated by radioactive aerosols, vapours or gases, or when constituents of a gas become activated in passing through the reactor core or in close proximity to it: see para. 6.6.

6.43. The air within a research reactor building can become contaminated from the leakage of gases or vapours from systems containing radioactive fluids, or from the resuspension of fine particulate material that exists within the facility as surface contamination, or from the handling of fine particulate material. Ventilation systems (see Requirement 64 of SSR-3 [1]) should include provisions for continuous monitoring of air contamination and, if necessary, the air should be treated before discharge from the facility. Such treatment may take the form of filtration for particulates and aerosols and/or adsorption for volatile species and gaseous contaminants. It should be ensured that gaseous contaminants are not present in quantities that would warrant special treatment.

6.44. Certain compartments or discrete pieces of equipment in the research reactor may have localized ventilation owing to the potential for airborne contamination. The air to be treated by such localized ventilation systems may also be treated by filtration and/or adsorption, either continuously or whenever monitoring indicates it to be necessary.

6.45. It may be necessary for certain systems within the research reactor to have off-gas systems to remove gases in order to control pressure within the system, to clean radioactive gases from the system

or to remove gases prior to systems being opened for operational reasons or for maintenance purposes. Such off-gas systems might contain significant amounts of contamination and before any maintenance work is undertaken on such systems it may be necessary to allow time for radioactive decay, or else to arrange for decontamination, adsorption of volatile species by particulate filtration, or the use of other chemical and physical cleaning methods

6.46. Treated air can generally be discharged from the research reactor following appropriate monitoring; process gases can be either reused or discharged, again with the necessary monitoring and control.

6.47. An assessment of the gaseous wastes that will be generated should be performed as part of the safety assessment for the research reactor. This assessment should be used in the design of the gaseous waste treatment systems, especially in terms of the necessary capacity and efficiency of such systems, with consideration being given to the properties of gaseous waste, such as corrosiveness, flammability or explosiveness, that might affect the system.

6.48. In assessing any changes to equipment or modes of operation of the facility or experiments introduced into the facility, due consideration should be given to any influences on the types and amounts of gaseous radioactive waste that might arise and the possible effects on the capacity of the treatment systems for gaseous radioactive waste.

Liquid radioactive waste

6.49. In processing liquid radioactive waste, the amount of waste to be treated, the radionuclides present, the activity concentration, the concentrations of particulates, the chemical compositions and the possible presence of corrosive substances should be taken into account.

6.50. For the effective processing of liquid radioactive waste at a research reactor, the following should be considered:

(a) The choice of processing option should be made with careful consideration given to all relevant factors, including occupational exposure and public exposure, and the generation of secondary waste.

(b) Waste of higher activity should not be diluted with lower activity waste. Liquid radioactive waste containing dissolved or dispersed solids of higher activity should not be mixed with waste containing solids of lower activity.

(c) The chemical compatibility of different waste streams and of the equipment for processing the waste should be considered. The possible incompatibility of radioactive waste with components of the treatment and conditioning plant (e.g. due to the potential for corrosion) should be taken into consideration and the chemical composition of the waste should be kept under strict control during its processing.

(d) If liquids are recycled after treatment and conditioning, attention should be paid to the possibility

of chemical cross-contamination so as to avoid unnecessary processing.

(e) If radioactive waste outside the normal range of composition is to be processed, consideration should be given to flushing the equipment before it is returned to normal operation.

(f) When processing waste, strict control over relevant process parameters should be maintained in accordance with the arrangements for protection and safety.

6.51. Aqueous liquid radioactive waste should be considered for processing in terms of chemical adjustment (pretreatment) and activity removal (treatment) [13]. Radioactive waste containing boric acid or organic matter may need special processing. Non-aqueous liquid radioactive waste such as oils should be segregated for separate treatment.

6.52. For the conditioning of liquid waste, suitable material should be used for the packaging, the waste containers and the matrix. The packaging and the container should be properly filled, closed and labelled to produce a waste package that is suitable for handling, transport, storage and disposal.

Solid radioactive waste

6.53. Solid radioactive waste may be inhomogeneous, and it should be ensured that any sampling before processing is representative so as to confirm compatibility with the intended process, and appropriate arrangements should be made for such sampling.

6.54. A wide range of processes are available for producing an acceptable form of solid waste from a research reactor. If possible, processes with high volume reduction should be applied, using proven techniques, such as compaction and incineration.

TRANSPORT OF RADIOACTIVE WASTE

6.55. Radioactive waste may be processed within the research reactor or at an on-site or off-site facility, depending on the local arrangements. Appropriate procedures and arrangements should be established by the operating organization for the on-site movement of radioactive waste. Requirements for the transport of radioactive waste off the site are established in IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material [32].

6.56. The radiation protection programme (see para. 5.2) should include handling procedures for packaged radioactive waste, to minimize the exposure of personnel and to ensure the safety of the on-site movement and off-site transport of radioactive waste.

CONTROL OF RADIOACTIVE DISCHARGES AND COMPLIANCE MONITORING AT A RESEARCH REACTOR

6.57. The operating organization is required to apply to the regulatory body for authorization to discharge gaseous and liquid radioactive waste in operational states: see para. 3.123 of GSR Part 3 [2]. Recommendations on the establishment of such discharge limits are provided in GSG-9 [16].

6.58. The purpose of setting authorized limits on discharges is to ensure that radiation doses to members of the public due to such discharges do not exceed the dose constraint (based on an assessment of the dose to the representative person) and that such doses to members of the public are as low as reasonably achievable. The operating organization should consider expected discharges from all operational states of the research reactor and from all possible future changes in operation in the application to the regulatory body for a discharge authorization.

6.59. The proposal for discharge levels should include an assessment of the expected radiological consequences, made by means of appropriate modelling. Expected doses to the representative person should be estimated. It may be useful to conduct a lifestyle survey to determine those members of the public who would be the most exposed as a result of the discharges.

6.60. The regulatory body, after proper consideration of the application from the operating organization of the research reactor, should set the authorized limits on discharges of liquid radioactive waste and gaseous radioactive waste. The operating organization is required to ensure that all discharges are below these authorized limits: see para. 7.107 of SSR-3 [1].

6.61. Compliance with authorized limits on discharges is required to be demonstrated by means of monitoring, involving approved methods of sampling and measurements: see para.7.117 of SSR-3 [1]. The results of monitoring are also required to be recorded.

6.62. Paragraph 7.113 of SSR-3 [1] states:

> "If…the authorized limits for radioactive releases are exceeded, the reactor manager, the safety committee, the regulatory body and other competent authorities shall be informed in accordance with the requirements."

Arrangements should also be established for dealing with any abnormal releases that exceed the administrative control levels, including, as necessary, notifying the reactor manager, safety committee, regulatory body and other competent authorities, and assessing any impact on members of the public or the environment.

6.63. In the event of a discharge exceeding the authorized limits, the operating organization should investigate the cause and should take appropriate measures, including the following:

(a) Suspend the discharge operation and take corrective actions;

(b) Estimate the amount of radioactive material released;

(c) Record relevant details of the discharge;

(d) Notify the regulatory body following documented procedures;

(e) Investigate and identify the cause of non-compliance;

(f) Correct the cause of non-compliance to prevent its recurrence.

Depending on the magnitude of the discharge, the emergency response arrangements for the research reactor should be implemented, as necessary: see Section 10.

7. RADIATION MONITORING PROGRAMME AT A RESEARCH REACTOR

7.1. The operating organization is required to provide workplace monitoring and environmental monitoring as part of the radiation protection programme for a research reactor: see para. 7.10(b) of SSR-3 [1]. To meet this requirement, an appropriate surveillance programme should be established, maintained and kept under review for the systematic evaluation of the radiological conditions at the research reactor, with the following objectives:

(a) To monitor individual exposures;

(b) To control any environmental impacts;

(c) To assess trends in radiological parameters;

(d) To detect degradation in systems, such as corrosion and leaks.

7.2. The monitoring systems should have a sufficient sensitivity, dynamic range and accuracy to ensure that changes to normal conditions are detected and to reliably indicate whether any administrative control levels (see paras 5.1–5.13), dose constraints, dose limits or authorized limits on discharge are likely to be exceeded. Recommendations on monitoring instrumentation are provided in Section 8.

7.3. Records of the findings of the monitoring programme at the research reactor should be kept and relevant data should be made available to site personnel.

TYPES OF MONITORING SURVEY AT A RESEARCH REACTOR

7.4. The types of survey that are appropriate in a research reactor depend on the radiological conditions that may be present, and on the objectives of the survey. The factors to be considered in planning and performing monitoring surveys include the following:

(a) The nature and magnitude of the radiological hazard in the workplace, and the type and duration of the work undertaken;

(b) Requirements of the regulatory body and, where relevant, other competent authorities;

(c) The radiation sources present in the workplace;

(d) The radionuclides present and their physical forms (solid, liquid, gaseous, airborne or particulate);

(e) Chemical forms of the radionuclides (elemental or compound);

(f) The nature of any contamination (loose or fixed, detected by smears or direct counting);

(g) Background radiation levels and contamination levels;

(h) Arrangement for monitoring workers and the workplace.

7.5. Three types of survey may be conducted for radiation protection purposes: routine surveys, task related surveys and special surveys.

Routine surveys

7.6. Routine surveys should be conducted to assess dose rates and levels of surface contamination and airborne contamination in the workplace. The survey should demonstrate that the work environment is satisfactory for continued operation and that no change has taken place that would call for a reassessment of either the designation of areas (see paras 5.42–5.50) or operating procedures. Factors that might affect the type and frequency of each routine surveys include the following:

(a) Typical dose rates and contamination levels normally present in the workplace;

(b) The degree of occupancy;

(c) Quantities and types of radionuclides encountered during the work;

(d) Types of work or experiments;

(e) Changes in radiological conditions in the area.

Task related surveys

7.7. Task related surveys should be conducted to supply information about a particular task or operation (e.g. maintenance of equipment, handling of samples, transport of spent fuel) and to provide, if necessary, a basis for immediate decisions on the execution of the task.

Special surveys

7.8. Special surveys should normally be undertaken as follows:

(a) At the commissioning stage for a new research reactor;

(b) After modifications to existing facilities or current procedures;

(c) In the event of new experiments or new shield configurations;

(d) During periodic but infrequent surveillance activities;

(e) When operations are being performed under abnormal circumstances, such as those following an incident or an accident.

7.9. Special surveys may also be undertaken to address specific problems, such as the following:

(a) Spills or abnormal contamination of the workplace;

(b) Contamination of persons;

(c) Cases where high levels of airborne contamination or loose surface contamination are suspected;

(d) When routine monitoring reveals unusual or abnormal conditions, such as the occurrence of a small area of elevated activity (‘hot spot’).

7.10. For each type of survey, administrative control levels (see paras 5.1–5.13) should be established together with the action needed if the level is exceeded. The survey results should be checked to determine whether they are consistent with the type of work being performed and, if appropriate, with the records of individual monitoring (see paras 7.19-7.34).

WORKPLACE MONITORING AT A RESEARCH REACTOR

Area monitoring

7.11. Instruments of various types, both fixed and portable, are used at research reactors. Various types of instrument are described in paras 7.12–7.19, together with the types of measurement for which they are suitable.

Dose rate monitoring

7.12. Fixed radiation monitors (gamma radiation and neutron radiation) with local alarms are required to be installed in research reactors to indicate changes in dose rates: see para. 6.193(a) and (b) of SSR-3 [1]. Most systems include an indication of the dose rate at each location and failure detection capability. Redundancy should be considered in the placing of fixed radiation monitors. The dose rate indication and alarms are required to be relayed to the main control room: see para. 6.193(b) of SSR-3 [1].

7.13. Fixed gamma radiation monitors should be installed in areas where the radiological conditions might change frequently or unexpectedly. Such monitors would normally be installed on the top of the reactor, near beam ports, at the thermal column, at the fuel area, radioactive effluent areas and waste storage areas, near the ion exchange system and other filter systems, at locations where handling for experiments is performed and in the main control room.

7.14. Portable radiation monitors include gamma dose rate meters and neutron dose rate meters. The use of such portable instruments is helpful, provided that the place and time of measurements are specified and recorded.

Contamination monitoring

7.15. *Surface contamination:* Various instruments and methods may be used for monitoring surface contamination in accordance with para. 6.193(f) and (g) of SSR-3 [1], including the following:

(a) Direct measurements of surface contamination (alpha and/or beta–gamma contamination) may be made with portable instruments.

(b) Indirect measurements may be made using smear tests. The smear samples may be measured in an area of low background radiation using a shielded counter.

(c) Frisking probes for beta–gamma surface contamination are typically used at control points, such as exits from potentially contaminated areas, to monitor personnel and equipment.

(d) Personnel contamination monitors ('gate monitors') located at exits, including:

  (i) Portal (pass-through type) monitors;

  (ii) Hand and foot monitors.

(e) Special purpose monitors for specific uses. Examples are laundry monitors, solid waste material monitors and floor monitors.

7.16. *Airborne contamination:* Monitoring for airborne contamination is required in certain areas of a research reactor: see para. 6.193(c) of SSR-3 [1]. Such monitoring should be performed by means of continuous on-line measurement or by sampling and off-line analysis. Whenever there is a potential for a significant release of airborne contamination, monitors should be installed in the workplaces and/or the ventilation system. Area gamma monitors may be used at locations of interest to warn of increasing airborne contamination. In such cases, care should be taken to distinguish between dose rates from airborne contamination and those from other possible sources.

7.17. *Monitoring for contamination in liquids:* In accordance with para. 6.193(d), such monitoring may be performed by continuous on-line measurement or by sampling and off-line analysis. Gamma monitors located close to pipes carrying liquids that might be contaminated may be used if the influences of other sources are known and can be taken into account.

7.18. *Monitoring in relation to the transport of radioactive material:* Any transport package containing radioactive material that is received at or dispatched from the research reactor should be checked for both external surface contamination and dose rates, which are required to conform to the relevant limits specified in SSR-6 (Rev. 1) [32]. The results of the measurements should be recorded, and copies should be included in the transport documents and archived in accordance with regulatory requirements.

7.19. *Monitoring of solid waste:* Stored solid waste should be monitored for dose rates and for surface contamination on the external surfaces of waste packages. If contamination is detected, the radionuclide(s) should be identified. A record keeping system is required to be established and maintained for all workplace monitoring (see para. 7.119 of SSR-3 [1]) including for accountability and traceability of solid waste.

INDIVIDUAL MONITORING AT A RESEARCH REACTOR

7.20. Requirements for the assessment of occupational exposures (by means of individual monitoring or by workplace monitoring, as appropriate) are established in paras 3.99–3.102 of GSR Part 3 [2]. Supporting recommendations are provided in paras 3.107–3.111 and 3.116–3.40 of GSG-7 [26].

7.21. Paragraph 3.100 of GSR Part 3 [2] states (footnote omitted):

"For any worker who usually works in a controlled area, or who occasionally works in a controlled

area and may receive a significant dose from occupational exposure, individual monitoring shall be undertaken where appropriate, adequate and feasible. In cases where individual monitoring of the worker is inappropriate, inadequate or not feasible, the occupational exposure shall be assessed on the basis of the results of workplace monitoring and information on the locations and durations of exposure of the worker."

7.22. Paragraph 3.101 of GSR Part 3 [2] states:

"For any worker who regularly works in a supervised area or who enters a controlled area only occasionally, the occupational exposure shall be assessed on the basis of the results of workplace monitoring or individual monitoring, as appropriate."

7.23. The operating organization of a research reactor should make arrangements with dosimetry services for assessing individual doses (from external exposure and internal exposure, as appropriate) received by relevant persons at the research reactor. This monitoring may include the assessment of effective dose (i.e. for the whole body) and equivalent dose (e.g. to the hands).

7.24. The operating organization is required to maintain records of individual monitoring: see para. 3.103 of GSR Part 3 [2]. These records should be used to demonstrate compliance with regulatory requirements. These records may also be analysed to determine trends in exposure and future needs.

7.25. The necessary nature, frequency and precision of individual monitoring at a research reactor should be determined with due consideration of the magnitude and possible fluctuation of individual exposures, and the likelihood and possible magnitude of potential exposures.

Individual monitoring for external exposure

7.26. The assessment of individual external exposure is readily performed by means of individual monitoring. For routine monitoring, passive personal dosimeters should be worn. These dosimeters should be processed, and the results determined at appropriate intervals by an authorized or approved dosimetry service. Task related and special individual monitoring is normally performed with active personal dosimeters, often with additional warning functions.

Individual dosimetry for exposure to beta–gamma radiation

7.27. There are a variety of dosimeters available for individual monitoring for exposure to uniform whole body beta–gamma radiation; a description of the various types is presented in Appendix II of GSG-7 [26]. Particular care should be taken when personnel work near radiation beams, since exposures may be highly non-uniform.

7.28. Most passive beta–gamma dosimeters are suitable for routine issue over periods of one to three months. For the purposes of short term dose control, these dosimeters should be supplemented with a direct reading active personal dosimeter. If persons are working in areas where the radiation dose can vary widely in space or time, for example, during refuelling or maintenance activities, they should be provided with an active personal dosimeter incorporating an alarm that sounds if a pre-set dose or dose

rate is reached.

Individual dosimetry for exposure to neutron radiation

7.29. Dosimetry for neutron radiation is technically one of the most difficult types of dosimetry. A description of the various types of dosimeter available for the assessment of exposure to neutron radiation is presented in Appendix II of GSG-7 [26]. The following detectors are commonly used for neutron dosimetry:

(a) Nuclear track detectors;

(b) Solid state nuclear track detectors;

(c) Activation detectors;

(d) Thermoluminescent albedo dosimeters;

(e) Bubble detectors;

(f) Electronic neutron dosimeters.

7.30. Different types of dosimeters may be arranged to distinguish between doses from neutrons and other types of radiation, see GSG-7 [26] for details. In cases where neutron detectors of the type mentioned in para 7.29 are not available film dosimeters and thermoluminescent dosimeters may be arranged in such a way that a distinction is possible between doses due to neutrons and doses due to radiation of other types. In other cases,, 'rem meters' should be used for the assessment of neutron doses.

Extremity dosimetry

7.31. Finger and wrist dosimeters are available for monitoring exposure of the extremities. The operating organization should ensure that personnel are trained to wear the dosimeters appropriately by placing the detector on the extremity closest to the source.

Individual monitoring for internal exposure

7.32. Persons who work under conditions in which internal exposures could occur are required to be appropriately monitored: see para. 3.102 of GSR Part 3 [2]. This monitoring should be performed routinely or on an occasional basis, depending on the working conditions and the purpose of the monitoring.

7.33. The assessment of internal exposure (in terms of committed effective dose) from intakes of radionuclides is seldom straightforward. The dose due to internal exposure should be evaluated from the estimated intake of radionuclides and the use of an appropriate dose coefficient (dose per unit intake): see Schedule III of GSR Part 3 [2].

7.34. Paragraphs 7.34 and 7.35 summarize the methods commonly used for individual monitoring for internal exposure. A description of the various methods available for monitoring internal exposures is also provided in Appendix V of GSG-7 [26]. If none of the available methods of individual monitoring

is appropriate, internal exposures should be assessed on the basis of workplace monitoring.

7.35. *Bioassay techniques:* Bioassay involves taking biological samples (e.g. urine, faeces, nose swabs, breath) and measuring their activity. From the activity of the samples, the intake and the resulting internal dose can be estimated.

7.36. *Whole body monitoring:* Whole body monitoring is generally the preferred method of dose assessment for radionuclides that emit penetrating radiation. Whole body counters are used to measure the activity of radioactive material deposited in different parts of the body. From the measured activity, the intake and the resulting internal dose can be estimated. Whole body counting should be performed periodically on relevant operating personnel at a research reactor.

7.37. *Partial body monitoring:* Partial body monitoring may be used if radionuclides are concentrated in a specific organ (e.g. iodine in the thyroid gland). In such cases, relatively simple measuring equipment (e.g. a portable detector) may be used.

EFFLUENT MONITORING AT A RESEARCH REACTOR

7.38. The operating organization is required to ensure that releases of liquid and/or gaseous radioactive effluents to the environment are monitored, and the design of the research reactor is required to include stationary equipment for this purpose: see paras 6.193(e) and 7.117 of SSR-3 [1].

Source monitoring

7.39. Source monitoring refers to the measurement of discharges from and the dose rates around the research reactor. The operating organization should ensure that the source monitoring programme is capable of verifying compliance with the authorized limits on discharges as prescribed by the regulatory body. The monitoring of radioactive discharges may involve making measurements of specific radionuclides or gross activity measurements, as appropriate. Effluent measurements are required to be made before release or at the point of release (see para. 6.193(e) of SSR-3 [1]), for example, at the stack used for discharges to atmosphere or at the pipeline used for liquid discharges. For batch discharges, the radioactive contents should be characterized by the volume of the batch and the radionuclide composition of a sample taken from the homogenized batch prior to discharge.

7.40. For both airborne and liquid effluents, the following three types of measurement should be considered:

(a) On-line monitoring of discharges;

(b) Continuous sampling and laboratory measurements of the activity in the samples;

(c) Periodic sampling and laboratory measurements of the activity in the samples.

7.41. When selecting the sampling methods and measurement procedures for effluent monitoring at a research reactor, the following should be taken into account:

(a) The characteristics and amounts of discharged radionuclides and the sensitivity of the measurement system;

(b) The expected variation with time, if any, in the rate of discharge of the radionuclide(s);

(c) The possibility of unplanned releases of gaseous radioactive material to the atmosphere, which necessitate prompt detection and notification.

Monitoring of gaseous discharges

7.42. Radioactive material to be discharged to the atmosphere is generally measured at various points in the ventilation system of the research reactor. Measurements performed in the stack (stack monitoring) provide data on the actual discharges. Gaseous effluents are best monitored continuously. Spot or grab sampling may be used for the investigation of suspected problems rather than for routine monitoring. Stack monitors should preferably have a detector alarm linked to the ventilation control circuit to reduce ventilation or recirculate ventilation air. For research reactors, stack monitoring is often required by the regulatory body. The results of stack monitoring are required to be reported periodically to the regulatory body: see para. 7.117 of SSR-3 [1]. The results should be made available at the main control room for quick action in the event that a release is higher than the relevant administrative control level.

7.43. *Noble gas monitoring:* There are a number of systems for noble gas monitoring. In the most frequently used system, the air is filtered (to remove vapour and particles) and then passed through a large volume shielded chamber with a sensitive gamma radiation and/or beta radiation detector inside. The filtered gas may also be passed directly through an ionization chamber or a proportional counter. These systems should be calibrated with known concentrations of suitable radionuclides.

7.44. *Particulate monitoring:* Particulate activity in gaseous effluents is assessed by drawing air through an appropriate filter and measuring the activity on the filter. These systems may use either fixed or moving filter papers. The radionuclides deposited on the filter should be identified to enable an assessment of the source of the activity and the potential for reducing the release.

7.45. *Monitoring of special radionuclides:* Depending on the reactor type, the reactor power or installed experiments, the potential for release of other radionuclides, such as tritium and halogens, should also be considered, as follows:

(a) *Tritium:* For detecting tritiated water vapour in gaseous effluents, a trap may be used to collect moisture from the flowing air. Samples of the trapped moisture are periodically analysed for tritium. Other methods, such as coincidence and anticoincidence counting, have been developed and are available.

(b) *Iodine*: Charcoal (or zeolite) filled cartridges may be used as a collecting medium for measurements of halogen isotopes such as radioiodine. Care should be taken with the sampling devices to prevent plate-out effects. After sampling, the cartridges should be analysed by gamma

spectrometry. Iodine measurements would be needed only in abnormal situations, such as when leaking fuel elements might be present, or when experiments or irradiations are being conducted in which iodine might be released.

Monitoring of liquid discharges

7.46. The concentration and total activity of radionuclides in liquid effluents that might be radioactive should always be checked before their release in the environment. Usually they are checked by spot sampling of the tank contents (after homogenization) rather than by continuous monitoring. The analysis may include gross beta–gamma measurements, and possibly activity measurements of low energy beta emitters, tritium and alpha emitters, and identification of radionuclides by gamma spectrometry.

7.47. The chemical characteristics of liquid effluents that might affect transport processes (e.g. the pH may affect soil retention properties) should also be monitored.

ENVIRONMENTAL MONITORING AT A RESEARCH REACTOR

7.48. In accordance with para. 3.137 of GSR Part 3 [1], the operating organization of a research reactor is required to establish an environmental monitoring programme. This programme should be conducted in accordance with the requirements of the regulatory body. Recommendations on environmental monitoring programmes are provided in GSG-9 [16] and IAEA Safety Standards Series No. RS-G-1.8, Environmental and Source Monitoring for the Purposes of Radiation Protection [33].

7.49. A pre-operational programme for environmental monitoring should be conducted two to three years before the planned commissioning of the research reactor. The pre-operational programme should provide for the measurement of background radiation levels in the vicinity of the research reactor and their variation over the seasons. This programme should include the collection and radioanalysis of various samples, such as vegetation, air, milk, water, sediment, fish and environmental media collected from several fixed and identified locations off the site.

7.50. The operational programme of environmental monitoring should be conducted as an extension of the pre-operational programme. The samples taken during the operational programme should be similar to those taken in the pre-operational programme, although the collection intervals may be different (e.g. milk may be sampled more frequently and sediment less frequently). The operational programme should be reviewed in the light of experience and should be modified if necessary. The objectives of the environmental monitoring programme should include the following:

(a) To confirm the adequacy of control of effluent discharges;

(b) To correlating the results of environmental monitoring with data obtained from effluent monitoring;

(c) To check the validity of environmental models used in setting discharge limits;

(d) To assess trends in the concentration of radionuclides in the environment.

8. INSTRUMENTATION FOR THE RADIATION PROTECTION PROGRAMME AT A RESEARCH REACTOR

8.1. Requirement 57 of SSR-3 [1] states that "**Equipment shall be provided at a research reactor facility to ensure that there is adequate radiation monitoring in operational states and accident conditions.**"

The radiation protection programme at the research reactor (see Section 7 of this Safety Guide) determines the type and number of instruments needed for individual monitoring and area monitoring.

RADIATION MONITORING INSTRUMENTS FOR USE AT A RESEARCH REACTOR

Number of instruments

8.2. The factors to be considered include the following:

(a) Characteristics of the research reactor, such as the extent of supervised areas and controlled areas, and the types of experiment performed concurrently;

(b) Frequency of use for each type of instrument;

(c) Feedback from audits and experience of users of the instruments;

(d) The number of instruments out of service for repair or calibration;

(e) The need to dedicate some instruments for emergency use only.

Technical specifications

8.3. The following technical features of the radiation monitoring instruments should be considered:

(a) Suitability for the type of radiation monitored;

(b) Accuracy;

(c) Appropriate ranges and sensitivity;

(d) Energy response (tissue equivalency for dose rate meters);

(e) Linearity;

(f) Measuring time (time constant);

(g) Effects of ambient factors, such as pressure, temperature and humidity;

(h) Adequacy of power supply and indication of battery condition;

(i) Adequacy of directional response and suitability for the radiation field size (e.g. narrow beams);

(j) Response to radiation of other types;

(k) Fail-safe features on dose rate instruments;

(l) Testability (appropriate sources and methods should be available for performing functional checks on instruments);

(m) The need for adequate reliability and maintainability;

(n) As appropriate, instruments should be sealed to prevent the ingress of contamination and should have surfaces that are easy to decontaminate.

Human factors

8.4. Human factors may be an important aspect in the selection of radiation monitoring instruments. The following should be considered in assessing portable instruments:

(a) Ease of use and preference of the users;

(b) Readability of the display;

(c) Ergonomic factors, such as weight, balance, ease of carrying, and the design and location of switches;

(d) Robustness for typical use;

(e) Risk of data loss or distortion (i.e. if results are to be transferred to a plantwide data collection system).

Some of these aspects should also be considered in the selection of stationary radiation monitoring instruments.

MAINTENANCE AND CALIBRATION OF INSTRUMENTS AT A RESEARCH REACTOR

8.5. The operating organization of the research reactor should assign responsibilities for the maintenance and calibration of radiation monitoring instruments, in accordance with established procedures. For maintenance purposes (and also for the ease of use), the number of different types of radiation monitoring instrument should be minimized. The following factors should also be considered:

(a) The availability of spare parts and repair services should be ensured;

(b) Instrument calibration facilities should be available. All calibration sources used should be certified to a national standard. The periodicity of calibration for each type of instrument should be specified and recorded;

(c) The ease by which instrument calibrations can be performed, especially for fixed monitoring systems.

(d) Some instruments (e.g. noble gas monitors) need special facilities to allow accurate calibration. The calibration of such instruments should be performed periodically, in accordance with the

national regulations and standards.

8.6. A preventive maintenance programme for radiation monitoring instruments should be established. This should include consideration of the need for recalibration following any major repair of an instrument.

8.7. Further recommendations on the maintenance and calibration of radiation monitoring instruments — including for providers of calibration and testing services — are provided in GSG-7 [26].

9. ORGANIZATIONAL ASPECTS OF RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

9.1. The organizational structure of the research reactor should be such as to ensure proper fulfilment of the objectives for radiation protection.

APPLICATION OF THE MANAGEMENT SYSTEM TO RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

9.2. A documented management system that integrates safety, health, environmental, security, quality and economic objectives for the research reactor project is required to be developed: see Requirement 4 of SSR-3 [1]. General requirements for the management system are established in GSR Part 2 [22]. Specific recommendations are provided in GS-G-3.5 [23] and GSG-16 [24].

9.3. The management system is required to include all processes relevant to radiation protection (i.e. in accordance with the radiation protection programme): see para. 4.10 of SSR-3 [1]. The management system for a particular research reactor should reflect the number, complexity and safety significance of the activities undertaken.

9.4. Individuals within the operating organization who are responsible for the management system should have the authority and organizational freedom to identify problems and to initiate corrective actions. They should have direct access to such levels of management as are necessary to achieve the goals, strategies and objectives of the organization.

9.5. The management system should establish processes for review and assessment of the effectiveness of the radiation protection programme and for its continual improvement.

ORGANIZATIONAL STRUCTURE FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

9.6. The organizational structure of a research reactor depends on the size and the nature of the facility and on its management system, and generally includes groups for reactor operation and for radiation

protection, and reactor users. The operating organization should ensure that there are direct lines of communication for reporting radiation protection and radioactive waste management issues to the senior management.

9.7. The structure of the organization is required to be clearly defined, and the responsibilities of each person in the organizational structure are required to be assigned and documented: see Requirement 68 of SSR-3 [1]. Job descriptions should clearly specify the responsibilities for radiation protection that are associated with each position.

RESPONSIBILITIES RELATED TO RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

Regulatory body

9.8. In accordance with the legal framework of the State, the regulatory body is required to assume responsibilities, functions and activities in relation to protection and safety: see Requirements 2 and 3 of GSR Part 3 [2]. These responsibilities, functions and activities include: development of regulations and guidance documents; authorization of practices; inspection of authorized facilities; and enforcement of the laws and regulations. The regulatory requirements for radiation protection include dose limits for occupational exposure and for public exposure, the optimization of protection and safety, and qualification and training requirements for personnel, and performing audits.

The operating organization

9.9. The operating organization is responsible for ensuring that exposures are controlled and kept within limits, that protection and safety is optimized, and that an appropriate programme for radiation protection is established and implemented. For such a radiation protection programme to be effective, there should be a high level of commitment by the reactor management (see also Requirement 2 of GSR Part 2 [22]). Such a commitment should be demonstrated in the safety policy of the operating organization, the provision of resources and in fostering a strong safety culture: see paras 9.9–9.12 of this Safety Guide.

Safety policy

9.10. A clear and concise written policy on protection and safety is required to be established and implemented by the operating organization: see Requirement 3 of SSR-3 [1]. This policy is required to give safety the highest priority, overriding all other demands: see para. 4.4 of GSR Part 3 [1]. The policy should also demonstrate a commitment to the application of standards for protection and safety, with emphasis on the application of the optimization principle and the independence of the functions for radiation protection.

Provision of resources

9.11. The provision of resources for radiation protection at a research reactor is an essential part of the

commitment by management. These resources include adequate equipment and supplies, and the personnel necessary to maintain an effective radiation protection programme: see para. 4.15 of SSR-3 [1]. This includes the following:

(a) Appropriate and sufficient staffing;

(b) Programmes and facilities for training and retraining;

(c) A programme of individual monitoring;

(d) Provision of personal protective equipment and radiation monitoring instruments;

(e) Programmes for radiation monitoring in the research reactor and in the environment;

(f) Provisions for emergency preparedness and response;

(g) Provisions for operating procedures and operational limits and conditions;

(h) Provisions for appropriate documentation and record keeping.

Safety culture

9.12. The operating organization of a research reactor is required to foster a strong safety culture that encourages a questioning and learning attitude to safety and radiation protection, and that discourages complacency; see paras 4.1(f) and 7.9(l) of SSR-3 [1] and Requirement 5.2 of GSR Part 2 [22]. A strong safety culture should ensure the following:

(a) The operating environment and the attitude of all site personnel facilitate good radiation protection practice;

(b) Problems affecting protection and safety are promptly identified and corrected in a manner commensurate with their importance;

(c) Clear lines of authority for decisions on protection and safety are designated;

(d) Organizational arrangements and lines of communication are established that allow an appropriate flow of information on safety and radiation protection to and between all relevant levels in the operating organization.

Reactor manager

9.13. In accordance with Requirement 69 of SSR-3 [1], the reactor manager[3] is responsible for safely managing the research reactor in accordance with regulatory requirements. The reactor manager should act in accordance with the policies of the operating organization and is required to ensure that there are adequate and appropriately trained staff to perform all operational functions: see para. 7.14 of SSR-3 [1]. The reactor manager should ensure that requirements for radiation protection at the research reactor

[3] The reactor manager is the member of the reactor management to whom the direct responsibility and authority for the safe operation of the reactor are assigned by the operating organization and whose primary duties comprise the discharge of this responsibility.

are met.

Reactor safety committee

9.14. The reactor safety committee is an important part of the operating organization of the research reactor. Paragraph 7.26 of SSR-3 [1] states:

> "The reactor safety committee (or advisory group) shall advise the reactor manager on the safety aspects of the day to day operation and utilization of the reactor. In particular, the safety committee shall review the adequacy and safety of proposed experiments and modifications and shall provide the reactor manager with recommendations for action."

9.15. With regard to radiation protection, the reactor safety committee should review and advise on the following (see also para. 4.37 of SSR-3 [1]):

(a) The radiation protection programme;

(b) Aspects of radiation protection associated with the operation, maintenance, testing and utilization (including experiments) of the reactor;

(c) Procedures for radiation protection;

(d) Reports on accidents and incidents, and any follow-up actions taken to prevent their recurrence.

Radiation protection personnel

Radiation protection officer

9.16. The radiation protection officer is an individual who is technically competent in matters of radiation protection relevant to the research reactor and who is designated by the operating organization to oversee the implementation of the radiation protection programme. The radiation protection officer's functions may include the following:

(a) Providing advice and assistance to the reactor management in specifying and performing its responsibilities for the radiation protection programme;

(b) Reviewing for approval, in accordance with established policy, all procedures, work practices, manuals, equipment, proposed changes to the design of the facility, training programmes and reactor experiments, as they pertain to radiation protection;

(c) Providing advice and assistance to site personnel in matters of radiation protection;

(d) Assessing the effectiveness and efficiency of all aspects of the operation of the research reactor relating to radiation protection;

(e) Providing radiation protection services in accordance with the policy of the management and the needs of the research reactor;

(f) Guiding and overseeing the application of the optimization principle;

(g) Implementing an internal programme of review and verification (e.g. audits) to ensure that approved procedures relating to radiation protection have been properly documented and are being performed accordingly;

(h) Initiating corrective actions when deviations from approved procedures are observed.

9.17. The radiation protection officer should be responsible for assessing the quality of any external radiation protection services provided to the research reactor.

Other radiation protection personnel

9.18. Radiation protection technicians perform most of the routine radiation protection tasks, such as radiation monitoring; senior technicians may also interpret the results of such monitoring and make recommendations on the basis of the results.

9.19. Some operating organizations may assign specific radiation protection functions to other staff, such as reactor operators, maintenance personnel or researchers. In such cases, these functions should be clearly specified, documented and audited by the radiation protection officer.

QUALIFICATION, TRAINING AND RETRAINING OF PERSONNEL FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

Qualification of personnel in radiation protection and radioactive waste management

9.20. Recommendations on the recruitment, training and qualification of personnel — including an example training course curriculum for research reactor operators — are provided in DS509E [8]. The regulatory body should provide guidance on requirements for the qualification of operating personnel in relation to radiation protection and radioactive waste management and, where appropriate, should review and approve any proposals made by the operating organization.

9.21. For each position in the operating organization, a job description should be prepared that includes the minimum qualifications and experience for the post. The minimum requirements for qualification may also depend on regulatory requirements or the national practice in the State This may include the successful completion of specialized training courses and/or experience in operational radiation protection and/or radioactive waste management.

Training of personnel in radiation protection and radioactive waste management

9.22. The operating organization is responsible for the recruitment, training and periodic retraining of all personnel and for the specification of the levels of competence necessary to perform various duties: see Requirement 70 of SSR-3 [1]. Training should be provided so as to ensure that site personnel attain and maintain the necessary level of competence to perform the duties and functions assigned to their level of responsibility.

9.23. The operating organization of a research reactor should make arrangements for facility personnel to be adequately trained (and confirmed to be proficient) in measures for radiation protection and radioactive waste management, as appropriate for their duties and responsibilities.

9.24. The training for workers should cover all topics relevant to their assigned tasks and the potential hazards. Those workers who work in zones where there are high radiation levels (i.e. controlled areas) should be trained in their specific work activities so as to enable them to successfully perform their duties in the minimum time, in accordance with the principle of optimization. This could include, for example, training on mock-ups, rehearsing the planned work and practising actions that would need to be undertaken quickly in response to an accident.

9.25. Training on emergency procedures is required to be given periodically to ensure that all persons expected to respond in an emergency know which actions they should take: see para. 6.28 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [19]. All relevant personnel are required to participate in periodic exercises that simulate a nuclear or radiological emergency at the research reactor: see paras 6.30 and 6.31 of GSR Part 7 [19] and para. 7.92 of SSR-3 [1]. This includes, where appropriate, off-site personnel, such as fire fighters, medical staff and police, who would be involved in the response to an emergency at the research reactor.

9.26. Due to the individual nature of most research reactors, in addition to any externally organized training courses, in-house training programmes that are tailored to the facility's needs should also be developed. When establishing a training programme, a task analysis should be performed for each relevant position. The successful completion of external training programmes may be a regulatory requirement for specified positions: see para. 7.5 of SSR-3 [1].

9.27. The training programmes should include laboratory and/or workshop training, on the job training relevant to radiation protection and/or radioactive waste management, as well as classroom lectures. Where necessary, the training programmes should include evaluations of the trainees to ensure that they meet the minimum criteria for qualification. Training requirements should be documented for all staff positions, and each employee's level of training should be recorded so that it is available for audit by the pertinent authority.

Training in radiation protection and radioactive waste management for operating personnel

9.28. Operating personnel at all levels should receive sufficient education and training in radiation protection and radioactive waste management so that they can make appropriate judgements and decisions on questions within their competence, in particular, in abnormal situations or emergencies.

Training in radiation protection and radioactive waste management for users of research reactors

9.29. The training programme for users of research reactors should cover the particulars of radiation protection and radioactive waste management for the actual facilities and experimental devices that they are planning to use. This may be achieved through on the job training.

Training of temporary workers in radiation protection and radioactive waste management

9.30. The operating organization should ensure that any personnel of external organizations who are temporarily employed on the site (itinerant workers: see paras 6.21–6.100 of GSG-7 [26]), particularly personnel of contractors, have received proper training to enable them to perform their work in accordance with the radiation protection programme.

9.31. Special arrangements should be made so that temporary personnel from external organizations can become familiar with the relevant radiation protection measures relating to their tasks at the research reactor. In particular, any itinerant workers categorized as occupationally exposed to ionizing radiation should receive appropriate training on the arrangements for radiation protection at the research reactor prior to commencing their jobs. The amount of training needed will depend on the duration and nature of the work. In specific instances, a qualified person from the research reactor may provide continuous supervision in lieu of providing temporary personnel with training.

Instructions to visitors

9.32. Before visitors enter a controlled area, they should be provided with information and instructions on the provisions for radiation protection. Visitors to research reactor facilities are often escorted by qualified persons, and if such escorts are provided, specific training on radiation protection for visitors need only be minimal.

Retraining of personnel in radiation protection and radioactive waste management

9.33. Periodic retraining (see para. 7.30 of SSR-3 [1]) in radiation protection and radioactive waste management should be conducted to ensure that facility operating personnel maintain the necessary level of competence for their positions. Retraining is generally required only for certain functions or positions.

9.34. The operating organization is responsible for prescribing the type and frequency of refresher courses. Retraining should be conducted as often as necessary, but particularly after changes in tasks or significant changes or modifications to equipment or procedures at the research reactor. The aim is to ensure that the necessary level of competence is maintained and that the implications of any changes for radiation protection and radioactive waste management are understood.

9.35. Records should be kept of all training in radiation protection and radioactive waste management that each person has received.

9.36. Training programmes should be updated at regular intervals. In updating the training programmes for radiation protection and radioactive waste management, the operating organization should take into account recommendations and feedback from internal audits and inspections by the regulatory body, as well as operational feedback on events at the research reactor and at other relevant facilities.

9.37. The operating organization is required to establish a system for keeping records and reports: see Requirement 82 of SSR-3 [1]. This includes the generation, collection, retention and archiving of records to maintain current and historical information on important aspects of the radiation protection programme and the waste management programme: see paras 7.95 and 7.97 of SSR-3 [1]. This information should be retained for use in meeting the objectives of these programmes and in the preparation of reports by the operating organization for the regulatory body.

Records of individual monitoring

9.38. The operating organization is required to keep records of individual monitoring in accordance with paras 3.103–3.107 of GSR Part 3 [2]. Such records should be available for purposes of review and assessment. Paragraph 3.104 of GSR Part 3 [2] states:

> "Records of occupational exposure for each worker shall be maintained during and after the worker's working life, at least until the former worker attains or would have attained the age of 75 years, and for not less than 30 years after the cessation of the work in which the worker was subject to occupational exposure."

9.39. All records of individual monitoring should be carefully and periodically assessed by the radiation protection officer for the following:

(a) Any trends relating to particular individuals or groups of individuals working in the same area of the research reactor;

(b) Consistency of the results of individual monitoring with the results of workplace monitoring;

(c) Anomalous or unusual doses;

(d) The effectiveness of any revised procedures in reducing doses.

Records of health surveillance

9.40. Programme for the health surveillance of occupationally exposed workers are required to follow the general principles of occupational health: see para. 3.108 of GSR Part 3 [2]. Such a programme should involve an initial examination and periodic surveillance thereafter. In the initial examination, the health of workers and their fitness for the intended tasks should be assessed; also, any workers who have a medical condition that may necessitate taking particular precautions in their work should be identified.

9.41. Health surveillance records should be confidential, and they should be preserved in a manner that has been agreed with the regulatory body. The minimum period of record keeping should be the lifetime of the worker concerned. However, a longer period of retention should be considered if needed for legal purposes.

Records of workplace monitoring

9.42. The results of workplace monitoring at the research reactor (including routine, task related and special surveys: see paras 7.6–7.18) should be kept in accordance with Requirement 82 of SSR-3 [1]. This includes records associated with the calibration, testing and maintenance of radiation monitoring instruments (see Section 8). The record keeping system should be implemented so that the testing and repair history of each instrument can be retrieved.

10. RADIATION PROTECTION DURING AN EMERGENCY AT A RESEARCH REACTOR

INTRODUCTION

10.1. An emergency plan[4] and emergency procedures are required to be developed to cover all foreseeable aspects of emergencies at a research reactor, in accordance with the requirements established in GSR Part 7 [19] and Requirement 81 of SSR-3 [1]. The plan should specify the infrastructure necessary to provide the capability to respond to a nuclear or a radiological emergency at a research reactor. The emergency plan and procedures are required to be tested, reviewed and updated as necessary, taking into account experience from emergency exercises: see para. 7.92 of SSR-3 [1].

10.2. Due to the small inventories of nuclear material and radioactive material, the radiological aspects of the emergency plan for most research reactors is expected to concern mainly the on-site response. However, notification of an emergency to external authorities and the performance of limited off-site radiation monitoring should be included in the emergency plan. Research reactors with high power levels with the potential for causing significant off-site effects may warrant more extensive emergency arrangements, as described in GSR Part 7 [19].

10.3. In the event of a nuclear or a radiological emergency, the on-site response may include a wide range of activities, such as a preliminary assessment of hazards, rescue operations, actions to stop accident propagation, mitigation of a release of radioactive material to the environment, on-site sheltering and evacuation, workplace monitoring and effluent monitoring, individual dosimetry and dose control, access control, issuing of information to the public and submission of notification and reports to the regulatory body.

RADIOLOGICAL ASSESSMENT

10.4. Paragraph 7.90 of SSR-3 [1] states:

"Emergency plans and procedures shall be based on the accidents analysed in the safety analysis

[4] The emergency plan is a description of the objectives, policy and concept of operations for the response to an emergency and of the structure, authorities and responsibilities for a systematic, coordinated and effective response. The emergency plan serves as the basis for the development of other plans, procedures and checklists.

report as well as those additionally postulated for the purposes of emergency preparedness and response on the basis of the hazard assessment."

Specific radiological assessments should be made when a nuclear or radiological emergency occurs and develops. The assessment should be made on the basis of the available information, and should be promptly updated to take into account any information that would yield more accurate results. Post-emergency assessments to confirm the nature and magnitude of any radiological hazards should also be performed.

Radiation and contamination surveys

10.5. The emergency plan should include radiation monitoring teams that can be promptly deployed to gather data. These teams should be given training commensurate with their role in an emergency at the research reactor and should be equipped with the necessary protective equipment, monitoring instruments and sampling equipment (see also para. 7.93 of SSR-3 [1]).

10.6. Fixed radiation monitors, including emergency monitors and other instruments with remote readouts in the control room or elsewhere, may be used to make an initial assessment of the radiological hazard, and to help determine the need for any urgent protective actions to protect workers and the public.

Individual monitoring

10.7. In the event of an emergency, an evaluation of personal dosimeter readings and an assessment of internal doses, if necessary, should be made as soon as practicable, to assess doses to personnel and also to help in estimating dose rates and contamination levels in different areas of the research reactor. Doses received by personnel involved in the response to the emergency should be measured, recorded, evaluated and entered in their personal dose records.

ON-SITE PROTECTIVE ACTIONS FOR A NUCLEAR OR RADIOLOGICAL EMERGENCY AT A RESEARCH REACTOR

10.8. Arrangements should be established to ensure safety for all persons on the site in the event of a nuclear or a radiological emergency. To minimize the risks to such persons, appropriate protective measures, such as notification, accounting for persons on the site, sheltering, evacuation, relocation and the use of iodine thyroid blocking, should be included in the emergency plan, as necessary.

CORRECTIVE ACTIONS FOR A NUCLEAR OR RADIOLOGICAL EMERGENCY AT A RESEARCH REACTOR

10.9. Corrective actions should be taken to bring a nuclear or radiological emergency under control and to restore safe conditions at the research reactor. Typically, these actions are of an operational nature, but they may have radiological implications.

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ANNEX I

EXAMPLES OF SOURCES OF RADIATION IN A RESEARCH REACTOR

I–1. This Annex provides general examples of the different sources of radiation that there might be in a research reactor; it is not intended to be comprehensive. It is expected that operating organizations will identify the sources of radiation at their own research reactors.

REACTOR CORE AND FUEL

Fission products

I–2. The main sources of radiation in the reactor core are the fission products that are contained in the reactor fuel. Typically, after one year of continuous operation, a 10 MW reactor will contain around 10^{18} Bq of fission products at the time of shutdown. A large fraction of the activity is short lived; after one day the total activity will decrease by a factor of 2–3, and after one month by a factor of 5–10, depending on the length of the period of operation.

I–3. There are approximately 35 chemical elements and over 200 different radionuclides formed in the fission process. Table I–1 lists the most important radionuclides and gives some details with regard to their characteristics and other safety related properties.

TABLE I–1. FISSION PRODUCTS OF IMPORTANCE IN RESEARCH REACTORS

Fission products	Relevant characteristics
$^{88,89}Rb$; $^{134,137,138}Cs$	Most easily detected indicators of leakage from fuel elements, as measured by particulate air monitors
$^{85m,85,88}Kr$; $^{133m,133,135,138}Xe$	Early indicators of leakage from fuel elements, detectable by gaseous air monitors and in coolant water
$^{131,132,133,134,135}I$; $^{134,137}Cs$	Usually the most significant contributors to exposure from releases of fission products
^{132}Te; $^{103,106}Ru$; $^{141,144}Ce$; $^{89,90}Sr$; ^{99}Mo ; ^{140}La ; ^{140}Ba	Additional significant contributors to exposure in incidents involving severe core degradation

Transuranic nuclides

I–4. In research reactors, especially those using low enrichment fuel, neutron capture by ^{238}U continuously produces ^{239}Np (half-life 2.3 d), which contributes a large fraction (up to 30%) of the total gamma activity of the core at shutdown. ^{239}Np is transmuted by beta decay into ^{239}Pu. Small quantities of other actinides are produced by neutron capture by other isotopes of uranium and their decay products.

Activation products

I–5. Activation products such as ^{60}Co are produced in cladding, in structural components of the fuel (e.g. coolant plena) and in possible impurities in the fuel. They are usually of minor importance in comparison with the inventory of fission products in the core.

Prompt radiation

I–6. During operation, the core becomes a source of intensive radiation, as follows:

(a) Neutron radiation: the most probable energy of the neutrons produced in the fission process is 0.73 MeV (with the maximum at 10 MeV). In thermal reactors, the neutrons slow down in the moderator to an average energy of 0.025 eV. Neutrons are also produced by the decay of some fission products (delayed neutrons) and by γ–n reactions in certain core materials, such as deuterium (in heavy water) and beryllium.

(b) Gamma radiation: photons are produced in the fission process with energies from approximately 0.2 MeV to several MeV. The total energy of the prompt gamma rays is about 7 MeV per fission and the average energy per photon is roughly 1 MeV.

COOLANT AND MODERATOR

I–7. The activity present under normal conditions in the coolant and moderator is due to several sources, as follows:

(a) Activation of the moderator or coolant material (e.g. activation of oxygen to ^{16}N or deuterium to tritium). For sodium cooled reactors, ^{24}Na has to be considered as an activation product;

(b) Activation of impurities contained in the coolant or moderator (e.g. ^{23}Na present as an impurity in water is activated by neutron capture to produce ^{24}Na);

(c) Activation of corrosion products from structural components that enter the moderator–coolant system (e.g. ^{24}Na is also produced by the activation of aluminium through an (n, α) reaction; and iron and common impurities contained in aluminium produce radioactive isotopes, such as ^{59}Fe, ^{51}Cr, ^{60}Co, ^{65}Zn and others);

(d) Very small quantities of fission products might be found in the coolant or moderator as a result of uranium contamination that might be present on the external surface of fuel. Fission products might also be found in the coolant in the event of leakage from fuel elements;

(e) Air is usually present in water coolant–moderator systems in dissolved or gaseous form and is activated by neutrons producing ^{16}N, ^{41}Ar and other isotopes.

I–8. Examples of the radioactive isotopes that are commonly found in the coolant of light water reactors are given in Table I–2.

TABLE I–2. TYPICAL CONCENTRATIONS OF THE MAJOR RADIONUCLIDES IN THE PRIMARY COOLANT OF A 10 MW POOL TYPE RESEARCH REACTOR

Radionuclide	Activity concentration (Bq/m^3)
H-3	3.0×10^8
Na-24	9.0×10^8
Cr-51	3.0×10^4
Co-58	7.4×10^4
W-187	1.0×10^7
Hf-181	1.0×10^4
Sb-125	3.0×10^4
Zn-65	3.7×10^4
Ar-41	1.0×10^5

AIR

I–9. Dissolved air in the water coolant of a research reactor and pneumatic conveyors will contain radioactive gases, such as ^{16}N from the activation of oxygen, and ^{14}C from the activation of carbon in CO_2, ^{41}Ar from the activation of ^{40}Ar present in air, tritium from the activation of vapour in the air in heavy water reactors, and other gases or vapours that might be released from the core cover gas systems or through leakages from experimental devices. Air might also contain dust particles that have become radioactive, as well as activated corrosion product particles that were released into the air during maintenance operations. Particulate contamination in the air might also arise from irradiations and from various experiments.

BEAM TUBES AND SAMPLES

I–10. Research reactors usually have built-in facilities and devices for a variety of experiments, irradiations or isotope production, which present a radiological hazard. These include beam tubes enabling neutron radiation and gamma radiation to reach experimental set-ups outside the reactor core, and shielding and special irradiation tubes for introducing samples into the core or next to it. Usually, samples may be introduced in specified locations in or next to the core by loading them from the top of

the reactor.

I–11. Sample materials as well as their containers and other components of the irradiation set-ups might become highly radioactive and present a radiation and contamination hazard.

I–12. Prompt gamma radiation due to the activation of samples situated in a neutron beam outside the shielding might pose a significant hazard due to dose rates and contamination.

REACTOR COMPONENTS

I–13. The reactor components located in and around the core become radioactive owing to activation and contamination after the reactor has been operated for some time. These components include: structural materials, parts of the core, such as control rods and their associated components, guide tubes, flux and temperature gauges, dummy fuel elements, components of experimental assemblies, reflectors and shielding materials.

I–14. Some other components, such as filters placed in the ventilation system or in gas purging systems, or filters and ion exchange resins for water purification, collect radioactive impurities that might become significant sources of radiation. Deposits of radioactive crud and sludge along pipes can also contaminate reactor components and might constitute a radiation and contamination hazard even outside the core region.

RADIOACTIVE WASTE

I–15. Handling of experiments, maintenance operations, decontamination of contaminated items, replacement of water in the spent fuel storage pool, and spills or leakages from the core and related systems lead to the production of radioactive waste. This waste accumulates with time and needs to be properly collected and treated. Recommendations on the management of radioactive waste are provided in Section 6.

SPENT FUEL

I–16. Spent fuel is a very significant source of radiation and special considerations are needed with regard to its handling, storage, transport and subsequent disposal. The radiological hazard is due to direct radiation and to releases in the event of cladding failure.

SPECIAL RADIOACTIVE SOURCES

I–17. Various types of radiation source may be utilized in a research reactor. These include the following:

(a) A neutron source for the startup of the reactor, which may be stored in the core or outside the core in the reactor pool or in shielded containers;

(b) Neutron and gamma sources for the calibration and checking of monitoring instrumentation;

(c) Irradiated targets for radioisotope production and neutron activation analyses.

I–18. Research reactors are widely used for the production of radioisotopes. Various target materials are irradiated in the reactor as part of the production process. In many cases, enriched fissile material (^{235}U) is irradiated for producing radioisotopes (e.g. ^{99}Mo). The irradiated targets may be temporarily stored in a hot cell complex in the reactor building and subsequently transferred to a dedicated radioisotope processing facility. A significant amount of radioactive waste originates from chemical processing of the irradiated targets, performed in a dedicated facility. The management of radioactive waste generated from the chemical processing of irradiated targets for the production of radioisotopes is not specifically addressed in this Safety Guide. Some solid waste can be generated from failed irradiations, damaged irradiation cans and the processes involved in the checking and transfer of irradiated samples. The irradiated samples become sources of radiation in a research reactor. Generally, processing of irradiated samples is done outside the research reactor in specially designed laboratories.

I–19. A variety of samples are irradiated in research reactors for research and development. The geometry of such samples is not usually standard. Such samples may be temporarily stored in the auxiliary facilities of the reactor prior to their transfer to designated areas for further analysis and examination. Such irradiated samples are also sources of radiation in research reactors.

ANNEX II

PERSONAL PROTECTIVE EQUIPMENT AT A RESEARCH REACTOR

II–1. Personal protective equipment comprises clothing and other special equipment that is issued to individual persons to provide protection against exposure to radiation. It is used to provide protection in circumstances in which it is not reasonably practicable to provide complete protection by means of physical control measures and/or administrative measures. Adequate personal protection depends on personal protective equipment being correctly selected, fitted and maintained. Appropriate training for the users and arrangements to monitor usage are also necessary to ensure that personal protective equipment effectively provides the intended degree of protection.

USE OF PERSONAL PROTECTIVE EQUIPMENT

II–2. As an administrative measure to restrict exposure, and where neither physical control measures nor other measures are reasonably practicable, the operating organization needs to provide personal protective equipment. The use of personal protective equipment may be an effective means of controlling the exposure of workers involved in emergency operations. Personal protective equipment includes clothing or other special equipment that is issued to protect individuals. It is essential that all persons involved in the management and use of personal protective equipment are aware of its capabilities and limitations, to ensure that an adequate and reliable degree of protection is provided.

II–3. Different personal protective equipment may be used to protect against external exposure and internal exposure. Protective clothing may be designed to shield large areas of the wearer's body or individual organs, such as the eyes, against certain types of external irradiation. However, personal protective equipment is more commonly used to protect against internal exposures by preventing radioactive substances from making direct contact with the skin or being ingested.

II–4. Respiratory protective equipment is intended to prevent the inhalation of radioactive substances.

SELECTION OF PERSONAL PROTECTIVE EQUIPMENT

II–5. The following information is needed before selecting personal protective equipment:

(a) *The nature of the exposure:* Both qualitative and quantitative information is needed about radiological conditions in the workplace, including the radionuclide(s) present, their physical and chemical form, the distribution and levels of any workplace contamination, the exposure pathways and the possible magnitude of exposures.

(b) The radiological risks need to be considered together with other hazards that are present to appreciate the difficulties in accomplishing the work wearing personal protective equipment.

(c) *Performance data of personal protective equipment:* Data are needed on the degree of protection

provided by the personal protective equipment. This information will usually be available from the manufacturers, who will have performed tests under controlled conditions as specified in international or national regulations and standards.

(d) The acceptable level of exposure*:* The level of protection that is to be achieved by the use of personal protective equipment needs to be determined. In practice, a decision will be made, preferably by a qualified expert, on whether the personal protective equipment can provide adequate protection.

II–6. Maximum protection will only be obtained in practice if the personal protective equipment is fitted, used and maintained to the standards specified for the manufacturer's tests.

II–7. Personal protective equipment is manufactured in a range of sizes. It may be necessary to try different products of a similar specification to find personal protective equipment that is comfortable and a good fit and that provides the necessary protection. The training provided to the users of personal protective equipment needs to emphasize the importance of a good fit and the correct use of the equipment every time.

II–8. Personal protective equipment needs to be cleaned, checked and maintained in accordance with the manufacturer's recommendations.

PROTECTIVE CLOTHING

II–9. Protective clothing may be used to shield the body from some types of external radiation, or to prevent contamination from reaching the body surfaces and to reduce the spread of contamination. Procedures associated with the wearing and removal of protective clothing are designed to control loose contamination. The following are examples of typical protective clothing:

(a) Flexible aprons containing up to the equivalent of 0.33 mm of lead are available to shield the front of the body. Double sided aprons also shield the back of the body. The aprons attenuate, by about 90%, low energy radiation such as scattered X rays (of tens of kiloelectronvolts).

(b) Shielding gloves and sleeves containing shielding equivalent to up to 0.33 mm of lead are available. Like aprons, they are only effective shields against beta radiation and low energy X rays.

(c) Laboratory coats are commonly used where there is a risk of minor radioactive contamination. Laboratory coats for dry conditions are worn with appropriate gloves and overshoes and, in some cases, a cap or hood to protect the hair and head.

(d) One-piece suits, coveralls or overalls are used in workplaces where more widespread contamination of the body might occur.

(e) Protective gloves range from lightweight disposable polythene gloves to gloves made of other synthetic materials, various fabrics and elastomers, leather, mineral fibres, glass fibre, or from

a mixture of materials.

(f) Protective footwear includes overshoes (or boots) and dedicated shoes (or boots). Overshoes allow personal footwear to be worn in areas where there is a risk of a minor floor contamination. In their simplest form, overshoes are disposable, single size, foot shaped plastic bags with elasticated openings. Outsized plastic shoes are more expensive and durable but can be less effective if they do not fully cover the personal footwear and do not provide a tight fit. Fabric overshoes or boots with hard soles, and fabric overshoes with leggings supported at the knee by elastic or drawstrings are also an option.

(g) Pressurized plastic clothing, worn with a full hood, rubber gloves and shoes, protects in damp conditions and against airborne tritium contamination. This type of clothing is also appropriate for work involving large amounts of loose surface contamination. The clothing is provided with an adequate air supply to cool the body and to purge contaminated air. Breathing air is supplied to the hood.

(h) Additional special items of clothing may be used as necessary, e.g. heavy rubber gloves and rubber boots when working with contaminated water.

II–10. Some items of protective clothing are available in either reusable form or disposable form. The reuse of clothing involves laundry facilities. It is important that highly contaminated clothing is segregated from clothing with just trace levels of contamination, through the use of appropriate administrative control levels and monitoring. Administrative control levels and monitoring also need to be available for the reissue of clothing that has been laundered. The laundry process might increase the volume of low level liquid radioactive waste to be handled.

II–11. Disposable clothing may be relatively costly when the rate of usage is high, and it contributes to the volume of solid radioactive waste to be processed. It is appropriate, however, to use disposable clothing when laundering becomes impracticable due to high levels of contamination.

TYPES OF RESPIRATORY PROTECTIVE EQUIPMENT

II–12. There are different types of respiratory protective equipment, as follows:

(a) Respirators purify the air by filtering out particulate materials such as dust or gas or vapour in low concentrations. The most common types are as follows:

  (i) Filter respirators;

  (ii) Half mask respirators;

  (iii) Full face mask respirators;

  (iv) Powered respirators fitted with a fan and filter(s) to supply air to a half mask, full face mask, visor, hood or helmet, blouse, half suit or full suit.

(b) Breathing apparatus provides clean air or oxygen from an independent uncontaminated source. The most common types are as follows:

(i) Fresh air hose equipment;

(ii) Constant flow compressed air equipment;

(iii) Breathing apparatus that includes full face masks and full suits supplied from either compressed air lines or self-contained cylinders of compressed air.

II–13. Respiratory protective equipment and some other types of personal protective equipment typically have an assigned protection factor or nominal protection factor defined by national standards and referred to in national regulations. Respiratory protective equipment is performance tested to determine the inward leakage Manufacturers normally quote the nominal protection factor, which is the expected ratio of the concentration of the contaminant in the ambient atmosphere to the concentration of the contaminant inside the respiratory protective equipment.

II–14. The effectiveness of a respirator in minimizing inward leakage depends on two separate parameters, as follows:

(a) The integrity of the face seal;

(b) The filtration capability of the canister or filter.

SELECTION OF RESPIRATORY PROTECTIVE EQUIPMENT

II–15. Several types of respiratory protective equipment may be capable of providing the necessary protection. All types of respiratory protective equipment will protect against low concentrations of an airborne particulate contaminant. If protection against radioactive vapours and gases is needed, the choice is restricted to certain types of respirator or breathing apparatus. For protection against high concentrations of airborne contamination, breathing apparatus may be the only effective option. Tritium gas has a high diffusivity and necessitates special considerations to prevent its inhalation, ingestion or absorption through the skin.

II–16. The use of respiratory protective equipment needs to be carefully controlled. Users need to be medically fit to use the equipment and be able to wear the equipment as intended. This may include fit tests.

RESPIRATORY PROTECTIVE EQUIPMENT THAT IS AVAILABLE FOR USE AT A RESEARCH REACTOR

II–17. The following respiratory protective equipment is typically provided for use at research reactors (listed in increasing order of protection):

(1) Filtering face piece respirators. Some models are capable of filtering malodorous (but not toxic) gases and vapours.

(2) Elastomer half masks or oronasal respirators. These cover the nose and mouth and are held in place by adjustable straps. Air is drawn through one or more filters and, where fitted, an inhalation valve. The filters are contained in one or more cartridges (canisters). Exhaled air is discharged to the atmosphere through an exhalation valve in the face piece.

(3) Full face respirators. These cover the entire face and are held in place with an adjustable head harness. Air is drawn through one or more approved filter canisters and, where fitted, inhalation valves. It may be used for conditions involving loose contamination and airborne particulates. Full face respirators with activated charcoal canisters may be used for protection against airborne radioiodine. The charcoal canisters must have adequate removal efficiencies for all chemical forms of radioactive iodine that might be encountered at the research reactor.

(4) Powered air-purifying respirators. These provide a continuous flow of air into the mask to minimize inward leakage of contaminated air around the face seal. Ideally, the nominal protection factors are then determined only by the filter characteristics and are higher than the nominal protection factors of non-powered respirators. Contaminated air is drawn through one or more filters by a battery powered fan, and the filtered air is delivered to the mask. Air-supplied masks and air-supplied hoods are provided with breathing air under pressure and give good respiratory protection against all airborne contaminants, but will not provide protection against skin absorption if tritiated water vapour is present.

(5) Constant flow compressed air hose equipment, in which a compressed air line is used to supply a face mask, a hood or a blouse. The system provides an adequate flow of air and is conditioned for temperature and humidity. Compressors need to be oil-free. To ensure that breathing air and instrument air are not accidentally interchanged, different, non-compatible fittings may be used on each system. When portable compressors are used, the air intake needs to be from a contamination-free area.

(6) Self-contained breathing apparatus consisting of a full face mask supplied with air or oxygen from compressed gas cylinders carried by the individual. Air is supplied to the mask through a positive pressure demand valve.

(7) Ventilated pressurized suits enclosing the whole body (including arms and legs), which may consist of one or two parts. Halved suits are sealed together at the waist. Full suits may have a gastight zipper. The front section of the hood is transparent, offering minimum distortion or interruption of the wearer's vision. The compressed air supply hose is attached to a belt to withstand the stresses of being dragged. A valve may be attached to the belt to allow the wearer to control the air supply, either to the whole suit or to the hood. Exhaust gases are discharged through exhaust valves in the suit body. Part of the air supply may also be used to cool the suit.

II–18. Respiratory equipment needs to be well maintained, periodically inspected and tested. Facilities are needed for cleaning respiratory protective equipment, and the reissue of equipment for use needs

to be closely controlled through monitoring and administrative control levels.

OTHER PERSONAL PROTECTIVE EQUIPMENT

II–18. Other types of personal protective equipment may also be necessary, for example for head protection, eye protection or toe protection. A safety helmet may be worn when wearing enclosed suits or hoods. Safety goggles may be worn inside ventilated suits. It is an advantage if the one item of personal protective equipment used incorporates all necessary protection, such as a ventilated helmet, a respirator with protective eyepieces, or integral boots with protective toecaps. Use of an eye shield with a respirator will severely limit the already restricted vision. Welding in areas in which there is a contamination hazard necessitates specially modified personal protective equipment, with the hoods of ventilated garments fitted with a welder's mask, eye protection and an outer protective apron to protect against hot debris.

II–19. Suits made from aluminized fire resistant materials are available to protect against extreme radiant heat. In very hot environments, a cooled suit might be used. Suits designed to be resistant to chemicals may also be suitable to protect against radiological hazards, but this needs to be assessed before use.

ANNEX III

DESIGN CONSIDERATIONS FOR COLLECTION OR HOLDING (DELAY) TANKS FOR LIQUID RADIOACTIVE WASTE AT A RESEARCH REACTOR

III–1. At a research reactor where significant volumes of radioactive liquid waste are generated, collection or holding tanks may be installed. In general, such tanks are constructed in sets of two or more, so that one can be filling while the contents of another are being sampled, analysed or discharged. The design of such tanks needs to include the following:

(a) Tanks are constructed of material that is resistant to chemical attack, such as steel, plastic, rubber lined carbon steel or fibreglass.

(b) Tanks are constructed so that they can be expected to remain leak-free for their design lifetime.

(c) Tanks are fitted with visual indicators of the volume of the contents at any time and have warning devices that operate when a tank is almost full so that the incoming effluent may be manually or automatically switched to fill another tank.

(d) Tanks are fitted with appropriate equipment for stirring, venting and transfer to prevent the sedimentation of sludge and the accumulation of hazardous gases in the tanks.

(e) Provision is made for sampling.

(f) A reserve capacity is provided to allow for unplanned events.

(g) Means of access to the tanks are provided to allow for visual inspection of the buildup of any internal deposits on the base and sides, and to allow access for cleaning if this becomes necessary.

(h) Secondary means of confinement is provided around the tanks to prevent the spread of contamination in the event of leakage.

(i) Shielding around the tanks is also provided if necessary.

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