

IAEA Safety Standards

for protecting people and the environment

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

Specific Safety Guide

No. SSG-85



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IAEA SAFETY STANDARDS

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

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

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

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RADIATION PROTECTION AND
RADIOACTIVE WASTE
MANAGEMENT IN
THE DESIGN AND OPERATION OF
RESEARCH REACTORS



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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA SAFETY STANDARDS SERIES No. SSG-85

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SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2023

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FOREWORD

by Rafael Mariano Grossi
Director General

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

Safety measures and security measures¹ have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

Safety Fundamentals

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

Safety Requirements

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

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

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

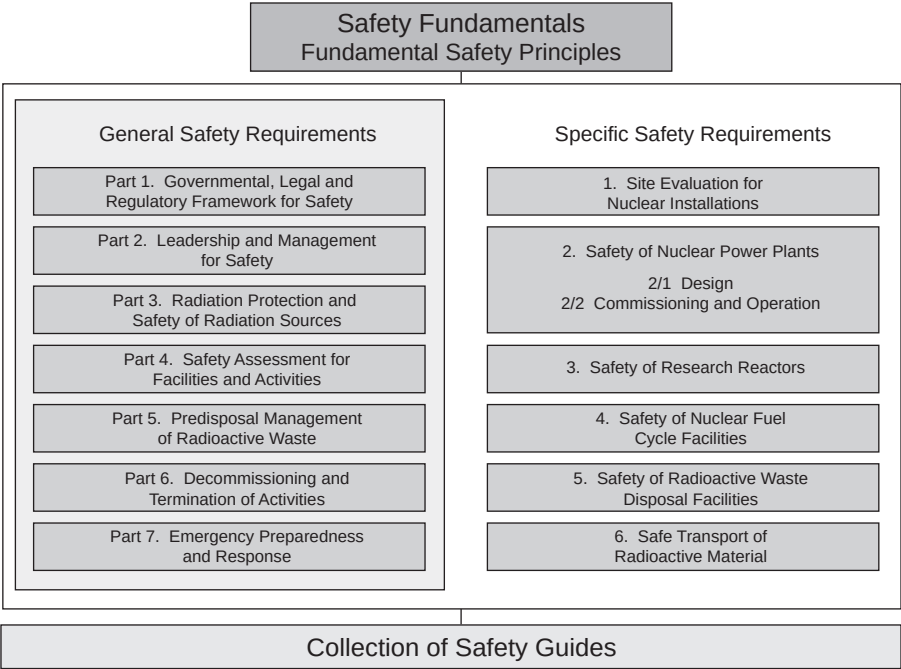


FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

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

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be

used by States as a reference for their national regulations in respect of facilities and activities.

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

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

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

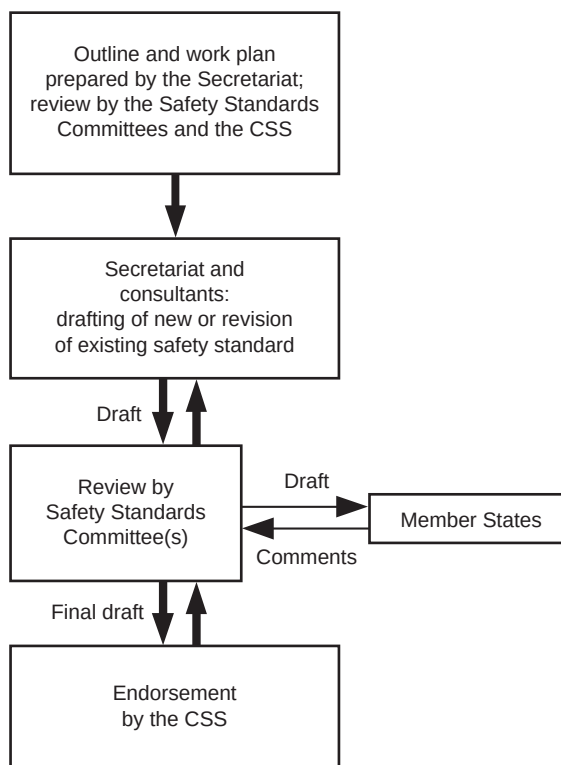


FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

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

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

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

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

CONTENTS

1.	INTRODUCTION.	1
	Background (1.1–1.6).....	1
	Objective (1.7, 1.8).....	2
	Scope (1.9–1.13).....	2
	Structure (1.14, 1.15)	3
2.	REQUIREMENTS, OBJECTIVES AND CONCEPTS FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT FOR A RESEARCH REACTOR	4
	Fundamental safety objective (2.1–2.3)	4
	Dose limits, dose constraints and authorized limits on discharges for a research reactor (2.4–2.7).....	5
	Application of the principle of optimization (2.8–2.10).....	6
	Design target doses for operational states for a research reactor (2.11, 2.12).....	7
	Design target doses for accident conditions for a research reactor (2.13, 2.14)	8
3.	RESEARCH REACTORS AND ASSOCIATED RADIOLOGICAL HAZARDS (3.1, 3.2)	8
	Radiological hazards associated with different reactor types and utilization (3.3–3.14).....	9
	Radiation sources at a research reactor (3.15–3.17).....	11
4.	RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT IN THE DESIGN OF A RESEARCH REACTOR	12
	Approach to design for radiation protection and radioactive waste management (4.1–4.10).....	12
	Design strategy for radiation protection at a research reactor (4.11–4.21).....	15
	Design features for radioactive waste management at a research reactor (4.22–4.32)	19

Lifetime management and decommissioning of a research reactor (4.33, 4.34)	23
Design approaches for dealing with accident conditions at a research reactor (4.35–4.40)	23
Design of structures, systems and components for radiation protection and radioactive waste management at a research reactor (4.41–4.49)	25
Design features to facilitate decontamination at a research reactor (4.50–4.53)	26
Design of shielding at a research reactor (4.54–4.62)	27
Design of ventilation systems at a research reactor (4.63–4.67)	28
 5. RADIATION PROTECTION IN THE OPERATION OF A RESEARCH REACTOR (5.1–5.5)	 29
Dose limits (5.6)	30
Optimization principle and dose constraints (5.7–5.11)	31
Administrative control levels (5.12–5.14)	32
On-site radiation protection and dose control at a research reactor (5.15–5.57)	32
Optimization of protection of the public (5.58–5.61)	43
 6. RADIOACTIVE WASTE MANAGEMENT IN THE OPERATION OF A RESEARCH REACTOR (6.1–6.4)	 44
Management of the generation of radioactive waste at a research reactor (6.5–6.10)	45
Optimization of protection in the management of radioactive waste (6.11)	48
Packaging and confinement of radioactive waste at a research reactor (6.12–6.14)	49
Storage of radioactive waste at a research reactor (6.15–6.23)	50
Documentation relating to the management of radioactive waste at a research reactor (6.24–6.28)	52
Classification, characterization and segregation of radioactive waste at a research reactor (6.29–6.37)	53
Processing of radioactive waste at a research reactor (6.38–6.54)	55
Transport of radioactive waste (6.55, 6.56)	58
Control of radioactive discharges and compliance monitoring at a research reactor (6.57–6.63)	59

7.	RADIATION MONITORING PROGRAMME AT A RESEARCH REACTOR (7.1–7.3)	60
	Types of monitoring survey at a research reactor (7.4–7.10)	61
	Workplace monitoring at a research reactor (7.11–7.19)	63
	Individual monitoring at a research reactor (7.20–7.37)	65
	Effluent monitoring at a research reactor (7.38–7.47)	68
	Environmental monitoring at a research reactor (7.48–7.50)	70
8.	INSTRUMENTATION FOR THE RADIATION PROTECTION PROGRAMME AT A RESEARCH REACTOR (8.1)	71
	Radiation monitoring instruments for use at a research reactor (8.2–8.4)	72
	Maintenance and calibration of instruments at a research reactor (8.5–8.7)	73
9.	ORGANIZATIONAL ASPECTS OF RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR (9.1)	74
	Application of the management system to radiation protection and radioactive waste management at a research reactor (9.2–9.5)	74
	Organizational structure for radiation protection and radioactive waste management at a research reactor (9.6, 9.7)	75
	Responsibilities relating to radiation protection and radioactive waste management at a research reactor (9.8–9.19)	75
	Qualification, training and retraining of personnel for radiation protection and radioactive waste management at a research reactor (9.20–9.36)	79
	Documentation for radiation protection and radioactive waste management at a research reactor (9.37–9.42)	82
10.	RADIATION PROTECTION DURING AN EMERGENCY AT A RESEARCH REACTOR	84
	Introduction (10.1–10.3)	84
	Radiological assessment (10.4–10.7)	85
	On-site protective actions for a nuclear or radiological emergency at a research reactor (10.8)	86

Corrective actions for a nuclear or radiological emergency at a research reactor (10.9).....	86
REFERENCES.....	87
ANNEX I: EXAMPLES OF SOURCES OF RADIATION IN A RESEARCH REACTOR.....	91
ANNEX II: PERSONAL PROTECTIVE EQUIPMENT AT A RESEARCH REACTOR.....	97
ANNEX III: DESIGN CONSIDERATIONS FOR COLLECTION OR HOLDING (DELAY) TANKS FOR LIQUID RADIOACTIVE WASTE AT A RESEARCH REACTOR.....	104
CONTRIBUTORS TO DRAFTING AND REVIEW.....	105

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 for radiation protection are established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [2]. Requirements for radioactive waste management are established in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [3].

1.2. This Safety Guide provides recommendations on 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:

- (a) IAEA Safety Standards Series No. SSG-80, Commissioning of Research Reactors [4];
- (b) IAEA Safety Standards Series No. SSG-81, Maintenance, Periodic Testing and Inspection of Research Reactors [5];
- (c) IAEA Safety Standards Series No. SSG-82, Core Management and Fuel Handling for Research Reactors [6];
- (d) IAEA Safety Standards Series No. SSG-83, Operational Limits and Conditions and Operating Procedures for Research Reactors [7];
- (e) IAEA Safety Standards Series No. SSG-84, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [8];
- (f) IAEA Safety Standards Series No. SSG-10 (Rev. 1), Ageing Management for Research Reactors [9];
- (g) IAEA Safety Standards Series No. SSG-37 (Rev. 1), Instrumentation and Control Systems and Software Important to Safety for Research Reactors [10].

1.4. Additional recommendations on the safety of research reactors are provided in IAEA Safety Standards Series Nos SSG-20 (Rev. 1), Safety Assessment for Research Reactors and Preparation of the Safety Analysis

Report [11], and SSG-24 (Rev. 1), Safety in the Utilization and Modification of Research Reactors [12].

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

1.6. This Safety Guide supersedes IAEA Safety Standards Series No. NS-G-4.6, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors¹.

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 reactors to meet the relevant requirements of SSR-3 [1], in particular Requirements 8, 15, 34, 57, 59, 84 and 85.

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

SCOPE

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

1.10. Some research reactors, critical assemblies and subcritical assemblies with a low hazard potential might need less comprehensive radiation protection and

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

radioactive waste management programmes. While all recommendations in this Safety Guide are to be considered, some might not be applicable to such research reactors, critical assemblies and subcritical assemblies (see Requirement 12 and paras 2.15–2.17 of SSR-3 [1], as well as IAEA Safety Standards Series No. SSG-22 (Rev. 1), Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [15]).

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

1.12. Owing to their specialized nature, certain topics are not dealt with in detail in this Safety Guide, such as environmental monitoring, disposal of radioactive waste, management of spent fuel and off-site emergency response. These topics are covered in other IAEA Safety Standards Series publications (Refs [3, 16–20]); this Safety Guide should be used in conjunction with these publications.

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

STRUCTURE

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

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

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

FUNDAMENTAL SAFETY OBJECTIVE

2.1. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [21], states that “**The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.**” To meet this objective for a research reactor, a comprehensive safety assessment is required to be performed (see Requirement 5 of SSR-3 [1]) to identify all sources of radiation and to evaluate the radiation doses that could be received by workers and the public, as well as potential effects on the environment. This safety assessment should examine the following:

- (a) All planned normal modes of operation of the research reactor;
- (b) The performance of the research reactor during and following anticipated operational occurrences;
- (c) The effects of design basis accidents;
- (d) Event sequences that might lead to design extension conditions.

2.2. The safety assessment is used to demonstrate compliance with national legislation and requirements issued by the regulatory body (see para. 3.8 of SSR-3 [1]), the robustness of the engineering design of the research reactor in withstanding postulated initiating events and accidents, and the effectiveness of the safety systems and of safety related items. The safety assessment should also be used as a basis for determining the necessary arrangements for emergency preparedness and response.

2.3. Paragraph 2.1 of SF-1 [21] states:

“[M]easures have to be taken:

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

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

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

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

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

2.6. A dose constraint should be set for workers who do not enter supervised areas or controlled areas, to ensure that they receive the same level of protection as members of the public (see para. 3.78 of GSR Part 3 [2]). With regard to dose constraints for public exposure, para. 3.36 of IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [22], states that “The dose constraint for a particular source is intended to ensure that the

sum of the doses from planned operations for all sources that may contribute to the exposure of the representative person remains within the dose limit.” Studies should be performed prior to the operation of a research reactor to identify the representative person and the main pathways of exposure for such a person.

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

APPLICATION OF THE PRINCIPLE OF OPTIMIZATION

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

- (a) The radiation exposure of workers and the public should be reduced by the implementation of radiation protection measures such that further expenditure for radiation protection measures in design, construction and operation would not be warranted by the associated reduction in radiation exposure.
- (b) The design of a research reactor should aim to avoid any major disparities in the individual doses received by different types of worker.

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

2.10. The fundamental role of optimization in the design of a research reactor and its components is to provide a basis for determining which engineered provisions for controlling radiation exposures are reasonably practicable. This is frequently a matter of judgement, based on past experience. In most cases, the principle of optimization should be applied to achieve a balance between the need for

dose reduction, the need to ensure safe and reliable research reactor operation, and the costs involved. A qualitative approach based on expert judgement and on the utilization of the best available and proven technology may be sufficient to make decisions on the optimum level of protection that can be achieved. At the design stage of a research reactor or for a major modification or at the decommissioning stage, where significant expenditure is involved, the use of a more structured approach may be appropriate and techniques that aid decision making may be useful.

DESIGN TARGET DOSES FOR OPERATIONAL STATES FOR A RESEARCH REACTOR

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

2.12. To focus the design effort on those aspects that contribute most to the collective dose and individual doses to workers, it is useful to set design target doses in terms of the collective dose to the groups of workers who are expected to receive the highest doses, such as maintenance workers or radiation protection personnel. It is also useful to set design target doses in terms of the collective dose for each category of work, such as the maintenance of major components, in-service inspections, refuelling and radioactive waste management. The design target dose should be verified during hot commissioning (see SSG-80 [4]) of the research reactor.

DESIGN TARGET DOSES FOR ACCIDENT CONDITIONS FOR A RESEARCH REACTOR

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

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

3. RESEARCH REACTORS AND ASSOCIATED RADIOLOGICAL HAZARDS

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

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

RADIOLOGICAL HAZARDS ASSOCIATED WITH DIFFERENT REACTOR TYPES AND UTILIZATION

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

- (a) The power level of the research reactor and its radionuclide inventory;
- (b) The type of research reactor;
- (c) The transient (power excursion) characteristics of the research reactor;
- (d) The irradiation facilities and experimental devices in and around the reactor core, in particular beam tubes and any pneumatic ‘rabbit’ systems used to transfer irradiation samples to and from the core.

Power level of the research reactor and radionuclide inventory

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

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

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

Type of research reactor

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

Transient (power excursion) characteristics of the research reactor

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

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

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

Irradiation facilities and experimental devices

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

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

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

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

RADIATION SOURCES AT A RESEARCH REACTOR

3.15. In the radiation protection programme for any research reactor, account should be taken of all sealed radioactive sources and unsealed radioactive sources that may be present in the facility. A full listing of such sources and their forms, locations and levels of activity during normal operation, as well as during anticipated operational occurrences, provides the basis for shielding calculations, zoning (area designation), design of ventilation systems, surveillance planning, dose assessment, radioactive waste management and determination of effluent discharges [10]. Spent fuel presents a radiological hazard from direct radiation and from the release of radioactive material in the event of cladding failures. Consideration should be given, therefore, to the handling, storage, transport and subsequent disposal of spent fuel. Specific recommendations are provided in IAEA Safety Standards Series No. SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel [18].

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

- (a) Fuel in the reactor core;
- (b) Spent fuel and core components stored in the reactor building;
- (c) Transuranic nuclides, fission products, and activation and corrosion products in the spent fuel pool or the coolant system;
- (d) Equipment, systems and piping containing activation products;
- (e) Solid and liquid radioactive waste and material arising from the treatment of radioactive waste;
- (f) Gaseous radioactive material;

- (g) Experimental facilities with the potential to generate activated material or other radioactive material;
- (h) Tools and devices for the storage and handling of radioactive material, including sample activation and/or irradiation facilities, in-core experiments, beam ports and hot cells;
- (i) Material that has been irradiated in the reactor;
- (j) Neutron sources and calibration sources;
- (k) Neutron detectors (particularly fission chambers and self-powered neutron detectors);
- (l) Components of purification systems, such as filters and ion exchange columns.

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

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

APPROACH TO DESIGN FOR RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT

Human resources

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

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

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

- (a) Their expertise in the production, handling and transport of radioactive material at a research reactor and the release of radioactive material to the environment;
- (b) Their ability to provide safety related inputs, such as evaluation of sources of radiation and the associated radiation exposures, using available software and data from relevant operating experience;
- (c) Their familiarity with regulatory requirements and guidance and with best practices.

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

Safety culture

4.5. Requirements for fostering a strong safety culture are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [23], and associated recommendations are provided in IAEA Safety Standards Series Nos GS-G-3.5, The Management System for Nuclear Installations [24], and GSG-16, Leadership, Management and Culture for Safety in Radioactive Waste Management [25].

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

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

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

- (b) Knowledge of the practices that result in exposure of site personnel and of members of the public;
- (c) A comprehensive training programme;
- (d) Management of the relationship between operation, maintenance and design;
- (e) Familiarity with the main factors, including exposure pathways, that influence individual doses and collective doses;
- (f) Familiarity with available software to assist in optimization of the design;
- (g) Recognition that specialists in radiation protection should be consulted whenever necessary to ensure that aspects of the design that affect radiation protection are properly evaluated and taken into account.

Organizational aspects

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

4.9. The whole design process is required to be performed under an appropriate management system, established in accordance with Requirements 6–8 of GSR Part 2 [23]. The operating organization should establish a process as part of the management system to ensure that the designers implement the necessary measures for radiation protection and waste safety at all stages of the design process.

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

- (a) Radiation protection specialists and waste safety specialists are consulted at the early stages of the design when options for the major aspects of the design are being evaluated.
- (b) The design incorporates proven engineering practices that operating experience has shown to be effective in optimizing protection, and 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 are reviewed by radiation protection specialists and waste safety specialists, respectively.

- (d) There is an appropriate review mechanism to resolve any design or operational issues that might be in conflict with requirements for radiation protection and radioactive waste management.
- (e) The design incorporates measures to ensure that the activation of structures, systems and components (SSCs) will not prevent their removal or replacement over the lifetime of the research reactor.

DESIGN STRATEGY FOR RADIATION PROTECTION AT A RESEARCH REACTOR

4.11. The design target doses set at the start of the design process, as described in paras 2.11–2.14, should include the following:

- (a) Design target doses in terms of the annual collective doses and annual individual doses to site personnel;
- (b) Design target doses in terms of the annual individual dose to members of the public.

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

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

Design for radiation protection of site personnel

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

- (a) A strategy for controlling the exposure of site personnel should be developed so that important aspects are considered early in the design. In most types of research reactor, design provisions in two areas may have

major impacts in reducing exposures. The first area is preventive and corrective maintenance. Appropriate design provisions should be made for inspection and maintenance. Hence, a proven design with high reliability and less need for maintenance, especially corrective maintenance, would be the preferred option. The second area is design features that minimize the production and buildup of radionuclides, which will reduce radiation levels and contamination levels throughout the facility. In contrast, design features that increase shielding or ventilation might offer less benefit.

- (b) General design criteria for radiation protection should be developed and documented. These criteria include the principles on which the layout of the facility will be based, taking into account the interface between safety and nuclear security, provisions for accident conditions and restrictions on the use of particular materials in the design of the reactor.
- (c) A logical layout for the research reactor should be developed, and zones should be defined on the basis of predicted dose rates and contamination levels, access requirements and other design requirements, such as the need to separate components that perform safety functions (see Requirements 26 and 27 of SSR-3 [1]). The predicted dose rates may be calculated by using source terms that provide the basis for the radiation protection aspects of the design, or they may be determined on the basis of operating experience at similar research reactors (provided that there are no significant differences in the relevant parameters for design and operation). The designation of zones within the research reactor needs to be consistent with regulatory requirements and with Requirement 24 of GSR Part 3 [2].
- (d) The maintenance programme and operations programme should be specified, preferably on the basis of well established concepts for optimizing exposure. The collective doses and individual doses associated with these programmes should be evaluated. Use should be made of relevant operating experience, where available, particularly for work that is difficult to predict precisely, such as corrective maintenance. In such cases, the use of mock-ups would assist in the reduction of uncertainties.
- (e) At each stage of the design, the doses that are evaluated should be compared with the design target doses and dose constraints. When there are options for the design, optimization studies should be performed. These studies are particularly important in cases where it is predicted that the design target doses will be exceeded.

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

- (a) Design features that enable in-service inspection to facilitate the replacement of ageing SSCs. Structures and components that are relatively inaccessible should be identified, and in each case their design life should be specified and clearly documented.
- (b) Measures for the prevention of corrosion or other deterioration of the containment or means of confinement and of other items important to safety, including the use of protective coatings and/or cathodic protection.
- (c) Provisions for the installation and removal of irradiated sample coupons, where possible, to facilitate periodic inspection.

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

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

4.17. A review of design features for the purpose of facilitating decommissioning is required to be performed at the design stage for the research reactor (see Requirement 33 of SSR-3 [1]). In general, design features that are conducive to maintenance and inspection during the operating lifetime of the reactor will also facilitate decommissioning. In accordance with para. 6.92 of SSR-3 [1] and the requirements established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [26], to facilitate decommissioning, the following are required to be considered:

- (a) Careful selection of materials to minimize the following:
 - (i) Activation of materials;
 - (ii) Generation of radioactive waste;
 - (iii) Spread of activated corrosion products;
 - (iv) The effort needed for the decontamination of surfaces;

- (v) Use of potentially hazardous substances (e.g. oils, flammable materials, chemically hazardous materials, fibrous insulations).
- (b) Optimization of the design, layout and access routes of the facility to facilitate the following:
 - (i) 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) Storage of waste;
 - (vi) Control of radioactive material within the installation.
- (c) The predisposal management of radioactive waste (i.e. pretreatment, treatment, conditioning and storage of waste arising from operation and decommissioning of the reactor).

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

- (a) Reduction of dose rates in working areas by implementing the following:
 - (i) Source reduction (e.g. by material selection, decontamination, control of corrosion, control of water chemistry, filtration and purification);
 - (ii) Improvement of shielding;
 - (iii) Increasing the distance between workers and sources (e.g. through provisions for remote handling);
 - (iv) Ensuring good ventilation.
- (b) Reduction of occupancy times in areas with significant dose rates by implementing the following:
 - (i) Specifying high reliability equipment to ensure very low failure rates;
 - (ii) Ensuring ease of maintenance, ease of removal or remote operation of equipment;
 - (iii) Simplifying operating procedures;
 - (iv) Ensuring ease of access and good lighting.

Design for radiation protection of members of the public

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

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

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

4.21. The measures that are included in the design to provide an optimized level of radiation protection in operational states are usually adequate for radiation protection in the commissioning stage. Special care should be taken, however, during power and overpower tests. During these tests, comprehensive measurements, tests and verifications for radiation protection purposes should be performed to ensure the adequacy of the design (see SSG-24 (Rev. 1) [12]).

DESIGN FEATURES FOR RADIOACTIVE WASTE MANAGEMENT AT A RESEARCH REACTOR

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

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

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

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

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

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

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

- (a) Selection of materials that are less prone to activation (e.g. use of plastic for the pneumatic ‘rabbit’ system irradiation target carriers) or of materials that generate activation products that decay quickly (e.g. use of aluminium components near the core);
- (b) Making allowance for the thermal expansion and contraction of pool water in a manner that avoids or minimizes overflow to liquid retention tanks;
- (c) Minimization or evacuation of air spaces near neutron sources to reduce the production of radionuclides (e.g. ^{41}Ar).

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

- (a) Provisions that reduce the quantity of radioactive waste generated and transferred within the facility and that minimize the activity in effluents released to the environment.
- (b) Provisions for the isolation of radioactive waste from site personnel and the public, with access control for radiation protection purposes. Controlled areas and supervised areas are required to be designated in accordance with the potential for radioactive contamination and radiation exposure (see

paras 3.88–3.92 of GSR Part 3 [2]). Recommendations on the designation of areas are provided in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [27].

- (c) Provisions for local detection, collection and treatment of liquid leakages and spills before they are discharged as effluents.
- (d) Provisions for the decontamination of personnel and equipment, including mobile devices.
- (e) Provisions for handling the radioactive waste arising from decontamination activities.

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

4.29. The varying nature of the radioactive waste generated during the lifetime of a research reactor should be taken into account in the design of the shielding, the means of confinement and the isolation features associated with facilities, equipment and components for waste management. Safety assessment of radioactive waste management activities and facilities, in accordance with the recommendations provided in IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [28], could be used for the optimization of these design features.

Management of gaseous effluents at a research reactor

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

- (a) Provision for radioactive gases to be channelled through proper ducting to a common release point;
- (b) Provisions for the proper selection of process gases and decay systems (e.g. the use of delay tanks for ^{16}N) to minimize releases of gaseous radioactive material;
- (c) Provision of appropriate filters to clean gaseous effluent to minimize the discharges while also minimizing the production of secondary waste (see also para. 6.211 of SSR-3 [1]);
- (d) Provision of means for the discharge of gases, such as a suitable stack;
- (e) Provision of means and methods for sampling and monitoring discharges of gaseous effluents in accordance with para. 7.117 of SSR-3 [1].