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Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors (Revision of NS-G-4.6)

DS 509F

DRAFT SPECIFIC SAFETY GUIDE

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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].

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 [3];
- IAEA Safety Standards Series No. DS509B, Maintenance, Periodic Testing and Inspection of Research Reactors [4];
- IAEA Safety Standards Series No. DS509C, Core Management and Fuel Handling for Research Reactors [5];
- IAEA Safety Standards Series No. DS509D, Operational Limits and Conditions and Operating Procedures for Research Reactors [6];
- IAEA Safety Standards Series No. DS509E, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [7];
- IAEA Safety Standards Series No. DS509G, Ageing Management for Research Reactors [8];
- IAEA Safety Standards Series No. DS509H, Instrumentation and Control Systems and Software Important to Safety for Research Reactors [9].

1.4. Additional recommendations on the safety of research reactors are provided in IAEA Safety Standards Series Nos SSG-20, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report [10] and SSG-24, Safety in the Utilization and Modification of Research Reactors [11].

1.5. The terms used in this Safety Guide are to be understood as defined and explained in the IAEA Safety Glossary [12].

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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].

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. Research reactors of higher power, specialized reactors (e.g. homogeneous reactors, fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops) may need additional guidance. 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 [13] might be more suitable.

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 a less comprehensive commissioning programme than that outlined here. While all recommendations in this Safety Guide are to be considered, some might not be applicable to these research reactors with low hazard potential 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 [14]).

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. Certain topics are not dealt with in detail here, owing to their specialized nature, 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 [15, 16, 17, 1854 7, 116] and this Safety Guide should be used in conjunction with them.

1.13. This Safety Guide does not provide guidance 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. Section 3 deals with the radiological hazard of different types of research reactors and identifies possible sources of radiation in the reactor facility. Section 4 provides recommendations on design aspects to be considered in order to facilitate radiation protection and radioactive waste management, including design approaches for operation, decommissioning and managing accident conditions. Section 5 deals with physical and administrative control of the radiological hazards identified in Section 3. Section 6 describes the issues to be considered in the operation of research reactors pertaining to radioactive waste management. Section 7 provides recommendations on monitoring for facilities and personnel, and for the possible releases associated with research reactors. Section 8 provides recommendations on instrumentation and its required characteristics for radiation protection purposes. Section 9 discusses organizational aspects that could affect the meeting of the requirements on radiation protection at research reactor facilities, including the responsibilities of the operating organization, the safety committee and the regulatory body, as well as recommendations on staffing, qualification and training. Section 10 provides an outline of the radiological aspects of emergencies at research reactors. Annex I provides examples of radiation sources in research reactors to be considered in the design and operation 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. APPLICABLE REQUIREMENTS, OBJECTIVES AND CONCEPTS

SAFETY OBJECTIVE

2.1. "The fundamental safety objective is to protect people and the environment from ~~the~~ harmful effects of ionizing radiation" [20~~3~~]. To meet this objective, a comprehensive safety ~~analysis~~ assessment should be carried out 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 ~~analysis~~ assessment should examine:

(1) All planned normal operational modes of the facility;

(2) Performance of the facility following anticipated operational occurrences;

(3) Design basis accidents;

(4) Event sequences that may lead to ~~a beyond design basis accident.~~design extension conditions (DEC).

On the basis of this analysis, the robustness of the engineering design in withstanding postulated initiating events and accidents should be established, the effectiveness of the safety systems and of

safety related systems should be demonstrated, and requirements for emergency response should be established.

2.2. Measures should be taken to minimize the likelihood of an accident that could lead to loss of control over the source of radiation. Even so, there remains a very low probability that an accident may happen, therefore, measures should be in place to mitigate the radiological consequences of an accident. 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 AND CONSTRAINTS

2.3. The design of a research reactor facility should be such as to ensure that ~~authorized~~ annual dose limits[2] ~~for~~ authorized by the regulatory body (in accordance with the requirements in GSR Part 3 [2]) for site personnel and the public will not be exceeded in operational states (normal operation and anticipated operational occurrences) or in decommissioning. ~~The authorized dose limits should not exceed the values of the dose limits established in Ref~~GSR Part 3 [2].

~~2.3.~~2.4. As part of the implementation of the principle of optimization (as discussed in the following section), a dose constraint may ~~also~~ be applied to individual doses associated with occupational exposure and public exposure. ~~The purpose of defining a~~ Ddose constraints are set for each source of radiation to serve as boundary conditions in defining the range of options for the purposes of optimization of protection and safety [2] and~~is~~ to ensure that the sum of doses to the ~~critical group~~representative person remains within the authorized annual dose limits. For workers who do not enter the designated supervised areas and controlled areas, the dose constraints should be set at the same level as the individual dose limit for members of the public [21~~8~~]. The dose constraints for members of the public apply to the average dose to the ~~critical groups~~representative person[3] of the population~~, that is, groups of persons that are reasonably homogeneous with respect to their exposure for a given radiation source, and are representative of individuals who receive the highest dose as a result of the operations that are being undertaken~~ [2, 22~~4~~9]. ~~A critical group may be specific with respect to age or gender [9].~~ Pre-operational studies should be carried out to identify the ~~critical groups~~representative person and critical pathways for the exposure of such ~~groups~~a person. The annual discharge limits for specific radionuclides in liquid and gaseous effluents ~~permit~~ allow for increased release rates over short time periods, thus, operational flexibility should be defined, with consideration of the technical possibilities, and account taken of

~~[2] An authorized dose limit is one that has been established or formally accepted by a regulatory body.~~

~~[3] The new ICRP publication on Assessing Dose of the Representative Individual for the Purpose of Radiation Protection of the Public represents the public by the 'representative individual' rather than the 'critical group'.~~

the dose constraints for ~~members of the critical groups~~the representative person, using the approved critical exposure pathways for all relevant operations involving the use of radioactive material. In setting discharge limits, it should be ensured that the maximum dose to ~~an individual in the critical group~~the representative person does not exceed the dose constraint.

APPLICATION OF THE PRINCIPLE OF OPTIMIZATION

~~2.4.~~2.5. In principle, specification of the authorized dose limits is left to the appropriate regulatory body. For keeping all doses within limits and ~~as low as reasonably achievable,~~for minimizing exposure, the following economic and social factors should be taken into account:

(a) The radiation exposure resulting from a practice should be reduced by radiation protection measures to values 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) In the design, issues such as reducing major disparities in the doses due to occupational exposure received by workers of different types who work in the controlled area (see paras 5.46–5.49~~50~~), and avoiding arduous working conditions in controlled areas (social factors), should be taken into account.

~~2.5.~~2.6. In general, application of the optimization principle in radiation protection implies that a choice is to be made from a set of possible protective measures. To this end, feasible options should be identified, the parameters to serve as criteria for comparison and their appropriate values should be determined and, finally, the options should be evaluated and compared. The optimization principle should also be applied to design features, ~~such as intervention~~and procedures whose purpose is to prevent, or to mitigate the consequences of, accidents at the facility that could lead to radiation exposure of the site personnel or the public.

~~2.6.~~2.7. 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 reliable research reactor facility ~~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.

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DESIGN TARGET DOSES

~~2.7.~~2.8. In order to ensure that the design is such that doses are kept to levels which ~~are as low as reasonably achievable~~minimize exposure and represent best practice, 'design target doses' should be set for individual doses and the 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[4] for individual doses to site personnel and to members of the public should be consistent with the concept of dose constraints, ~~which is~~as discussed in paras ~~2.24~~1.22 and ~~2.26~~1.23 of Ref. [2]. The design target doses should be set in accordance with the dose constraints, and with consideration given to technical and economical alternatives.[5] In this regard, if a further reduction in dose below a design target dose can reasonably be achieved, it should be ~~done~~implemented.

~~2.8.~~2.9. 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 compliance with national regulations for the collective dose to the groups of workers who are likely to receive the highest doses, such as maintenance workers or health physics staff. It is also useful to set design target doses for the collective dose for each category of work, such as maintenance of the major components, in-service inspection, refuelling and waste management. ~~These target doses, combined with assessments at key stages in the design process of predicted doses, can be used to monitor the major contributions to the dose.~~

DESIGN TARGET DOSES FOR ACCIDENTS

~~2.9.~~2.10. The adequacy of the design provisions for the protection of workers and the public against postulated accident conditions should be judged by a comparison of calculated predicted doses with specified criteria for exposure, which are the design target doses for accidents. In general, the higher the probability of the accident considered, the lower the specified design target dose should be. The operator and/or designer may take account of this principle by setting different design target doses for 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 off-site radiological consequences or only minor radiological consequences outside the site boundary, depending on the national regulatory requirements. The regulatory body should specify what constitutes minor radiological consequences. Typically, these would correspond to very

~~[4] The term 'target dose' is used throughout this Safety Guide in relation to both individual doses and collective doses.~~

[5] It is important to recognize that design target doses are not limits. They are useful design tools in the optimization process. Provided that any excess can be justified, design target doses may be exceeded. However, the achievement of a design target dose does not, in itself, demonstrate that the design satisfies the optimization principle.

restrictive dose levels in respect of which there would be no need for any off-site measures.

~~2.10.~~2.11. It is beneficial to address design basis accidents and ~~beyond design basis accidents~~design extension conditions separately and to set design target doses for each of them. ~~In the case of beyond design basis accidents, it is appropriate to set design target doses~~. In addition to providing assurance to the regulatory body, design target doses may be set to meet the concerns of members of the public.

3. RESEARCH REACTORS AND ASSOCIATED RADIOLOGICAL HAZARDS

INTRODUCTION

3.1. Research reactors ~~constitute a diverse~~are a diverse group of facilities that can be classified in many ways (e.g. as research reactors, training reactors and prototype reactors, ~~and~~ critical and subcritical assemblies, by type of moderator and coolant or by purpose of utilization). ~~No matter how these facilities are categorized, it is clear that their~~The radiological hazards and the methods of control ~~of these hazards will vary greatly~~vary depending on the hazard potential of the research reactor. This section and those that follow discuss issues relating to radiation protection and radioactive waste management at research reactors. It is the duty of the radiation protection officers to evaluate which of these issues are applicable to the type of facility for which they are responsible.

RADIOLOGICAL HAZARD ACCORDING TO REACTOR TYPE AND UTILIZATION

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

3.3. The radiological hazard will depend predominantly on the following aspects, as explained in the following sections:

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

(b) Type of research reactor,

(c) Transient characteristics of the research reactor;

(d) Irradiation facilities and experimental facilities 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 and radionuclide inventory

3.4. In the case of research reactor facilities with very low power levels, including critical and subcritical assemblies, the typically low neutron fluxes generally result in an insignificant

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production of activation products. Systems for the retention or reduction of gaseous releases, as well as the continuous and sophisticated monitoring of releases, may therefore not be required. 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 required only against direct radiation from the fuel elements.

3.5. For reactors with power levels above the 10 kW range, the requirement 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 stringent. Similarly, additional measures should be taken to cope with normal and possible abnormal releases. The need for programmes for radiation monitoring and waste management also increases.

3.6. For reactors with power levels above the 1 MW range, depending on the design of the core and the fuel elements, fuel ~~melting~~ damage, including damage from residual decay heat, becomes a possibility and special design features are required to preserve the integrity of the fuel (e.g. in the event of a loss of coolant accident). In this case, special consideration should be given to operational aspects, as well as to staff training, to ensure an appropriate response to such events. These aspects should also include the capability to monitor adequately any possible accidental releases.

Type of reactor

3.7. The type of reactor significantly influences needs 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.

3.8. This applies, for instance, to heavy water moderated reactors, for which special attention should be paid to the production of ^{3}H (tritium), including its monitoring. This also applies to liquid metal cooled reactors, for which special precautions should be prescribed for coping with incidental leaks of coolant. Another example is that of light water reactors, for which gamma radiation dose rates from ^{41}Ar and ^{16}N should be taken into consideration.

Power excursions (transients)

3.9. Power excursions (transients) may 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 ~~of low power level~~, ~~or subcritical assemblies~~ which in normal operating conditions do not require significant shielding. In the design of the irradiation facilities (e.g. beam tubes), the direct radiation caused by transients should be considered.

3.10. If such power excursions are part of the design for normal operation, the equipment and

installations for both monitoring and 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.

3.11. 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 the design of pulsed reactors (with the intentional generation of transients).

Irradiation facilities and experimental facilities

3.12. Irradiation facilities and experimental facilities use direct radiation from the core, the irradiated fuel or activated sources, which may pose a radiation hazard to personnel. For such facilities, special design features should be put in place 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.

3.13. 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 facility.

3.14. Special precautions should be taken in irradiating materials that may readily decompose or change state or whose chemical reactivity may be enhanced, causing an overpressure or the release of gases that may be flammable and/or explosive.

3.15. 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 or fertile material.

SOURCES OF RADIATION

3.16. In the radiation protection programme for any facility, account should be taken of all sealed and unsealed radiation sources that may be present in the facility. A full listing of such sources and their forms, locations and related activities during normal operation, as well as during possible abnormal situations, provides the basis for shielding calculations, zoning, the design of ventilation systems, surveillance planning, dose assessment, radioactive waste management and the determination of effluent releases [10]. It should be noted that spent fuel presents a radiological hazard due to direct radiation and due to releases in the event of cladding failures. Consideration should be given, therefore, to the handling, storage, transport and subsequent disposal of spent fuel. Specific guidance is provided in Ref.~~s~~ [18~~1~~, ~~12~~].

3.17. 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;

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

(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 ~~facilities~~ 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) ~~Startup n~~Neutron source~~s~~s 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.

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

4. ASPECTS OF RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT IN DESIGN

APPROACH TO DESIGN FOR OPERATIONAL STATES AND DECOMMISSIONING

Human resources

4.1. The designer should be fully aware of the measures for operational radiological protection and for the management of radioactive waste 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 interrelation between design aspects and operating procedures can be properly addressed.

4.3. To implement this structured approach, the designer should have an appropriate ~~safety~~ culture for safety [23~~12~~] in which the importance of radiation safety and of the safety of radioactive waste at each stage of the design is recognized. As part of the application of the optimization principle at the design stage, project management should set up 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.

4.4. A good ~~operating~~ culture for safety associated with the application of the optimization principle should be established by ensuring that all participants in a project are aware of the general requirements for ensuring radiation protection and of the direct and indirect effects of their individual activities or functions on the provision of radiation protection for workers and the public.

4.5. More precisely, a good ~~operating culture~~culture for safety in the application of the optimization principle should be established on the basis of:

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

(b) A comprehensive training programme;

(c) Management of the relationship between operation, maintenance and design by means of appropriate methodology;

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

(e) Familiarity with the available software to assist in optimization of the

design;

(f) 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.

4.6. In order to ensure that optimal safety is provided for in the design, specialists in radiation protection and radioactive waste management should be closely involved in the design process. Use should be made in the design process of advice from specialists based on:

(a) Their expertise in all areas relating to the production, handling and transport of radioactive material on the site and the ~~transport~~ 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 doses, using available software and data from relevant operating experience;

(c) Their familiarity with the relevant regulations, guidance and best practices.

Organizational aspects

4.7. The need to achieve an adequate level of radiation protection affects a wide range of issues associated with the design. Hence, there is a need to ensure that all design decisions that affect exposure to radiation are consistent with the requirements for radiation protection. However, the design process should be planned in such a manner that implementation of these requirements is an

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integral part of the project and is not on the critical path (i.e. affecting its scheduling and completion). There should be a means of ensuring that the design engineers implement the measures required for radiation protection and waste safety at all necessary stages of the design process.

4.8. In the organization of the design project, it should be ensured that:

(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 good engineering practices that operating experience has shown to be effective in reducing exposure, and 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 and waste safety specialists;

(d) There is an appropriate review mechanism in place to resolve any design or operational issues that may be in conflict with requirements for radiation safety and waste safety.

~~(d)~~(e) ~~Lifetime analysis should be conducted to allow for~~Activation of SSCs will not prevent their removal or~~and~~ replacement ~~of all components throughout~~ the lifetime of the facility.

4.9. The whole design process should be carried out under an appropriate management system [24~~13~~, ~~14~~]

4.10. There should be a strong commitment on the part of management to ensure that the optimization principle is effectively applied.

Design strategy

General approach

4.11. As discussed earlier, the design target doses set at the start of the design process should include:

(a) Constraints on the annual collective dose and annual individual dose to site personnel;

(b) Constraints on the annual individual dose to members of the public.

4.12. In practice, these ~~two~~ design target doses can be addressed independently, although in principle they 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 carried out by site personnel, with a consequent increase in their exposures. In providing the best practicable means for reducing radioactive releases, the implications should be assessed to ensure that there is no undue increase in the exposures of site personnel.

4.13. In setting the design target doses, account should be taken, where practicable, of experience at similar research reactor facilities where there has been a good operating record in terms of radiation

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protection. Account should be taken of any differences in the design and in operation between these reference facilities and the facility that is being designed. Such differences might include differences in the power level, the materials used, the type of fuel, the burnup and the experimental set-ups.

Design for radiation protection of site personnel

4.14. The design provisions to ensure the radiation protection of site personnel should be developed along the following lines:

(a) A strategy for controlling exposures should be developed so that important aspects are considered early in the design. In most types of reactor design, design provisions in two areas may have major impacts in reducing exposures. The first area is ~~scheduled and unscheduled~~preventive and corrective maintenance. Appropriate design provisions should be made for inspection and maintenance. Hence, a proven design with high reliability and a low level of requirements for maintenance, especially ~~unscheduled~~ corrective maintenance, would be the preferred option. The second area that should be considered involves design features that minimize the production and buildup of radionuclides, since reducing amounts of radionuclides will reduce radiation levels and contamination levels throughout the facility, whereas a local solution, such as increasing the shielding or improving the ventilation, will have only a local benefit.

(b) General requirements should be developed and documented. These will include the principles on which the layout of the facility will be based, taking due account of the requirement for the interface between nuclear safety and nuclear security [1], provisions for accident conditions, and restrictions on the use of particular materials in the design of the reactor.

(c) A logical layout should be developed for the facility and zones should be defined on the basis of predicted dose rates and contamination levels, access requirements and specific requirements, such as the need to separate components that perform safety functions. The dose rates may be calculated by using the source terms that provide the basis for the radiation protection aspects of the design, or they may be calculated on the basis of operating experience at similar research reactors, provided that there are no significant differences in the relevant parameters for design and operation. ~~The zoning~~ Zoning of the facility needs to~~should~~ be consistent with ~~regulatory~~ national requirements and with GSR Part 3 [2].

(d) The maintenance programme and operational tasks should be specified, preferably on the basis of well established concepts. The collective and individual doses should be evaluated. Full use should be made of relevant operating experience, where available, particularly for work that is difficult to predict, such as corrective~~unplanned?~~?? 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.

4.15. The design provisions to facilitate subsequent life extension should include the following features:

(a) Design features to enable in-service inspection that may 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.

4.16. 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.

4.17. An outline of the decommissioning plan should be developed to facilitate verification that the design includes the necessary features required to reduce and control exposures during decommissioning. In many cases, these features are the same as those required in operational states, but some additional special features may be required to facilitate decommissioning. If these additional features are significant, the requirements for operational states and those for decommissioning should be optimized.

4.18. A thorough review of design features, for the purpose of facilitating decommissioning, should be performed at the design stage for the research reactor facility. In general, design features that are conducive to maintenance and inspection during the operational lifetime of the reactor will also facilitate decommissioning. To facilitate decommissioning, attention should be given to a number of factors, including the following:

(1) Careful selection of materials to minimize:

(i) Activation;

(ii) Generation of radioactive waste;

(iii) The spread of activated corrosion products;

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

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

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

(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) Adequate space for the storage of waste;

(vi) Control of radioactive material within the installation.

4.19. The design should facilitate compliance with the design target doses for occupational exposures using all or some of the following measures:

(1) Reduction of dose rates in working areas by:

(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.

(2) Reduction of occupancy times in radiation fields by:

(i) Specifying high standards for 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

4.20. As discussed in the previous section, 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.

4.21. 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

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the identification of ~~critical groups~~the representative person and the exposure pathways for the representative person~~. for these critical groups.~~

Commissioning

4.22. 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 carried out for ensuring the adequacy of the design (see ~~paras 5.34 and 5.35 of~~ Ref. [11~~15~~14]).

DESIGN FEATURES FOR RADIOACTIVE WASTE MANAGEMENT

General

4.23. In accordance with Requirement 15 of SSR-3 [1] and Requirement 17 of GSR-Part 5 [15~~4~~], t~~T~~he design should incorporate features that facilitate the safe handling of waste generated at the facility, storage on site, removal of waste, transport and future disposal of radioactive waste and the control of 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.

4.24. In accordance with Requirement 59 of SSR-3 [1], The design should include provisions for the safe management of separate streams of solid, liquid and gaseous radioactive waste within the facility and in associated experimental devices. Provisions for the storage of waste in transit and for the removal of waste should also be considered.

4.25. Where necessary, provision should be made for an interim containment in areas where radioactive effluent or radioactive waste is stored prior to its treatment and discharge.

4.26. The design should be such as to ensure adequate flexibility 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.

4.27. In accordance with Requirement 59 of SSR-3 [1], t~~T~~he design should ~~be such as to~~ minimize the generation of radioactive waste in all operational stages in the lifetime of the facility, including decommissioning. Design features should be implemented to ensure discharges remains below authorized limits. ~~Such~~ Design considerations for waste minimization should be compatible with the safety analysis and with regulatory limitations on radiation doses. Measures deployed in research reactors of different types generally include the following:

(a) Selection of materials that do not activate easily (e.g. use of plastic for the pneumatic ‘rabbit’ system irradiation target carriers) or materials that decay quickly when activated (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 of air spaces near neutron sources to reduce the production of ^{41}Ar.

4.28. The following features for limiting radiation exposure due to the radioactive waste generated during operation of the facility should be considered in the design:

(a) Provisions that reduce the quantity ~~and concentration~~ of the radioactive waste generated and transported within the facility and~~or~~ the concentration of effluent released to the environment;

(b) Provisions for the isolation of radioactive waste from site personnel and the public, with access control and zoning for radiation protection. For example, this might be accomplished by zoning the facility in accordance with the potential for radioactive contamination and radiation exposure;

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

(d) Provisions for the decontamination of personnel and equipment including the use of mobile devicesr;

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

4.29. 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.

4.30. The varying nature of the waste both as waste accumulates and as the circumstances that influence the characteristics of the waste vary should be taken into account in the design of the shielding, the containment and the isolation features associated with facilities, equipment and components for waste management.

Management of gaseous effluents

4.31. Measures for the management of gaseous effluents should be considered in the design, including the following:

(a) Provision for radioactive gases to be channelled through proper ducting as appropriate and brought to a common release point;

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

(c) Provision of appropriate filters to clean the gaseous effluent to minimize the discharges while also minimizing the production of secondary waste;

(d) Provision of means, such as stacks for the discharge of ~~gaseous~~ gases ~~low level radioactive waste~~, and of methods for sampling and monitoring those discharges.

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Management of liquid effluents

4.32. Provisions for the management of liquid effluents should be considered in the design, including 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) The potential for reconcentration downstream of some discharged radionuclides should be recognized and addressed in relation to the collection of 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 methods of monitoring such discharges;

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

(d) Provisions for decay devices (e.g. use of ~~delay tanks for~~ ~~^{16}N or~~ retention tanks) to minimize releases of radioactive material;

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

(f) Provisions for treating liquid effluents either for reuse (e.g. treatment using resins) or because the activity levels are too high for their release to the environment.

4.33. The following provisions should also be considered in the design:

(a) Provisions for measuring, ~~and~~ treating and controlling discharges of radioactive liquid effluents so that the amounts and the concentrations of discharges of radioactive material ~~are commensurate with the safety analysis report and~~ comply with the regulatory requirements [~~4~~16~~5~~];

(b) Provisions for filtration in liquid waste collection lines to prevent the release of solids. ~~Provisions for controlling the discharge of liquid radioactive material to the environment in compliance with the regulatory requirements [4].~~

Management of solid ~~effluents~~radioactive waste

4.34. The provisions for the management of solid radioactive waste (generated ~~in~~at the research reactor facility, including waste generated by ~~and~~ experiments)~~al devices~~ considered in the design should include the following:

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

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(b) The packaging, handling and storage of solid low level radioactive waste,

such as contaminated cleaning equipment, 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 internals, performed in on-site facilities;

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

(f) Equipment and tools for radiation protection;

(g) Provisions for compacting waste if required;

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

(i) Space for storing waste on-site and/or off-site until its transport;

~~(j) Provisions for filtration in liquid waste collection lines to prevent the release of solids;~~

~~(k)~~(j) Provisions for ensuring that any solid materials that may be discharged in liquid effluents are within authorized limits.

LIFETIME MANAGEMENT AND DECOMMISSIONING OF THE FACILITY

General

4.35. 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". The expected lifetime of the research reactor should be set out in the design. Issues that would affect the possible extension of the lifetime of the facility and its eventual decommissioning should be identified.

Considerations at the stages of design and construction

4.36. The design should include features for the storage and/or processing of radioactive waste generated throughout the lifetime of the facility.

DESIGN APPROACHES FOR DEALING WITH ACCIDENT CONDITIONS

4.37. The principal design measures that are taken to protect workers and the public against the possible radiological consequences of accidents should have the objectives of reducing the likelihood that accidents will occur (prevention of accidents) and reducing the source term (mitigation of the consequences) associated with accidents if they do occur [20~~3~~].

4.38. 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 facility and the risks

to site personnel from such releases and from direct radiation exposure. These design objectives should be achieved through high quality design and special features, such as protection and safety ~~and protection~~ systems, that are incorporated into the design of the facility. The aim is to ensure the reliable control of the reactivity of the reactor and, in particular for reactors with non-negligible thermal power levels and radioactive inventories, to ensure the removal of residual heat and the confinement of radioactive material. Achievement of the design objectives should be confirmed by means of a safety analysis. Deterministic safety analyses and the associated dose assessments, together with, as appropriate, complementary ~~with any~~ probabilistic safety analyses, used for demonstrating compliance with the radiation dose limits should be based on conservative assumptions for the analyses of design basis accidents and on realistic or best estimate assumptions for the analyses of ~~beyond design basis accidents.~~design extension conditions.

4.39. To achieve the design objectives mentioned, the necessary design provisions and procedures (e.g. shutdown systems, systems for residual heat removal, actions on receipt of alarms) should be such as to enable the operators to manage the event adequately.

4.40. Practices that are similar to those used for operational states should also be employed to ensure that the design will provide adequate radiation protection for site personnel and the public under accident conditions.

4.41. The design of the structures, systems and components for radiation protection under accident conditions may be developed through consultation with experts in radiation protection, reactor operations, design and accident analyses and regulatory matters. There should be continuous interaction among these experts throughout the design process to yield 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

4.42. The ~~requirements and recommendations for the~~ design of structures, systems and components for research reactors should be based on experience that has been gained in reducing radiation exposure at operating facilities, and ~~should~~ needs to be consistent with ~~Ref~~SSR-3. [1].

4.43. The following features for reducing radiation exposure should be incorporated into the system design:

(a) The work area in a high radiation zone that contains components requiring 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 of controlled areas with high radiation levels;

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

(d) Techniques for sampling radioactive liquids with minimal exposure necessary should be provided;

(e) Sedimentation of radioactive sludge in piping and containers should be prevented, and necessary countermeasures should be provided.

4.44. In seeking to minimize radiation exposure, the approach in design should be to provide for components of high reliability that require minimum surveillance, maintenance, testing and calibration (see para. 4.14).

4.45. Exposure of site personnel should be reduced by minimizing the possible amount of radioactive material in reactor components. Thus, traps and rough surfaces where active particulates can accumulate should be avoided as far as practicable.

4.46. Components and areas of buildings that may become contaminated should be designed for ease of decontamination. This includes using 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.

4.47. 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.

4.48. The uncontrolled buildup of particles containing radioactive substances 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.

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

4.50. 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.

DECONTAMINATION

4.51. The need for decontamination should be considered at the design stage. Provisions should be made for decontamination facilities if it is considered that a worthwhile reduction in radiation exposure would result.

4.52. 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

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decontamination.

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

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

DESIGN OF SHIELDING

4.55. In designing a shield for a specific radiation source, the target dose rate should be set on the basis of the expected frequency and duration of occupancy of the area. Account should also be taken in setting this target dose rate of the uncertainties associated with the strength of the radiation source and with the analysis made to determine the expected dose rate.

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

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

4.58. Losses in the efficiency of shielding may occur as a result of environmental conditions. The design of shielding should take into account effects due to interactions of neutrons and gamma radiation with the shielding materials (e.g. the burnup of radionuclides that have high neutron absorption cross-sections, radiolysis and embrittlement), those due to reactions with other materials (e.g. erosion or corrosion), and temperature effects (e.g. the removal of hydrogen and/or water from concrete). Penetrations through shielding (e.g. pipelines, ducts, etc.) should be appropriately designed so as to avoid direct streaming of radiation (e.g. use of multiple bends in pipes). The generation of heat in the shielding material should be taken into consideration and, if necessary, an appropriate cooling system should be designed for the shield.

4.59. 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 ~~elastic or~~ 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 ~~cross-section~~ threshold for nuclear inelastic scattering of the shielding material(s).

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

VENTILATION

4.61. Provision of a dedicated active ventilation system should be considered and, if necessary, such a system should be provided for maintaining appropriate clean conditions in working areas within the controlled area.

4.62. 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. Provisions should be made for meeting ventilation requirements in the event of an accident. These provisions should be specified in the design.

4.63. Both the spread of contamination and the amount of radioactive material released to the environment should be limited by providing features such as air cleaning filters and by maintaining appropriate pressure differentials between areas.

4.64. 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) 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) The need to ensure that the discharge point for exhaust air is not close to an intake point of the ventilation system.

4.65. 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 ~~regions~~areas.

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5. RADIATION PROTECTION IN OPERATION

INTRODUCTION

5.1. Exposure of people to radiation may be in the form of direct radiation, immersion, ingestion, inhalation or absorption through the skin or through wounds. The goals of radiation protection are to ensure the effective control of external doses and internal doses to workers and to the public, and of releases to the environment, to ensure conformance with all regulatory requirements and to enable further optimization[6] of operational practices. In accordance with Requirement 84 of SSR-3 [1], a~~A~~ radiation protection programme should be ~~put in place~~ established and implemented by the operating organization to achieve these goals~~[7]~~.

5.2. The basis of the radiation protection programme is a set of measures (design features, administrative controls, adequate training, etc.) aimed at reducing the radiation hazard and radiation doses. This includes imposing restrictions in the operating procedures in combination with an extensive monitoring system for radiation and contamination. Details of the contents of a radiation protection programme are set out in Ref. [~~7~~218].

5.3. Good housekeeping contributes to radiation protection. Control of contamination and of radiation exposure is greatly facilitated in a clean and tidy area. Good housekeeping should be encouraged by requiring operators and experimenters to arrange for the cleaning up and restoration of an area when they leave or at the completion of the work or the experiment, and to promptly identify, report and correct any equipment that is degraded, to maintain a high standard of material conditions at the facility.

5.4. At most research reactor~~s~~ facilities, areas accessible to the general public ~~are~~ 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. Doses to the public would generally only be incurred as a consequence of the various routine discharges of radioactive effluents from the reactor and its associated facilities. Nevertheless, any ~~direct~~ exposure pathway should be considered (e.g. the food chain, ground deposition).

5.5. Radioactive releases from research reactors arise in gaseous, liquid and solid forms. Control and monitoring of each of these three types of release should be carried out. These provisions will help to ensure that the doses from these releases are well below dose constraints for the public. With all types of releases, consideration should be given in the first instance to minimizing both the radioactivity content and the volume generated.

[6] Optimization: ~~exposures are required to be optimized such that the magnitudes of individual doses, the number of people exposed and the likelihood of incurring exposures are all kept as low as reasonably achievable, economic and social factors being taken into account~~ The optimization of protection and safety, when applied to the exposure of workers and members of the public, is a process for ensuring that the likelihood and magnitude of exposures and the number of individuals exposed are as low as reasonably achievable, with economic, societal and environmental factors taken into account. [2].

DOSE ~~LIMITATION~~LIMITS

5.6. The normal exposure of individuals should be restricted so that neither the total effective dose nor the total equivalent dose to relevant organs and tissues exceeds any relevant dose limit set by the national regulations. (Dose limits set by international bodies are provided in Ref. [2].)

Optimization principle and dose constraints

5.7. The operating organization should take appropriate measures to ensure that radiation protection 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. For each radiation source, the level of radiation protection provided should be optimized so that both individual and collective exposures (both normal and potential) are kept as low as reasonably achievable.

5.8. Optimization of the measures for radiation protection associated with any particular source or operation in a practice should be subject to dose constraints.

5.9. 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 [2, 21~~8~~].

5.10. To apply the optimization principle, individual doses should be assessed at the operational planning stage, and the predicted individual doses for the various options should be compared with the appropriate dose constraint. Options predicted to give doses that would exceed the constraint should be rejected or revised to bring the predicted doses within the constraint.

5.11. The regulatory body is responsible for ~~specifying the values~~ensuring the establishment or approval of dose constraints [~~4~~2]. However, rather than itself stipulating values of dose constraints, the regulatory body should generally encourage the operating organization to develop dose constraints, subject to regulatory approval.

5.12. The dose constraint should be used to ensure that the cumulative effects of all releases of radioactive material to the environment are restricted so that the effective dose due to normal or potential exposures for any member of the public, including populations distant from the radiation source and future generations, is unlikely to exceed any relevant dose limit.

~~REFERENCE~~ ADMINISTRATIVE CONTROL LEVELS

5.13. ~~Reference~~ Administrative control levels are values of measured quantities (e.g. dose rates, activity concentrations) above which some specified action or decision should be taken. These ~~system of reference~~ levels (recording levels, investigation levels, operational intervention levels ~~and action levels~~) ~~is~~ are an important part of a dose control system (discussed in detail in Ref. [2] ~~and successive Safety Guides and other publications~~). The ~~reference~~ administrative control levels should

be set by the operating organization on the basis of national regulations ~~on dose limits~~.

5.14. The recording level is a level of dose, exposure or intake at or above which the values of dose, exposure or intake received by workers are to be entered in their individual exposure records. The 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 should be conducted. (Further guidance on the use of recording levels and investigation levels is provided in Ref. [218].)

5.15. Operational ~~I~~intervention levels ~~and action levels~~ generally serve to protect members of the public in the event of an emergency. They should be set for the relevant protective actions but they should not allow certain levels of doses, for which intervention will almost always be justified, to be exceeded. The values of operational intervention levels included in emergency plans 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.

DOSE CONTROL ON THE SITE

Source control

5.16. Considerations for source control in design, operation and decommissioning are covered in this section, while physical controls and administrative controls are discussed from para. 5.25.

Control by material selection

5.17. A large reduction in potential exposure can be achieved at the design stage of reactors and experiments. In particular, for experiments and facilities, 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.

Control by quality of reactor coolant

5.18. The reactor coolant should be kept as free from impurities as possible to minimize dose rates from exposures due to coolant activation. This also has the advantage of minimizing problems of contamination. Nevertheless, 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. Maintaining good coolant chemistry has the additional advantage of reducing corrosion and thereby minimizing the activation of corrosion products.

5.19. Cleanup systems such as filters and ion exchange columns should be used to remove contaminants from reactor coolant. These can become significant sources of radiation, however. 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

5.20. Beam ports and similar devices can be significant sources of direct radiation and, therefore,

should be well designed, shielded, controlled and monitored. Beam ports that are not in use should be properly closed and shielded.

Control by minimizing the production of gaseous material

5.21. Noble gases produced in 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:

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

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

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

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

Control of irradiated fuel

5.23. Irradiated fuel is a significant source of radiation and should be handled and stored in such a way that the corrosion and failure of cladding, melting of fuel and the possibility of unintentional criticality are avoided. Specific guidance is provided in Ref.~~s~~ [18~~1~~, ~~12~~].

Control by minimization of waste

5.24. Radioactive waste produced in and around the reactor facilities is also a source of radiation and should be minimized to the extent possible. It should be segregated, contained and shielded, and placed in designated locations prior to its storage, ~~treatment~~ predisposal management or disposal.

Physical controls

5.25. In addition to appropriate controls on sources, physical controls should be put in place, such as the provision of shielding and ventilation, and separation in distance and time. Shielding is used to reduce the dose rates everywhere outside the shield, ventilation is used to control airborne radioactive material, distance is used to reduce the dose rate at occupied locations, and time limits are used to reduce the total dose for a given dose rate. The ways in which these factors apply in the operation of research reactors are considered in the following section.

Shielding

5.26. Shielding that is adequate for all foreseen variations in activity and dose rate should be provided. Adequate shielding should be provided for all sources of radiation in and around the

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reactor facility, including the sources identified in Section 3, as appropriate:

(a) The reactor core;

(b) The reactor coolant system;

(c) Filters;

(d) Ion exchange columns;

(e) Argon vent pipes;

(f) Beam ports and beam targets;

(g) Irradiated samples and equipment;

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

(i) Irradiated fuel.

5.27. 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

5.28. Ventilation is a simple 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. Depending on the design of the ~~reactor~~facility,

ventilation should be in place above the research reactor to control concentrations of ^{41}Ar and, in the event of fuel failure, fission product gases.

5.29. Ventilation systems with appropriate filtration should be considered and should be provided if necessary for use in normal operation. and in design basis accident conditions. ~~This~~ Depending on the design of the facility, the ventilation system ~~may~~ should include a separate subsystem with charcoal filters to be used in accident conditions with release of radioactive material and to mitigate the consequences for design extension conditions. ~~for use in accident conditions. In the case of a beyond design basis accident involving a release of fission products, the ventilation system may also be used for the mitigation of consequences.~~ In many research reactors, ventilation systems are essential for the fulfilment of the confinement function.

5.30. 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.

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Distance

5.31. Personnel should be made aware of the need to make effective use of distance and to stay in regions of lower dose rates whenever possible. The use of special tools may be beneficial to 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

5.32. Time spent in hazardous areas should always be minimized. This may be the only viable method of dose control in some cases, where the dose rates are high in spite of shielding and distance protection. For tasks that must 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 not only enables the workers to refine the procedure and to perform the task faster, but also provides an opportunity to anticipate problems before they arise.

Decay

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

Decontamination

5.34. Routine operations and experiments involve the handling of contaminated components and equipment, and may 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. A proper procedure should be established in which both the contaminant and the contaminated material are considered.

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

5.36. To facilitate efficient decontamination procedures, potentially contaminated surfaces 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

5.37. Priority should be given to preventing, eliminating or controlling the sources of radiation hazards. However, there can be situations or working areas where residual sources of airborne and surface contamination continue to pose an exposure hazard. In such areas where personnelal contamination is possible, appropriate protective clothing and equipment should be provided.

5.38. Protective clothing and equipment should be chosen in accordance with the degree of hazard

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anticipated, 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 are useful. The protective clothes and equipment should be used in accordance with appropriate procedures, and training and retraining in their usage should be included in these procedures. Further details are provided in Annex II.

5.39. Tasks requiring the use of some specific personal protective equipment, such as breathing equipment, should be assigned only to workers who, on the basis of medical advice (see ~~Section 7 in~~ Ref. [21~~8~~] ~~on health surveillance~~), are capable of safely sustaining the extra effort necessary and have been trained as required.

5.40. Whenever the use of personal protective equipment is considered for any given task, consideration should be given to any additional exposure that could result owing to the additional time taken because of its inconvenience.

Administrative controls

5.41. Administrative controls defined in local rules, requirements and 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.

5.42. In administrative controls, natural human behaviour and tendencies should be taken into account. The need for relying on administrative controls should be minimized by providing appropriate protective measures and safety measures, including well engineered controls, and satisfactory working conditions.

5.43. A workplace monitoring programme should be established, maintained and regularly reviewed. The programme may include monitoring of:

(a) Leaks by means of a ‘leak and wipe’ test;

(b) External ionizing radiation;

(c) Surface contamination;

(d) Airborne contamination.

Area classification and access control

5.44. Depending on the potential for radiation hazard, the areas of the research reactor facility should be classified. The area classification should be indicated with appropriate signs to give warning of the hazard. To minimize the potential for ingestion of radioactive material, actions such as eating, drinking, smoking or applying cosmetics should be prohibited in areas presenting contamination hazards. In accordance with Ref. [2], the areas of the facilities should be classified as described in the following section.

5.45. *Controlled area:* An area in which specific protective measures or safety provisions are required for:

(a) Controlling normal exposures or preventing the spread of contamination in normal working conditions;

(b) Preventing or limiting the extent of potential exposures.

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 required procedures for protection and safety should be taken into account. Controlled areas are normally surrounded by supervised areas.

5.47. Controlled areas should be:

(a) Delineated by appropriate physical barriers or other suitable means (which could include locks and interlocks), the degree of restriction being commensurate with the magnitude and likelihood of the potential exposures;

(b) Indicated by warning symbols and appropriate instructions in appropriate languages at access points and other appropriate locations within controlled areas;

(c) Provided with measures for occupational protection and safety, including local rules and procedures;

(d) Regulated by means of administrative procedures, restrictions on access, permits for work, etc.;

(e) Equipped at entrances with appropriate protective clothing and devices and monitoring instrumentation;

(f) Provided, as appropriate, at exits with monitoring equipment for contamination of skin, clothing and any object or substance being removed from the area, with washing or showering facilities and with suitable storage for contaminated protective clothing and equipment;

(g) Reviewed periodically to determine the possible need to revise the protective measures or safety provisions, or their boundaries.

5.48. In some parts of the controlled area, compliance with relevant limits 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 setting conditions for restricted and special entries. Local rules or radiation work permits may be used for administrative control of admittance into these zones.

5.49. Warning symbols such as those recommended by the International Organization for Standardization (ISO) and appropriate information (such as information on radiation levels or

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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 specified zones and at other appropriate locations within the controlled area. Persons crossing a zone boundary should be made aware immediately (e.g. by using 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. *Supervised area:* An area where occupational exposure conditions need to be kept under review even though specific protective measures and safety provisions are not normally needed.

5.51. Supervised areas should be:

(a) Delineated by appropriate means;

(b) Indicated by approved signs at appropriate access points;

(c) Periodically reviewed to determine any need for protective measures and safety provisions or changes to their boundaries.

Approval

5.52. A formal approval is required for undertaking work involving the control of radiological hazards. The application for such an approval also enhances the awareness of personnel with regard to the radiological hazard. Submissions should be required and justified in accordance with appropriate criteria in most cases, 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. Types of approvals that may be considered are listed in the following discussion.

5.53. *Work permits:* Work permits are approvals to perform some work in areas where the worker is subject to radiological and/or industrial hazards. Work permits are approved within the reactor facility 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 officer;

(c) Identification of workers and their previous doses;

(d) Qualification of the workers;

(e) Exposure control data, doses or time authorized, actual doses received, time in and out;

(f) Radiological conditions, preliminary survey results;

(g) Dosimetry required;

(h) Protective clothing and equipment required;

(i) Special precautions or instructions;

(j) Actions on completion of radiation work;

(k) Provisions for restoration of the normal configuration;

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

(m) Authorization signatures.

5.54. *Experiment approvals:* All new experiments should go through a process that includes review and approval by the safety committee (see Ref. [1], ~~para. 4.15~~Requirement 6) and the regulatory body, as required (further guidance is provided in Ref. [1~~5~~]). The review should include an assessment of the radiological hazards likely to arise during normal and abnormal operation of the experiment (including commissioning, decommissioning, ~~and handling~~ predisposal management and disposal of waste and activated components).

5.55. *Routine irradiation approvals:* Routine operations, such as isotope production or irradiation of samples for activation analysis, should have a practical procedure for control such as a simple irradiation form containing only the most important information. They should be approved within the reactor facility. Non-routine irradiations should be treated as a new experiment (see para. 5.54).

5.56. *Modifications to the facility:* The radiological implications of any modification to the facility should be examined. Implementation of any modifications may need the review and approval of the safety committee and the regulatory body (see Ref. [1~~5~~]).

5.57. *Procedures:* Procedures should be prepared and approved for all activities that have radiological significance, and for all routine activities for radiation protection purposes. These procedures and their modification should be subject to review and approval, as required (by the reactor manager, safety committee and regulatory body). Methods should be established to ensure that current and valid procedures are in use. The management system should cover all radiation protection procedures (for further guidance see Ref. [24~~13~~]).

OFF-SITE (PUBLIC) DOSE CONTROL BY MINIMIZATION OF RELEASES

5.58. A dose constraint should be defined for a research reactor to ensure that the effective dose to ~~any member of the critical group~~the representative person is unlikely to exceed any relevant dose limit, with account taken of cumulative releases from all sources and practices under control.

Limitation of discharges

5.59. Authorized limits on discharges should be set for the facility. These authorized limits should be consistent with public dose constraints, as determined by using appropriate release and dispersion models. Authorized limits should be set on the discharges per unit time (i.e. weekly, monthly and annually).

5.60. Authorized limits on discharges for a given facility should not be exceeded. However, the management of a research reactor may set its own ~~reference~~ administrative control levels, derived

on the basis of optimization and within the authorized limits.

Record keeping for releases and waste

5.61. Releases of gaseous and liquid radioactive material should be recorded to demonstrate compliance with authorized limits on discharges. Records should be kept of the generated waste, including the planned duration and place of temporary on-site storage.

6. OPERATIONAL RADIOACTIVE WASTE MANAGEMENT

GENERAL

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) Type of reactorfacility;

(b) Specific design features;

(c) Operating procedures and practices;

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

(e) Maintenance;

(f) Modifications to the facility and life extension activities;

(g) Refuelling and operational occurrences;

(h) Operational history of the facility;

(i) Integrity of the fuel.

6.2. In accordance with Requirement 85 of SSR-3 [1], ~~T~~the operating organization ~~should~~ needs to establish and implement, as part of its plans and arrangements for safety management, a programme for the management of radioactive waste. A graded approach needs to be considered in the application of the requirements and ~~T~~the programme should include provisions for:

(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 based on 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 retrieved at the end of the storage period;

(g) ~~T~~Pretreatment~~ing~~, ~~retreating~~ 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 sources (of effluent discharges) and the environment, for the demonstration of regulatory compliance;

(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 containment for the radioactive waste in the storage location;

(m) Monitoring changes in the characteristics of radioactive waste by means of inspection and regular analysis, in particular, 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) Adoption and implementation of corrective actions on the basis of the results of monitoring.

Requirements for the predisposal management of radioactive waste are provided in GSR Part 5 [154], and further guidance is provided in SSG-40 [176].

GENERATION OF RADIOACTIVE WASTE

Gaseous radioactive waste

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

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

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

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

(a) The areas with potential for the generation of gaseous waste should have provisions for the renewal of air or the cover gas and ~~its purification through~~capturing of waste using 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 ~~such as nitrogen~~ 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

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

(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 waste water 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).

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

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

(b) Reducing leakage from the various systems;

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

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(f) Optimizing decontamination procedures;

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

Solid radioactive wastes

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

(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 (gloves, tissue paper, disposable glassware, etc.);

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

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

6.8. The generation of solid radioactive waste should be kept to the minimum practicable level by adopting suitable measures, such as:

(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 the controlled area;

(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 the pneumatic rabbit system), and reuse and recycle of materials wherever possible (e.g. use of titanium cans)

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OPTIMIZING EXPOSURE ASSOCIATED WITH THE MANAGEMENT OF RADIOACTIVE WASTE

6.9. In accordance with requirement 34 of SSR 3 [1], ~~T~~the operating organization should ensure that exposures of workers and members of the public due to waste management operations are as low as reasonably achievable by optimizing arrangements for radiation protection for the handling, treatment, transport, storage and transfer of radioactive waste.

6.10. In accordance with requirement 85 of SSR 3 [1] and requirement 19 of GSR Part 5 [15~~4~~] Such exposures should be maintained as low as reasonably achievable by adopting suitable measures, including:

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

(b) Following documented procedures for the handling, pretreatment, treatment, conditioning, transport, storage and ~~transfer or ultimate~~ disposal of all radioactive waste;

(c) Use of suitable equipment and shielding, and a proper monitoring system;

(d) Use of adequate ventilation for gaseous effluents and control of releases during normal operation or abnormal conditions;

(e) Clearly specifying limitations on ~~waste~~ authorized discharges for each ~~disposal~~ disposition pathway, with use made of reasonably conservative methods and modelling;

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

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

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

(i) Following documented procedures on the periodic measurement and survey of radiation levels outside any radioactive waste store.

PACKAGING AND CONFINEMENT OF RADIOACTIVE WASTE

6.11. In accordance with requirement 10 of GSR Part 5 [15~~4~~] ~~R~~radioactive waste arising from the operation of research reactors should 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.

6.12. The potential for migration or dispersion of radionuclides in radioactive waste should be minimized and the external dose rate should be limited by adopting suitable measures, such as:

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

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(b) Following appropriate documented procedures for the segregation, treatment and packaging of radioactive waste, on the basis of the physical and chemical form and the pyrophoric or other hazardous nature of the ~~radionuclide~~waste, and the method of storage and~~/or~~ eventual disposal;

(c) Applying documented procedures to ensure that containers are clearly labelled with the radiation warning sign, 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 and recording dose rate measurements at the surface of each package and at specified distances from the surface of each package to ensure compliance with the approved specifications;

(e) Use of spillage trays of adequate volume;

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

6.13. Radioactive waste varies in terms of both its activity content and its chemical composition, which in turn determine the suitable disposal options. It should be emphasized that the parameters of key importance for one particular option for waste disposal may be different from the most important parameters for another option. Ultimately, this is a matter that must be clarified between the waste generator, the operating organizations of disposal facilities and the regulatory bodies. There may be prescriptive regulatory requirements in place or other arrangements approved by the regulatory body. Account should be taken of the important parameters for waste packages as described in Section 4 of Ref. [25~~16~~].

STORAGE OF RADIOACTIVE WASTE

6.14. Management of radioactive waste involves storage for various reasons; some relate to operational demands and others to the unavailability of suitable disposal facilities. For whatever reason the waste is stored, safety (including criticality safety as necessary) has to be considered over the time periods involved. In the case of heat-generating packages, the need for cooling can pose additional demands on the waste package. As stated in Requirement 11 of GSR Part 5 [15~~4~~], "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". During storage, the waste package should be able to remain within specifications and, if necessary, overpacked with ease when subsequently retrieved for transport and disposal. Since some ageing of the package is probably inevitable, 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 may be partially susceptible to ageing and, in extreme cases, remedial action may be necessary. The material used for packaging may show radiation effects that affect the strength of the material. Radiation effects,

as well as the biodegradation of organic materials, may result in the generation of gases. If necessary, passive ventilation designs may be used to relieve internal pressure. Passive ventilation systems should be designed to minimize the entry of moisture and the ~~passage~~ release of waste material from the container.

6.15. 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, property and the environment against the radiological hazards associated with radioactive waste.

6.16. The location, design, construction, operation and maintenance of storage facilities should demonstrate the following characteristics:

(a) The store should be appropriately located to minimize the impact of potential natural or human-made hazards;

(b) The store should be sited above groundwater level;

(c) Durable fire resistant material should be used for the construction of the store;

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

(e) The store should be designed to control any contamination from gaseous or liquid radioactive releases that could conceivably occur;

(f) The store surfaces should be constructed for ease of decontamination;

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

(h) The capacity should be adequate to accept the maximum operational

holdings anticipated from the system in normal conditions and following anticipated operational occurrences;

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

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

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

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

(m) The maintenance programme should be documented, with records kept of all maintenance activities;

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(n) Appropriate radiation warning signs should be displayed on each entrance to the store;

(o) Provision should be made for an extraction system to extract gases or vapours originating from the controlled material;

(p) In the event of a fire in or around the storage facility, the dispersion of radioactive material should be prevented.

6.17. The development of the radioactive waste store is subject to the requirements established in Ref. [2617].

6.18. The packaging of radioactive waste should be suitable for the type of storage and for the foreseeable time frame of storage.

6.19. Documented procedures should be followed to ensure that short lived radioactive waste is stored so as to allow it to decay to a level of activity at which it can be cleared from regulatory control [2718].

6.20. Documented procedures should be applied to the use of storage containers for short lived radioactive waste. These should include clear labelling with details of the waste (content, activity, date of storage, responsible person, etc.).

6.21. An accurate inventory should be kept of all the waste packages and containers and their contents in the waste store at any time, following appropriate documented procedures.

6.22. Documented procedures should be followed for the periodic inspection and monitoring of stored radioactive waste, storage facilities and their environment.

6.23. The integrity of the waste packages in storage should be ensured and the storage facility should be capable of maintaining the 'as received' integrity of the waste package until it is retrieved for further treatment, conditioning or disposal [1754].

6.24. Radiation monitoring and visual inspection should be performed whenever the waste is handled or moved (placed into storage, retrieved or transported on or off the site) to protect the operator, to prevent the accidental spread of contamination and to provide an additional check for the record keeping system [1754].

DOCUMENTATION RELATING TO THE MANAGEMENT OF RADIOACTIVE WASTE

6.25. Documentation detailing the nature of the radioactive waste arising from the operation of the research reactor, the location of the waste, and all safety and security measures should be maintained.

6.26. A register or database should be kept that includes:

(a) The radionuclide type and content (physical, chemical and radiological characteristics);

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

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discharge and disposal);

(c) The waste matrix for immobilization;

(d) The method of treatment or conditioning;

(e) The identification number of the package.

6.27. The documented procedures and systems for the handling, pretreatment, treatment, conditioning, transport, storage and ~~ultimate~~ disposal of radioactive waste should be periodically reviewed.

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

6.29. Records of effluents should be periodically inspected.

6.30. Documentation detailing compliance with regulatory requirements should be properly maintained.

CLASSIFICATION AND CHARACTERIZATION OF RADIOACTIVE WASTE

6.31. The classification of radioactive waste assists in the development of management strategies and in the operational management of the waste. 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. To make the appropriate segregation of waste, it will be necessary to know its properties and, hence, it will be necessary to characterize the waste at various stages of its processing. Documented procedures should be followed for the characterization of radioactive waste and its segregation, and for assigning the waste to a particular class.

6.32. Details of the purpose, methods and approaches to the classification of radioactive waste are provided in Ref. [28~~19~~]. The distinction between radioactive waste and waste that can be cleared from regulatory control is addressed in Ref. [27~~18~~].

6.33. The programme for radioactive waste management of the operating organization 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:

(a) Its origin;

(b) Criticality;

(c) Its radiological properties (e.g. half-life, activity and 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.

6.34. One widely used classification system separates radioactive waste into three classes: low level, intermediate level and high level waste. A further distinction is made between short lived and long lived waste. In addition, special care should be taken with waste containing alpha emitting radionuclides, which could arise from failed fuel. Waste containing inflammable, pyrophoric, corrosive or other hazardous materials should also be treated with great care. Mixing of such waste with waste of other types should be avoided.

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

6.36. Liquid radioactive waste, principally water based, should be considered for processing in terms of activity concentration and the content of chemical substances. For example, 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.

6.37. Solid radioactive waste should be considered in accordance with its nature and activity concentration. For example, 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) may be segregated in accordance with the type of treatment and conditioning (e.g. compaction, incineration, immobilization) adopted for its processing.

6.38. The characterization of waste is performed for numerous reasons, including the need for a safety assessment of facilities for its treatment, conditioning and storage. Waste characterization is also necessary to qualify the treatment and conditioning processes, and to perform quality control of waste forms and packages during conditioning. In general, the waste generator is responsible for waste characterization, and related data are used as the bases for acceptance for ultimate disposal. Details of criteria, bases and other parameters as they relate to waste characterization are provided in Ref. [~~17~~2~~1~~6].

6.39. Waste characterization should be a necessity 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.

PROCESSING OF RADIOACTIVE WASTE

6.40. Material that falls within the definition of radioactive waste is required to be processed accordingly. Such processing may yield effluent that is suitable for authorized discharge or material suitable for some other authorized use, or material that meets the criteria for clearance from regulatory control [2718]. Elements of waste processing, including pretreatment, treatment and conditioning, are described in Sections 5.10–5.20 of Ref. [20156].

6.41. The radioactive waste generated by the operation of the research reactor facility should be processed in accordance with established procedures.

6.42. Radioactive waste should be processed as early as practicable to convert it into a passively safe state and to reduce the likelihood of its dispersal during activities related to its storage and disposal.

6.43. Waste packages resulting from the conditioning of radioactive waste should be in accordance with established criteria and should comply with the limits and conditions for the handling, transport, storage and disposal of radioactive waste.

Gaseous radioactive waste

6.44. 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.

6.45. The air within the reactor building can become contaminated owing to the leakage of gases or vapours from systems containing radioactive fluids, or from the resuspension of fine particulate material that may exist within the facility as surface contamination, or from fine particulate material being handled in the facilities. Ventilation systems are designed to collect general ventilation air whose likelihood of contamination is low. Such general ventilation air should be continuously monitored for contamination and, if necessary, treated before discharge from the facility. Such treatment may take the form of filtration for particulate aerosol materials or adsorption for volatile species. It should be ensured that gaseous contaminants are not present in quantities that would warrant special treatment.

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

6.47. Certain systems within the facility may also require 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

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prior to systems being opened for operational reasons or for maintenance purposes. Such off-gas systems may contain significant amounts of radioactive contamination and their treatment may require storage for radioactive decay, adsorption of volatile species by particulate filtration, or other chemical and physical cleaning methods.

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

6.49. The design of gaseous waste treatment systems will depend on the nature of the facility and the activities to be carried out within the facility. An assessment of the gaseous wastes that will be generated should be carried out as part of the safety assessment for the facility. This assessment should cover the fitness for the intended purpose of the system in terms of capacity and efficiency, consideration being given to any properties of the gases, such as corrosiveness, flammability or explosiveness, that may affect the system.

6.50. In assessing any changes to equipment or modes of operation of the ~~reactor~~ 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 may arise and their possible effects on the capacity of the treatment systems for gaseous radioactive waste.

Liquid radioactive waste

6.51. 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.

6.52. For the effective processing of liquid radioactive waste, the following should be considered:

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

(b) Waste of higher activity should not be diluted with lower activity waste;

(c) Radioactive waste containing dissolved or dispersed solids of higher activity should not be mixed with waste containing such solids of lower activity, since this would complicate the processing of the latter;

(d) The chemical compatibility of different waste streams and of the radioactive waste and the equipment should be considered;

(e) 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;

(f) 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;

(g) 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;

(h) Strict control over all parameters relevant to proper waste processing should be maintained and recommendations for safety and protection should be observed.

6.53. For waste conditioning, 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

6.54. Solid radioactive waste may be inhomogeneous. Special consideration should be given to representative sampling before processing so as to confirm compatibility with the intended process, and appropriate practicable arrangements should be made for such sampling.

6.55. A great number of processes are available for producing an acceptable waste form. Such processes should be selected on the basis of the characteristics of the waste concerned, with due account taken of radioactive decay. If possible, processes with high factors of volume reduction should be applied with the use of proven techniques, such as compaction and incineration.

TRANSPORT OF RADIOACTIVE WASTE

6.56. Radioactive waste may be processed within the reactor facility or at an on-site or off-site facility, depending on the local arrangements. On-site movement of radioactive waste may not need to meet all the same requirements as for off-site transport, since the arrangements will be under the control of the operating organization. However, appropriate procedures and arrangements should be followed for the on-site movement of radioactive waste. For transport off the site, regulatory requirements should be satisfied. The regulatory requirements should be consistent with the IAEA Regulations for the Safe Transport of Radioactive Material [2901].

6.57. Established radiation protection procedures and radioactive handling procedures should be followed to minimize the exposure of site personnel and to ensure the safety of the on-site movement of radioactive waste.

DISCHARGE CONTROL AND COMPLIANCE MONITORING

6.58. The operating organization should apply to the regulatory body for authorization to discharge gaseous and liquid radioactive waste from normal operation, indicating the expected levels prior to the commencement of the operation. Guidance on the establishment of such discharge levels can be found in Ref. [4165]. 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

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when applied to the ~~critical group~~representative person and that such doses to members of the public are as low as reasonably achievable. The expected discharges from all operational states of the reactor facility and from all possible future changes in operation should be considered in setting the proposed levels of discharges.

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 ~~critical group~~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 submission from the operating organization in which the discharge levels are indicated, should set the authorized limits. All discharges should be within these authorized limits.

6.61. Compliance with authorized limits on discharges should be demonstrated by means of monitoring, involving approved methods of sampling and measurements. The results of monitoring should be recorded.

6.62. Arrangements should be put in place for dealing with any abnormal releases that exceed the ~~reference~~ 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, site emergency response arrangements should be ~~effected~~ implemented as necessary.

7. MONITORING

INTRODUCTION

7.1. An appropriate surveillance programme should be established, maintained and kept under review for the systematic evaluation of the radiological conditions at the facility, with the following objectives:

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(a) Monitor exposures of individual workers;

(b) Control any environmental impacts;

(c) Assess trends in radiological parameters;

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

The following paragraphs present the basic provisions for surveillance.

7.2. The sensitivity of monitoring systems should be sufficient to ensure that limits and constraints (occupational dose limits, dose constraints, ~~reference levels,~~ authorized limits on discharge, etc.) set for the quantities measured are not exceeded.

7.3. Records of the findings of the monitoring programme at the facility should be kept appropriately and relevant data should be made available to the workers.

Types of surveys

7.4. The types of surveys that are needed in any workplace depend on the types and forms of the radionuclides and radiations that may be present, and on the objectives of the particular survey. Factors to be considered in planning and performing surveys should include:

(a) Hazard level (type of work and time spent);

(b) Requirements of the regulatory body or other competent authorities;

(c) Radionuclides and radiation sources;

(d) Physical forms of the radionuclides (solid, liquid, gaseous, airborne or particulate);

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

(f) Contamination (loose or fixed, detected by smears or direct counting);

(g) Background radiation levels or contamination levels;

(h) Arrangement of the workplace.

Three major groups of surveys should be conducted for radiation protection purposes: routine surveys, task related surveys and special surveys.

Routine surveys

7.5. Routine surveys should be conducted to assess workplaces for radiation dose rate, surface contamination and airborne contamination. 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 zone designations or operating procedures. Factors that may affect the type and frequency of each of these surveys will include:

(a) Typical dose rates and contamination levels normally found;

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(b) Degree of occupancy;

(c) Quantities and types of radionuclides encountered;

(d) Types of work or experiments;

(e) Changes in radiological conditions in the area.

Task related surveys

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

Special surveys

7.7. Special surveys should normally be undertaken:

(a) At the commissioning stage for new facilities;

(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 carried out under abnormal circumstances, such as those following an incident ~~accident~~ or ~~an~~ an accident ~~incident~~.

7.8. Special surveys may also be undertaken to address specific problems, for example:

(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').

7.9. For each type of survey, ~~reference~~ administrative control levels should be determined together with the action required if the ~~reference~~ level is exceeded. The data should be checked to determine whether they are consistent with the type of work being performed and, if appropriate, with the personal dosimetry records.

FACILITY MONITORING

Area monitoring

7.10. Instruments of various types, both fixed and portable, are used at research reactor facilities. Most of the required instrument types are listed in the following discussion, together with the types

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of measurement for which they are suitable.

Field monitoring

7.11. *Fixed radiation monitors.* Area gamma and neutron monitors with local alarms are used to indicate changes in radiation fields. Local dose rate indication and failure detection capability are provided with most systems. Redundancy should be considered in the placing of area monitors. Dose rate indication and alarms should be relayed to the main control room. Area gamma monitors should be installed in areas where the radiological conditions may change frequently or unexpectedly. Area gamma 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 and ~~ion~~ other filter systems, at locations where handling for experiments is performed and in the main control room.

7.12. *Portable radiation monitors.* Portable measuring instruments include gamma survey meters and neutron dose equivalent meters. The use of portable instruments is helpful, provided that the place and time of measurements are specified and recorded.

Contamination monitoring

7.13. *Surface contamination.* Various instruments and methods may be used for monitoring surface contamination, some of which are listed in the following:

- Direct measurements of surface contamination may be made with portable instruments. Instruments should be available for alpha and/or beta–gamma contamination.
- Indirect measurements may be made using smear sampling. The smear samples may be measured in an area of low background radiation with a shielded counter.
- 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.
- Personnel contamination monitors should be located at exits and include:

  — Portal (pass-through type) monitors;

  — Hand and foot monitors.

- Special purpose monitors should be available for specific uses. Examples are laundry monitors, solid waste material monitors and floor monitors.

7.14. *Airborne contamination.* Monitoring should be carried out 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. Sometimes area gamma monitors are used at locations of interest to warn of increasing airborne contamination. In this case, care should be taken to distinguish between dose rates from airborne contamination and those from other possible sources.

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7.15. *Liquid contamination.* Monitoring may be carried out by continuous on- line measurement or by sampling and off-line analysis. Gamma monitors located close to pipes carrying liquids that may possibly be contaminated may be used to monitor contamination levels if the influences of other sources are known and can be taken into account.

7.16. *Monitoring of shipments of radioactive material.* Any shipment of radioactive material that is received at or dispatched from the facility should be checked for both external surface contamination and radiation dose rates, which should conform to relevant national or international regulations (for further details see Ref. [~~21~~2~~90~~]). The measured results should be recorded, and copies of the recorded results should be attached to the dispatched shipment and archived in accordance with the relevant regulations.

7.17. *Monitoring of solid waste*. Stored solid waste should be checked for external radiation and for possible surface contamination. In the event of the detection of external contamination, appropriate identification of the contaminating radionuclides should be performed. An appropriate record keeping system should be established and maintained to provide accountability for and traceability of solid waste.

INDIVIDUAL MONITORING

7.18. Dosimetry services should be provided for assessing the effective doses (external and internal) received by persons. Effective doses and dose equivalents should be recorded to demonstrate compliance with regulatory requirements. Records of such data make further analyses possible for determining trends and future needs (for further details, see Ref. [~~22~~2218]).

7.19. The necessary nature, frequency and precision of individual monitoring should be determined with due consideration of the magnitude and possible fluctuation of exposure levels, and the likelihood and possible magnitude of potential exposures.

Measurement of external doses

7.20. The operating organization should be responsible for the assessment of the occupational radiation exposure of the workers on the basis of individual monitoring, where appropriate, and should ensure that adequate arrangements for individual monitoring are in place with appropriate dosimetry services.

7.21. The assessment of individual external exposure is readily performed by means of individual monitoring. For routine monitoring, integrating personal dosimeters should be worn. These dosimeters should be processed and the results should be determined at appropriate intervals by a competent monitoring service. Task related and special individual monitoring is normally performed with real time self-reading dosimeters, often with additional warning functions.

7.22. As stated in para 3.100 of GSR Part 3 [2], "For any worker who ~~is normally employed~~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 ~~of the worker~~ shall be assessed on the basis of the results of workplace monitoring ~~of the workplace~~ and ~~of~~ information on the locations and durations of exposure of the worker" ~~(Ref. [2], para. I.33)~~.

7.23. As stated in para 3.101 of GSR Part 3 [2], "For any worker who ~~is~~ regularly ~~employed~~ works in a supervised area or who enters a controlled area only occasionally, the ~~individual monitoring shall not be required but the~~ occupational exposure ~~of the worker~~ shall be assessed ~~. This assessment shall be~~ on the basis of the results of workplace monitoring ~~the workplace~~ or individual monitoring, as appropriate" ~~(Ref. [2], para. I.34)~~.

Dosimetry for beta–gamma radiation

7.24. The most common devices for personnel monitoring for uniform whole body beta–gamma radiation are film dosimeters and thermoluminescent dosimeters. Particular care should be taken when personnel work near radiation beams, since exposures may be highly non-uniform (for further guidance, see Ref. [~~23~~218~~8~~]).

7.25. Both film dosimeters and thermoluminescent dosimeters are suitable for routine issue over periods of one to three months. For the purposes of short term dose control, it is necessary that these dosimeters are supplemented with a direct reading dosimeter. In addition, 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 a dosimeter incorporating an alarm that sounds if a preset dose or dose rate is reached.

Dosimetry for neutron radiation

7.26. Dosimetry for neutron radiation is at present technically one of the most difficult types of dosimetry. The following detectors are most commonly used for neutron dosimetry:

(a) Nuclear track detectors;

(b) Activation detectors;

(c) Bubble detectors;

(d) Electronic dosimeters.

Film dosimeters and thermoluminescent dosimeters may be arranged in such a way that distinction is possible between doses due to neutrons and doses due to radiation of other types. In cases where detectors of the types mentioned previously are not available, 'rem meters' should be used for the assessment of neutron doses.

Extremity dosimetry

7.27. Thermoluminescent dosimeters are the best extremity dosimeters available, mainly owing to

their small size and low weight. It should be ensured that personnel are trained to wear the dosimeters appropriately by placing the detector on the extremity closest to the source.

Internal dose measurement

7.28. Persons who work under conditions in which internal exposures may occur should be appropriately monitored. This monitoring should be performed on a routine as well as an occasional basis, depending on the particular working conditions.

7.29. The determination of committed effective doses from internally deposited radionuclides (internal dose) is seldom straightforward [~~24~~2218~~8~~]. The internal dose should be evaluated using computational models from the quantity of incorporated radioactive material deposited in various parts of the body. The following paragraphs summarize the methods commonly used. If none of the listed methods is appropriate, adequate and feasible, the internal dose of the person should be assessed on the basis of workplace monitoring.

7.30. *Bioassay*. Bioassay involves taking biological samples (e.g. urine, faeces, nose swabs, breath) and measuring their activity. From the activity of the samples, the internal contamination can be assessed and the intake of the person and the resulting internal dose can be calculated using models.

7.31. *Whole body monitoring*. Whole body monitoring is 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 of the person and the resulting internal dose can be calculated using models. Whole body counting should be performed periodically.

7.32. *Partial body monitoring.* Partial body monitoring may be adequate where the radionuclides of concern tend to be concentrated in a specific organ (e.g. iodine in the thyroid gland) [~~24~~2218~~8~~]. To measure the activity in a specific organ, a relatively simple measuring set-up (portable detector) may be used.

EFFLUENT MONITORING

Source monitoring

7.33. Source monitoring refers to the measurement both of discharges and of the radiation field around the source itself. It should be ensured that the source monitoring programme enables verification of compliance with the authorized limits on discharges as prescribed by the regulatory body. The monitoring of radioactive discharges may entail making measurements for specific radionuclides or gross activity measurements, as appropriate. Measurements should be made before release or at the point of release (for example, at the stack for atmospheric discharges or at the pipeline for liquid discharges). For batch discharges, the radioactive material is more appropriately characterized by the volume of the batch and the radionuclide composition of a sample taken at the reservoir from the homogenized batch prior to discharge.

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

(a) On-line monitoring of discharges;

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

(c) Intermittent sampling and laboratory measurements of activity concentrations in the sample.

7.35. The choice of sampling method and measurement procedures should depend on:

(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 discharges that require prompt detection and notification.

Monitoring of airborne releases

7.36. Radioactive material to be released to the atmosphere is generally measured at various points in the ventilation system. Measurements performed in the stack (stack monitoring) have special importance because they provide data on the actual discharges. Airborne 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. Stack monitoring and reporting of the measured results are required by the regulatory body in several States. The results of stack monitoring should be made available at the main control room for quick action in the event that the release is higher than the relevant ~~reference~~ administrative control levels.

7.37. *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 or beta 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.

7.38. *Particulate monitoring*. Particulate activity 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 routinely identified to enable an assessment of the source of the activity and the potential for reducing the release.

7.39. *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 filled (or zeolite) cartridges may be used as a collecting medium for measurements of halogen isotopes such as radioiodines. 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 may be present, or when experiments or irradiations are being conducted in which iodine may be released.

Monitoring of liquid releases

7.40. The concentration and total activity of liquids that may potentially be radioactive should always be checked before their release in the environment. This usually means that they should be 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 measurements of low energy beta, tritium and alpha activity, and identification of radionuclides by gamma spectrometry. Parameters for chemical characteristics should also be monitored to take account of characteristics that may affect transport processes (e.g. the acidity (pH) may affect soil retention properties).

ENVIRONMENTAL MONITORING

7.41. An environmental monitoring programme should be conducted in accordance with the requirements of the regulatory body. A pre-operational programme for environmental monitoring should be conducted two to three years before the planned commissioning of the facility. The pre-operational programme should provide for the measurement of background radiation levels in the vicinity of the facility and their variation over and between the seasons. It should also provide the basis for the operational programme of environmental monitoring and should include the routine 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.

7.42. The operational programme of environmental monitoring should be conducted as an extension of the pre-operational programme; guidance is provided in Ref. [4165]. 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 programme should be designed to provide information for:

(a) Confirming the adequacy of control over effluent discharges;

(b) Correlating results of environmental monitoring with data obtained from monitoring at the

source of the discharges;

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

(d) Assessing trends in the concentration of radionuclides in the environment.

8. INSTRUMENTATION

INTRODUCTION

8.1. The radiation protection programme of the research reactor facility determines the type and number of instruments required for individual monitoring and area monitoring.

~~REQUIREMENTS~~ RECOMMENDATIONS FOR INSTRUMENTS

Number of instruments

8.2. The factors to be considered in the determination of the number of instruments required are given in the following list:

(a) Characteristics of the facility, such as the extent of supervised areas and controlled areas, and the types of experiments carried out concurrently;

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

(c) Feedback from audits and experience of workers;

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

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

Technical specifications

8.3. Features of the technical performance of the instruments that should be considered include:

(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 angular directional response and radiation field size;

(j) Response of the instrument to radiation of other types;

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

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

(m) Reliability and maintainability.

Human factors

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

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

(b) Clear display of the unit;

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

(d) Robustness for typical field use;

(e) Risk of data loss or distortion while in transit to the plantwide data collection system.

Some of these aspects should also be considered in the selection of non- portable radiation monitoring instruments.

MAINTENANCE AND CALIBRATION OF INSTRUMENTS

8.5. Responsibility should be assigned for the maintenance and calibration of instruments, in accordance with established procedures. Instruments should be reliable and easy to maintain. For the purposes of users and for maintenance purposes, the number of different types of instruments should be minimized. The following factors may be considered:

(a) Instruments should have acceptable reliability;

(b) Ease of maintenance is desirable;

(c) Availability of spare parts and services should be assured;

(d) Instruments should be sealed against the ingress of contamination and should have surfaces that are easy to decontaminate;

(e) 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;

(f) The calibration process should be easy to perform.

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

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8.7. ~~Although instruments should be reasonably easy to calibrate, s~~Some instruments (e.g. noble gas monitors) will require special facilities to allow accurate calibration. The calibration of such instruments should be performed periodically, in accordance with the national regulations and standards.

9. ORGANIZATIONAL ASPECTS

INTRODUCTION

9.1. The requirements for the organizational aspects of research reactor facilities are established in Ref. [1], which covers all aspects of the safety of research reactors, including those relating to radiation protection. Aspects related to predisposal management of radioactive waste are covered in SSG-40 [176]. The organizational structure of the research reactor facility should be such as to ensure proper fulfilment of the objectives for radiation protection.

ORGANIZATIONAL STRUCTURE

9.2. The organizational structure depends on the size and the nature of the facility and on its management system, and generally includes groups for reactor operation, radiation protection and facility users, among others. A direct reporting line to the senior management should be in place for fulfilling the function of radiation protection.

9.3. The structure of the organization should be clearly defined, and the responsibilities of each person in the organizational structure should be assigned and documented. Job descriptions should clearly specify the responsibilities for radiation protection that are associated with each position.

RESPONSIBILITIES

Regulatory body

9.4. In accordance with the legal framework of the State, the regulatory body should assume responsibilities, functions and activities in areas relating, among others, to radiation protection, as established in Ref. [~~25~~3021]. 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 regulations should set out requirements for radiation protection, including prescribing occupational dose limits and public dose limits, requiring the optimization of protection, defining the qualification and training requirements for personnel, and performing audits.

Management of the operating organization

9.5. The operating organization is responsible for ensuring that exposures are controlled and kept within limits, that protection and safety are optimized, and that an appropriate programme for radiation protection is set up and operated [218]. For such a radiation protection programme to be effective, there should be a high level of commitment by the management. Such a commitment

should be demonstrated as described in the following.

Policy

9.6. A clear and concise written policy on protection and safety should be approved and adopted by the operating organization. This policy should include consideration of the following:

(a) Priority of safety over reactor operations and research;

(b) A commitment to the application of standards for radiation protection.

Special attention should be paid to the application of the optimization principle;

(c) Independence of the functions for radiation protection.

Provision of resources

9.7. The provision of resources for radiation protection is an essential part of the commitment by management. The resources should include adequate equipment and supplies, and the personnel necessary to maintain an effective radiation protection programme. This requires:

(a) Appropriate staffing;

(b) Programmes and facilities for training and retraining;

(c) Conducting a programme for personal dosimetry;

(d) Provision of personal protective equipment, instruments, etc.;

(e) Conducting a programme for the surveillance of levels of radiation and contamination at the facility and in the environment;

(f) Emergency ~~planning and~~ preparedness and response;

(g) Adoption of operating rules for safety, together with derived limits and operational limits;

(h) Appropriate documentation.

Safety culture

9.8. An organizational safety culture that encourages a questioning and learning attitude to safety and radiation protection, and that discourages complacency, is required to be fostered and maintained by management [~~13~~23~~12~~, ~~14~~24~~13~~]. The operating environment and the attitude of all staff should be conducive to good practices for radiation protection ~~[26]~~.

9.9. A strong safety culture ensures that:

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

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

(c) Organizational arrangements and lines of communication are in place that allow an appropriate

flow of information on safety and radiation protection to and between all relevant levels in the operating organization.

Reactor manager

9.10. The reactor manager should act in accordance with the policies of the operating organization and should ensure that there are adequate and appropriately trained staff to carry out all operational functions. The reactor manager should be responsible for safely managing the facility in accordance with the regulatory requirements. The reactor manager should ensure that requirements for radiation protection are met during the course of the work.

Safety committee

9.11. The safety committee is an important part of the operating organization of the research reactor. The safety committee provides, for the benefit of the reactor manager, an independent review of the safety related aspects associated with the operation of the reactor and its research facilities. With regard to radiation protection, this committee should be asked to review and advise on:

(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 follow-up actions put in place to prevent their recurrence.

Radiation protection personnel[7]

Radiation protection officer

9.12. The radiation protection officer is an individual who is technically competent in matters of radiation protection relevant to the research reactor facility and who is designated by the operating organization to oversee the application of the requirements for radiation protection. One of the radiation protection officer's main responsibilities is to prepare and conduct the radiation protection programme. In this context, the radiation protection officer's functions may include:

(a) Providing expert advice and assistance to the facility management in specifying and carrying out its responsibilities for the radiation protection programme;

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

[7] See Ref. [1] Para. 7.23~~, paras 7.98–7.100~~.

(c) Providing expert advice on and assistance in matters of radiation protection to facility staff;

(d) Assessing the effectiveness and efficiency of all aspects of the facility's operations relating to radiation protection;

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

(f) Guiding and overseeing the application of the optimization principle [203];

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

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

9.13. The radiation protection officer should be responsible for assessing the quality of radiation protection services.

Other radiation protection personnel

9.14. *Radiation protection technicians.* Radiation protection technicians perform most of the work, making measurements for the purposes of radiation protection; senior technicians may also make interpretations and recommendations on the basis of such measurements.

9.15. *Other personnel.* Some organizations may assign specific radiation protection functions to other staff, such as operators, maintenance staff or research workers. In such cases, these functions should be clearly specified, documented and audited by the radiation protection officer.

Management systems

9.16. A management system [131232, 142413] is required to be established to implement all processes that include aspects of radiation protection, in accordance with the radiation protection programme. The management system required for a particular research reactorfacility should be based on the number, complexity and safety significance of the activities.

9.17. The responsibility for meeting the objectives of the radiation protection programme rests with the radiation protection officer.

9.18. Individuals 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 may be necessary to achieve the goals, strategies and objectives of the organization.

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

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Qualification

9.20. The regulatory body should provide guidance on requirements for the qualification of ~~reactor~~ operating personnel and, where appropriate, should review and approve any proposals made by the operating organization (see Ref. [7~~24~~]).

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

Training

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 carry out various duties. 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 should make arrangements for the facility operating ~~staff~~ personnel ~~of the research reactor facility~~ to be adequately trained and confirmed to be as proficient in measures for radiation protection as necessary for the duties that they will be expected to undertake and for the responsibilities allocated to them.

9.24. The training for workers should cover all topics relevant to the task assignments and the potential hazards. Those workers who need to work in zones of high radiation levels should be trained in their specific work activities so as to enable them to perform their duties in the minimum possible period of time, in accordance with the principle of optimization. This could include, for example, training on mock-ups, rehearsing the planned work and practising emergency actions.

9.25. Training on emergency procedures should be given periodically to ensure that all persons who would respond in an emergency know which actions they are expected to take. All site personnel should take part in periodic exercises that simulate nuclear or radiological emergencies of various types. It may be advisable to include off-site personnel, such as fire fighters, medical staff and police, who would be required to come onto the site in the event of an emergency.

9.26. In view of the individual nature of most research reactor facilities, in addition to externally organized courses, in-house training programmes that are tailored to the facility's needs should be developed. When establishing a training programme, a task analysis should be performed for each relevant position. The successful completion of externally available training programmes may be useful and may be required by national legislation for specified positions.

9.27. The training programmes should include field practice or on the job training, 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.

Specific training for operating personnel

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

Specific training for users of research reactors

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

Specific training of temporary workers

9.30. The operating organization should ensure that any personnel of other organizations who are employed on the site (outside workers), particularly personnel of contractors, have received proper training to enable them to perform their work in accordance with the relevant requirements for radiation protection.

9.31. Special arrangements should be made so that temporary personnel whose place of work is not within the ~~research reactor~~ facility can become familiar with the relevant radiation protection measures relating to their tasks. In particular, those temporary workers categorized as occupationally exposed to ionizing radiation need training on the arrangements for radiation protection at the facility prior to commencing their jobs. The degree of this training should depend on the duration and nature of the work they are doing. In specific instances, a qualified person may be provided as a full time escort 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 that are in place. 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

9.33. Periodic retraining should be conducted to maintain facility~~reactor~~ operating personnel and radiation protection personnel at the appropriate professional level required for their designated positions. Retraining is generally required only for certain functions or positions. The operating

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organization is responsible for prescribing the type and frequency of refresher courses.

9.34. Retraining should be conducted as often as necessary, but particularly after changes in tasks or significant changes or modifications to equipment and/ or procedures at the facility. This ensures that the required level of competence is maintained and that the implications of the changes for radiation protection and radioactive waste management are understood. Records should be kept of the type of training that each person has undergone and when.

9.35. Training programmes should be updated at regular intervals. In updating the training programme, 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 facility and at other relevant facilities.

DOCUMENTATION

Records

9.36. The operating organization is responsible for keeping operating records and reports [1]. It is responsible for the collection, storage and retrieval of records and for record keeping to maintain current and historical information on important aspects of the radiation protection programme and the waste management programme. This information should be retained for use in meeting the objectives of these programmes and in preparing reports by the operating organization to the regulatory body.

Personal dose records

9.37. Personal dose records should be available for purposes of review and assessment:

> "~~Exposure r~~Records of occupational exposure for each worker shall be ~~preserved~~ maintained during and after the worker's working life, ~~and afterwards~~ 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 ~~termination~~ cessation of the work in which the worker was subject ~~involving occupational~~to occupational exposure." (Ref. [2], para. ~~I.49~~3.104)

All personal dose records should be carefully and periodically assessed by the radiation protection officer and should be examined for the following points:

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

(b) Consistency of doses with area survey data;

(c) Anomalous or unusual doses;

(d) Effectiveness of the revised procedures in reducing doses.

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Personal medical records

9.38. Medical examinations of occupationally exposed workers should follow the general principles of occupational medicine. There should be 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, those workers who have a medical condition that may necessitate taking particular precautions in their work should be identified.

9.39. Health surveillance records should be confidential and they should be preserved in a manner that has been agreed to by the regulatory body. The minimum period of record keeping should be the lifetime of the worker concerned.

Survey data records

9.40. Survey data include all results of the routine and special monitoring that is performed to assess radiological hazards in and around the facility. Records of these data should be kept in accordance with the requirements of the facility's management system and any regulatory requirements.

9.41. All installed radiation monitors, hand-held radiation meters and dosimetry equipment should be tested and calibrated in accordance with the facility's management system and the operational limits and conditions (OLCs). The results of these tests should be recorded so that the testing and repair history of each instrument can be retrieved. Records of tests and calibrations should be maintained.

10. RADIOLOGICAL ASPECTS OF EMERGENCIES

INTRODUCTION

10.1. An emergency plan including detailed procedures should be developed to cover all foreseeable aspects of emergencies at a research reactor facility, in accordance with the requirements established in GSR Part 7 [317][8]. This plan should specify the set of infrastructural _elements necessary to provide the capability to respond to an emergency. This plan and procedures should be tested, reviewed and updated as necessary. The conclusions of periodic emergency exercises should be considered in the updating process.

10.2. Owing to the low core inventories of nuclear material and radioactive material, the radiological aspects of the emergency plan for most research reactor facilities would be expected to concern mainly the on-site response. However, notification of an emergency to external authorities and the performance of some off-site surveys 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 detailed in Ref. [6317].

[8] Requirements are established in Ref. [67].

10.3. This Safety Guide deals only with the radiological aspects of on-site emergency preparedness and response. In the event of an emergency, the on- site response may include a wide range of activities, such as a preliminary assessment of hazards, rescue operations, the termination of accident propagation, mitigation of a release of radioactive material to the environment, on-site sheltering and evacuation, area monitoring and effluent monitoring, dosimetry and dose control, access control, issuing of information to the public and submission of notification and reports.

RADIOLOGICAL ASSESSMENT

10.4. Radiological assessment is the process of evaluation of radiological hazards. An initial assessment should be carried out as part of a safety analysis to identify possible events that could warrant urgent protective actions. Specific 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 in the light of any information that would yield more accurate results. Post-accident impact assessments should also be carried out.

Radiation and contamination surveys

10.5. The operating organization should be able to arrange for radiation monitoring teams (sometimes called emergency response teams) to be promptly deployed to gather data. These teams should be given training commensurate with their possible degree of involvement in an emergency at the facility and should be equipped with the necessary protective equipment, monitoring instruments and sampling equipment.

10.6. Area 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 what has occurred, to decide on 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 early on, not only to assess doses to personnel, but also to help in estimating dose rates and contamination levels in areas of the ~~research reactor~~ facility. Doses received by personnel involved in the response to the emergency should be measured, recorded, evaluated and entered in their personal dose records.

PROTECTIVE ACTIONS

10.8. Arrangements should be put in place to ensure safety for all persons on the site in the event of an emergency. To minimize the hazards to workers, appropriate response measures and protective measures, such as notification, accounting for people on the site, shielding, evacuation, relocation and the use of radionuclide blocking agents, should be taken as necessary.

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CORRECTIVE ACTIONS

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

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Annex I

EXAMPLES OF POSSIBLE SOURCES OF RADIATION IN A RESEARCH REACTOR

This Annex provides general examples of the different sources of radiation that there may be in a research reactor facility and is not intended to be comprehensive. It is expected that operating organizations would determine the sources of radiation at their own research reactors.

Reactor core and fuel

Fission products

I–1. 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, and after a cooling time of 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–2. In all, there are about 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

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Transuranic nuclides

I–3. 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–4. 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–5. During operation, the core becomes a source of intensive radiation:

(a) Neutrons: 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 rays: 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–6. The activity present under normal conditions in coolant–moderator systems is due to several sources:

(a) Activation of the moderator–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 the structural components entering 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);

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

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

I–7. 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

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Air

I–8. 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, ^{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 may be released from the core cover gas systems or through leakages from experimental devices. Air may 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 may also arise from irradiations and from various experiments.

Beam tubes and samples

I–9. Research reactors usually have built-in facilities for a variety of experiments, irradiations or isotope production, which present a radiological hazard. These include beam tubes enabling neutron and gamma radiations 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–10. Sample materials as well as their containers and other components of the irradiation set-ups may become highly radioactive and present a radiation and contamination hazard.

I–11. Prompt gamma radiation due to the activation of samples situated in a neutron beam outside the shielding may pose a significant radiation and contamination hazard.

Reactor components

I–12. 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–13. Some other components, such as filters placed in the ventilation system or in gas purging systems, or filters and ion exchanger resins for water purification, collect radioactive impurities that may become significant sources of radiation. Deposits of radioactive crud and sludge along pipes can also contaminate reactor components and may constitute a radiation and contamination hazard even outside the core region.

Radioactive waste

I–14. 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 must be

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properly collected and treated. Aspects of the handling of radioactive waste are discussed in Section 6.

Spent fuel

I–15. Spent fuel presents a radiological hazard and special considerations are needed with regard to its handling, storage, transport and subsequent disposal. This radiological hazard is due to direct radiation and to releases in the event of cladding failure.

Special radioactive sources

I–16. Various types of radiation sources may be utilized in a research reactor facility. These may include:

(a) A neutron source for the startup of the reactor which may be stored safely 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–17. 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 major amount of the radioactive waste originates from chemical processing of the irradiated targets, carried out in a dedicated facility. However, some amount of solid waste can be generated owing to 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 reactor facility. Generally, processing of irradiated samples is done outside the reactor facility in specially designed laboratories.

I–18. Varieties 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 the dedicated controlled area for further analysis and examination. Such irradiated samples are also sources of radiation in research reactors. 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.

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Annex II

PERSONAL PROTECTIVE EQUIPMENT

INTRODUCTION

II–1. Personal protective equipment comprises clothing and other special equipment that is issued to individual workers to provide protection against exposure or potential exposure to radiation. It is used to protect each worker against the prevailing risk of external exposure or internal exposure in circumstances in which it is not reasonably practicable to provide complete protection by means of physical control measures and 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 provides the intended degree of protection effectively.

USE OF PERSONAL PROTECTIVE EQUIPMENT

II–2. As an administrative measure to restrict exposure, or as a last line of defence where neither physical control measures nor other measures are reasonably practicable, workers must use personal protective equipment. The use of personal protective equipment may be the only 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 each exposed worker. 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, reliable and planned degree of personal 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 external irradiation. However, protective clothing and protective equipment are more frequently used to prevent radioactive substances making direct contact with the body or entering the body and causing internal exposures.

II–4. Respiratory protective equipment is intended to prevent the inhalation of radioactive substances, which would result in radiation doses to the lungs and to other organs into which these substance(s) might ultimately pass or which might be irradiated by them.

SELECTION OF PERSONAL PROTECTIVE EQUIPMENT

II–5. Three items of information are necessary before selecting personal protective equipment:

(a) *The nature of the exposure*. Both qualitative and quantitative information is needed about conditions in the workplace. Surveys need to be performed to determine the radionuclide(s) present, the type of potential exposure(s) and the magnitude of possible doses, the physical form of the radiation source(s), and the nature and concentration(s) of any contamination. The radiological risks need to be considered together with other hazards to appreciate the

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difficulties of accomplishing the work wearing personal protective equipment.

(b) *Performance data of personal protective equipment*. Data are needed to assess the ability of available and/or approved personal protective equipment to reduce the particular exposure(s). This information will usually be available from the manufacturers, who will have carried out tests under controlled conditions as specified in international or national regulations and standards.

(c) *The acceptable level of exposure*. Personal protective equipment is used to minimize or even to eliminate exposure. In practice, a decision will be made, preferably by a qualified expert, on whether the personal protective equipment could in theory 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 ~~limited~~ a range~~s~~ of size~~s~~ ~~for workers of average build, often for men only~~. 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 workers' training must emphasize the need to fit and use the personal protective equipment correctly each time.

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

PROTECTIVE CLOTHING

II–9. Protective clothing is used as a barrier 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. Some protective clothing — particularly plastic clothing — is generally associated with the use of respiratory protection. The following are examples of typical protective clothing:

(a) *Flexible aprons* with a thickness up to the equivalent of 1/3 mm of lead (written as 0.33 mmPb) are available to shield the upper torso. Double sided aprons shield the chest and the back against radiation scattered behind the body. The aprons attenuate, by about 90%, low energy radiation such as scattered X rays (of tens of KeV kiloelectronvolts).

(b) *Shielding gloves* and sleeves containing up to 0.33 mmPb are manufactured. Like aprons, they are ineffective shields against most types of radiation other than electrons (beta particles) and low energy scattered X rays.

(c) *Laboratory coats* (lab coats) made of cotton or synthetic fibres are commonly used in research laboratories, where there is a risk of minor radioactive contamination. Laboratory coats for dry conditions are worn with appropriate gloves and shoe covers and, in some cases, a cap or hood to protect the hair and head.

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(d) *One piece suits*, *coveralls*, *overalls* or 'slicker suits' are used at industrial workplaces to protect those parts of the body covered by such clothing from radioactive contamination.

(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 and so on, or from a mixture of materials.

(f) *Protective footwear* includes overshoes, 'booties', shoes and boots.

Overshoes allow personal footwear to be worn in areas where there is a risk of a minor spill or drips contaminating the floor. 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 possibly less effective. These do not fully cover the personal footwear and may not provide a tight fit over it, especially over heels. Fabric overshoes with hard soles, and booties and fabric overshoes with leggings supported at the knee by elastic or drawstrings, provide further inexpensive options.

(g) Pressurized plastic clothing, worn with a full hood, rubber gloves and shoe rubbers, 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 conditions require,

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 requires that laundry facilities be available. It is important in this regard that highly contaminated clothing is segregated from clothing that has little more than trace levels of contamination. Procedures for clothing have to address this requirement through appropriate ~~contamination reference~~ administrative control levels and monitoring requirements. ~~Reference~~ Administrative control levels and monitoring requirements need to be available for the reissue of clothing that has been laundered. The laundry process may increase the volume of low level liquid radioactive waste that must 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 material that must be processed. It is appropriate, however, to use disposable clothing when laundering becomes impracticable owing to high levels of contamination.

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TYPES OF RESPIRATORY PROTECTIVE EQUIPMENT

II–12. There are two categories of respiratory protective equipment, with several subdivisions each:

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

(1) Filtering face piece respirators;

(2) Half mask respirators;

(3) Full face mask respirators;

(4) 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.

Breathing equipment provides clean air or oxygen from an independent uncontaminated source. The most common types are:

(1) Fresh air hose equipment;

(2) Constant flow compressed air equipment;

(3) 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 may have an assigned protection factor defined by national standards and referred to in national regulations. In a typical system, respiratory protective equipment is performance tested to determine the inward leakage as the ratio of the concentration of the test particles inside the respiratory protective equipment (or personal protective equipment) to the challenge concentration of test particles in the test chamber. This is expressed as a percentage, the challenge concentration corresponding to 100%. The manufacturer may quote the inverse (100:inward leakage), called 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 (or personal protective equipment).

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

(a) The integrity of the face seal;

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

Changing either or both of these factors may have a significant effect on the degree of protection actually achieved.

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SELECTION OF RESPIRATORY PROTECTIVE EQUIPMENT

II–15. Several types of respiratory protective equipment may have the necessary ratio of assigned protection factor to nominal protection factor when predictions and/or measurements have been made of the physical form and concentration of contamination in the workplace. The choice could include all types of respiratory protective equipment to protect against low concentrations of a particulate contaminant. Radioactive vapours and gases would restrict the choice to certain types of respirator or breathing equipment or, for adequate protection against contaminants at high concentrations in the ambient atmosphere, breathing equipment may be the only possibility. 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 must be carefully controlled. There must be assurance that users are medically fit to use the equipment and that they can and will wear the equipment as intended. This may include facial fit tests. The equipment must be capable of providing the degree of protection (protection factor) that is needed for the radiological conditions and must be compatible with the work that is to be performed.

RESPIRATORY EQUIPMENT

II–17. The respiratory protective equipment that is authorized for use varies somewhat between States; however, the following are typical of the items currently in use, listed in increasing order of protection:

(1) *Filtering face piece respirators* are made wholly or substantially of filter material that covers the nose and mouth. The face piece is held in place by straps and a nose clip, which helps to complete the seal. Air is drawn through the material by underpressure when the wearer inhales. Some models incorporate an exhalation valve. Filtering face piece respirators are mainly used for protection against low to moderately hazardous particles. Some models are capable of filtering malodorous (but not toxic) gases and vapours. They must not be confused with nuisance dust masks which filter only larger, low hazard dust particles.

(2) *The elastomer half mask or oronasal respirator* is a face piece of rubber or plastic moulded to cover the nose and mouth and 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* of rubber or plastic cover the entire face from just below the hairline to beneath the chin 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 approved impregnated activated charcoal canisters may be used for conditions involving

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airborne radioactive iodine. The charcoal canisters must have adequate removal efficiencies for all chemical forms of radioactive iodine. Full face respirators provide less protection than air supplied equipment.

(4) *Powered air-purifying respirators* provide a continuous flow of air into the mask to minimize inward leakage of contaminated air around an incomplete 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. It must be noted that respiratory equipment will not provide protection against absorption through the uncovered skin if tritiated water vapour is present.

(5) *Constant flow compressed air equipment*. A compressed air line may be used to supply a face mask, a hood or a blouse. The breathing air system must be capable of providing an adequate flow of air and must be conditioned for temperature and humidity. Compressors are required to be oil-free types. To ensure that breathing air and instrument air are not interchanged, different, non-compatible fittings may be used on each system. When portable compressors are used, the air intake must be from a contamination-free area. Self-contained breathing apparatus consists of a full face mask supplied with air or oxygen from compressed gas cylinders carried by the worker. Air is supplied to the mask through a positive pressure demand valve.

(6) *A ventilated pressurized suit* enclosing the whole body (including arms and legs) may consist of one or two parts. Halved suits are sealed together at the waist. Full suits may have a gastight zipper. The hood has at least the front section 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, in accordance with the design. Exhaust gases are discharged through exhaust valves in the suit body. Part of the air supply may cool the suit.

II–18. Decontamination facilities are needed for cleaning respiratory protective equipment, and the reissue for use must be closely controlled through monitoring levels and ~~reference~~ administrative control levels. Respiratory equipment must be well maintained, periodically inspected and tested.

OTHER PERSONAL PROTECTIVE EQUIPMENT

II–19. Several types of personal protective equipment may be necessary to work safely. To protect against physical injury, head protection, eye protection and toe protection may be necessary. A safety helmet ('bump cap') may be worn when wearing enclosed suits or hoods. Safety goggles may

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be worn inside ventilated suits. It is an advantage if the one item of personal protective equipment used incorporates all necessary protection, such as the ventilated helmet; if a respirator has eyepieces made from polycarbonate; or if integral boots have protective toecaps. Use of an eye shield with a respirator will severely limit the already restricted vision. Welding in a radioactive environment 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–20. Suits made from aluminized fire resistant materials are available to protect against extreme radiant heat, and in hot environments, a cooled suit might be used. Suits resistant to attack by specific chemicals must be assessed before use with regard to their contamination control.

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Annex III

DESIGN CONSIDERATIONS FOR COLLECTION OR HOLDING (DELAY) TANKS FOR LIQUID RADIOACTIVE WASTE

III–1. At facilities where significant volumes of radioactive liquid waste are generated, collection or holding tanks may be installed. In general, collection or holding tanks are constructed in sets of two or more, so that one may be filling while the contents of a full one are being sampled, analysed or discharged. The tanks must meet the following general requirements:

(a) The tank is 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 clearing if this becomes necessary.

Secondary containment is provided around the tanks to prevent the spread of contamination in the event of leakage. Shielding around the tanks is also provided if necessary.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Shokr, A.M. International Atomic Energy Agency

Hardesty, D. Nuclear Regulatory Commission, United States of America

D'Arcy, A Consultant, South Africa

Sears, D. International Atomic Energy Agency

Hargitai, T. Consultant, Hungary

Abou Yehia, H. Consultant, France

Kennedy, W International Atomic Energy Agency

Guskob, Andrey International Atomic Energy Agency

Bennett, David G. International Atomic Energy Agency

Kemp, Dunca International Atomic Energy Agency

DeLorenzo, Nestor INVAP, Argentina

Helvenston, Edward Nuclear Regulatory Commission, United States of America

~~Abou Yehia, H. International Atomic Energy Agency~~

~~Ciuculescu, C. International Atomic Energy Agency~~
~~Damette, G. Institut de radioprotection et de sûreté nucléaire, France~~
~~De Lorenzo, N. INVAP, Argentina~~
~~Dodd, B. BD Consulting, United States of America~~
~~Hargitai, T. International Atomic Energy Agency Keverling Buisman, A. Energy~~

~~Research Centre, Netherlands Koblinger, L. Hungarian Atomic Energy Agency, Hungary~~

~~Litai, D. International Atomic Energy Agency~~
~~Metcalf, P.E. International Atomic Energy Agency~~
~~Mück, K. Austrian Research Centres, Austria~~
~~Sági, L. KFKI Atomic Energy Research Institute, Hungary~~
~~Sarkar, S. Australian Radiation Protection and Nuclear Safety Agency, Australia~~

~~Shokr, A.M. International Atomic Energy Agency~~
~~Vincze, P. International Atomic Energy Agency~~
~~Willers, A. International Atomic Energy Agency~~
~~Woods, D. Australian Nuclear Science and Technology Organisation, Australi~~

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