

IAEA Safety Standards

for protecting people and the environment

Safety of Research Reactors

Safety Requirements

No. NS-R-4



IAEA

International Atomic Energy Agency

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SAFETY 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. NS-R-4

SAFETY OF RESEARCH REACTORS

SAFETY REQUIREMENTS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2005

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FOREWORD

by Mohamed ElBaradei
Director General

The IAEA's Statute authorizes the Agency to establish safety standards to protect health and minimize danger to life and property — standards which the IAEA must use in its own operations, and which a State can apply by means of its regulatory provisions for nuclear and radiation safety. A comprehensive body of safety standards under regular review, together with the IAEA's assistance in their application, has become a key element in a global safety regime.

In the mid-1990s, a major overhaul of the IAEA's safety standards programme was initiated, with a revised oversight committee structure and a systematic approach to updating the entire corpus of standards. The new standards that have resulted are of a high calibre and reflect best practices in Member States. With the assistance of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its safety standards.

Safety standards are only effective, however, if they are properly applied in practice. The IAEA's safety services — which range in scope from engineering safety, operational safety, and radiation, transport and waste safety to regulatory matters and safety culture in organizations — assist Member States in applying the standards and appraise their effectiveness. These safety services enable valuable insights to be shared and I continue to urge all Member States to make use of them.

Regulating nuclear and radiation safety is a national responsibility, and many Member States have decided to adopt the IAEA's safety standards for use in their national regulations. For the Contracting Parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by designers, manufacturers and operators around the world to enhance nuclear and radiation safety in power generation, medicine, industry, agriculture, research and education.

The IAEA takes seriously the enduring challenge for users and regulators everywhere: that of ensuring a high level of safety in the use of nuclear materials and radiation sources around the world. Their continuing utilization for the benefit of humankind must be managed in a safe manner, and the IAEA safety standards are designed to facilitate the achievement of that goal.

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IAEA SAFETY STANDARDS

SAFETY THROUGH INTERNATIONAL STANDARDS

While safety is a national responsibility, international standards and approaches to safety promote consistency, help to provide assurance that nuclear and radiation related technologies are used safely, and facilitate international technical cooperation and trade.

The standards also provide support for States in meeting their international obligations. One general international obligation is that a State must not pursue activities that cause damage in another State. More specific obligations on Contracting States are set out in international safety related conventions. The internationally agreed IAEA safety standards provide the basis for States to demonstrate that they are meeting these obligations.

THE IAEA STANDARDS

The IAEA safety standards have a status derived from the IAEA's Statute, which authorizes the Agency to establish standards of safety for nuclear and radiation related facilities and activities and to provide for their application.

The safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment.

They are issued in the IAEA Safety Standards Series, which has three categories:

Safety Fundamentals

- Presenting the objectives, concepts and principles of protection and safety and providing the basis for the safety requirements.

Safety Requirements

- Establishing the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements, which are expressed as 'shall' statements, are governed by the objectives, concepts and principles of the Safety Fundamentals. If they are not met, measures must be taken to reach or restore the required level of safety. The Safety Requirements use regulatory language to enable them to be incorporated into national laws and regulations.

Safety Guides

- Providing recommendations and guidance on how to comply with the Safety Requirements. Recommendations in the Safety Guides are expressed as 'should' statements. It is recommended to take the measures stated 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. Each Safety Requirements publication is supplemented by a number of Safety Guides, which can be used in developing national regulatory guides.

The IAEA safety standards need to be complemented by industry standards and must be implemented within appropriate national regulatory infrastructures to be fully effective. The IAEA produces a wide range of technical publications to help States in developing these national standards and infrastructures.

MAIN USERS OF THE STANDARDS

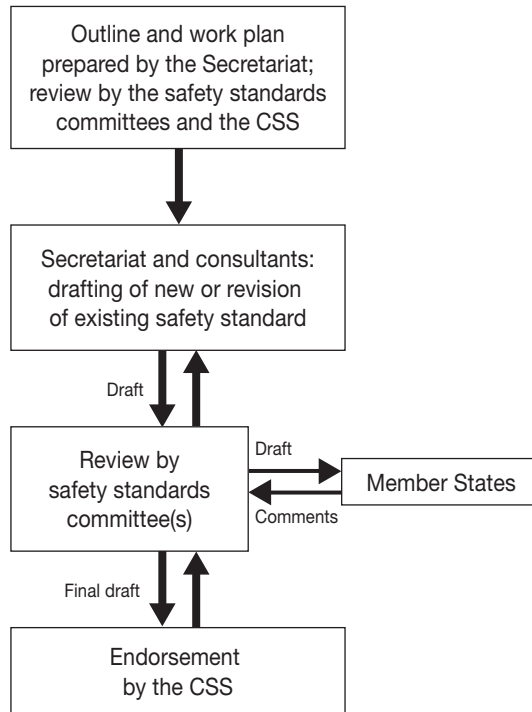
As well as by regulatory bodies and governmental departments, authorities and agencies, the standards are used by authorities and operating organizations in the nuclear industry; by organizations that design, manufacture and apply nuclear and radiation related technologies, including operating organizations of facilities of various types; by users and others involved with radiation and radioactive material in medicine, industry, agriculture, research and education; and by engineers, scientists, technicians and other specialists. The standards are used by the IAEA itself in its safety reviews and for developing education and training courses.

DEVELOPMENT PROCESS FOR THE STANDARDS

The preparation and review of safety standards involves the IAEA Secretariat and four safety standards committees for safety in the areas of 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 entire safety standards programme. All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of the CSS is appointed by the Director General and includes senior government officials having responsibility for establishing national standards.

For Safety Fundamentals and Safety Requirements, the drafts endorsed by the Commission are submitted to the IAEA Board of Governors for approval for publication. Safety Guides are published on the approval of the Director General.

Through this process the standards come to represent a consensus view of the IAEA's Member States. 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 standards. Some 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 International



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

Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

The safety standards are kept up to date: five years after publication they are reviewed to determine whether revision is necessary.

APPLICATION AND SCOPE OF THE STANDARDS

The IAEA Statute makes the safety standards binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA concerning any form of Agency assistance is required to comply with the requirements of the safety standards that pertain to the activities covered by the agreement.

International conventions also contain similar requirements to those in the safety standards, and make them binding on contracting parties. The Safety Fundamentals were used as the basis for the development of the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The Safety

Requirements on Preparedness and Response for a Nuclear or Radiological Emergency reflect the obligations on States under the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency.

The safety standards, incorporated into national legislation and regulations and supplemented by international conventions and detailed national requirements, establish a basis for protecting people and the environment. However, there will also be special aspects of safety that need to be assessed case by case at the national level. For example, many of the safety standards, particularly those addressing planning or design aspects of safety, are intended to apply primarily to new facilities and activities. The requirements and recommendations specified in the IAEA safety standards might not be fully met at some facilities built to earlier standards. The way in which the safety standards are to be applied to such facilities is a decision for individual States.

INTERPRETATION OF THE TEXT

The safety standards use the form 'shall' in establishing international consensus requirements, responsibilities and obligations. Many requirements are not addressed to a specific party, the implication being that the appropriate party or parties should be responsible for fulfilling them. Recommendations are expressed as 'should' statements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures) for complying with the requirements.

Safety related terms are to be interpreted as stated in the IAEA Safety Glossary (<http://www-ns.iaea.org/standards/safety-glossary.htm>). 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 within the 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 main text (e.g. material that is subsidiary to or separate from the main text, is included in support of statements in the main text, or describes methods of calculation, experimental 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 standard. Material in an appendix has the same status as the main 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. An annex is not an integral part of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material published in standards that is under other authorship may be presented in annexes. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

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1. INTRODUCTION

BACKGROUND

1.1. The present Safety Requirements publication, which was developed under the IAEA programme on the safety of research reactors, is a revision of two safety standards issued in the former IAEA Safety Series.¹ This publication supersedes and updates the content of these two safety standards.

1.2. This Safety Requirements publication establishes requirements for all the important areas of the safety of research reactors, with particular emphasis on requirements for design and operation.² Pursuant to requests from end users (mainly from Member States with small nuclear power programmes) to have a single self-standing publication, it also includes requirements on regulatory control, management, verification of safety, quality assurance and site evaluation.³

1.3. A number of requirements for the safety of nuclear research reactors are the same as or similar to those for nuclear power reactors. In view of the important differences between power reactors and research reactors and

¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Design, Safety Series No. 35-S1, IAEA, Vienna (1992); Code on the Safety of Nuclear Research Reactors: Operation, Safety Series No. 35-S2, IAEA, Vienna (1992).

² The important areas of research reactor safety include all activities performed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified. This includes maintenance, testing and inspection, fuel handling and handling of radioactive material (including the production of radioisotopes), the installation, testing and operation of experimental devices, the use of neutron beams, research and development work and education and training using the research reactor systems, and other associated activities.

³ The site area is the geographical area that contains an authorized facility, and within which the management of the authorized facility may directly initiate emergency actions. The site boundary is the boundary of the site area. Siting (site evaluation) is the process of selecting a suitable site for a facility, including appropriate assessment and definition of the related design bases.

between the different types of research reactors,⁴ these requirements are to be applied in accordance with the potential hazards associated with the reactor by means of a graded approach (see paras 1.11–1.14), thereby ensuring safety in the design and operation of research reactors.

OBJECTIVE

1.4. The main objective of this Safety Requirements publication is to provide a basis for safety and a basis for safety assessment for all stages in the lifetime of a research reactor. Another objective is to establish requirements on aspects relating to regulatory control, the management of safety, site evaluation, design, operation and decommissioning.

1.5. Technical and administrative requirements for the safety of research reactors are established in accordance with these objectives. This Safety Requirements publication is intended for use by organizations engaged in the site evaluation, design, manufacturing, construction, operation and decommissioning of research reactors as well as by regulatory bodies.

SCOPE

1.6. The requirements established in this Safety Requirements publication are applicable for the site evaluation, design, operation and decommissioning of research reactors, including critical assembly facilities, and are also to be applied to existing research reactors to the extent practicable. Subcritical assembly facilities are not subject to these requirements.

1.7. For the purposes of this publication, a research reactor is a nuclear reactor used mainly for the generation and utilization of radiation for research and other purposes, such as the production of radioisotopes. This definition excludes nuclear reactors used for the production of electricity, naval propulsion, desalination or district heating. The term covers the reactor core,

⁴ A research reactor is a nuclear reactor used mainly for the generation and utilization of the neutron flux and ionizing radiation for research and other purposes. In the context of this Safety Requirements publication, the term research reactor also includes associated experimental devices (see footnote 5) and critical assemblies.

experimental devices⁵ and all other facilities relevant to either the reactor or its associated experimental devices located on the reactor site. Additional safety measures may be required in some cases, as mentioned in para. 1.9.

1.8. The requirements established in this Safety Requirements publication form the basis for the safety of research reactors with a limited potential for hazard to the public and the environment.

1.9. Research reactors with power levels in excess of several tens of megawatts, fast reactors, and reactors using experimental devices such as high pressure and temperature loops, cold neutron sources and hot neutron sources may require the application of standards for power reactors and/or additional safety measures (e.g. in the case of reactors used for testing hazardous material). For facilities of these kinds, the standards to be applied, the extent of their application and any additional safety measures that may need to be taken are required to be proposed by the operating organization and to be subject to approval by the regulatory body.

1.10. All the requirements established here are to be applied unless it can be justified that, for a specific research reactor, certain requirements may be waived. For each such case the requirements to be waived shall be identified, with account taken of the nature and possible magnitude of the hazards presented by the research reactor and the activities conducted. Paragraph 1.14 sets out the factors to be considered in deciding whether certain requirements established here may be waived.

Graded approach

1.11. Research reactors are used for special and varied purposes, such as research, training, radioisotope production, neutron radiography and material testing. These purposes call for different design features and different operational regimes. Design and operating characteristics of research reactors may vary significantly since the use of experimental devices may affect the performance of reactors. In addition, the need for flexibility in their use requires a different approach to achieving and managing safety.

⁵ An experimental device is a device installed in or around a reactor to utilize the neutron flux and ionizing radiation from the reactor for research, development, isotope production or any other purpose.

1.12. Most research reactors have a small potential for hazards to the public compared with power reactors, but they may pose a greater potential for hazards to operators.

1.13. The scope, extent and detail of the safety analysis for low power research reactors may be significantly less than is required for high power research reactors because certain accident scenarios may not apply or may need only a limited analysis. For example, the treatment of loss of coolant accidents may differ significantly, depending on the power and design of the reactor. Paragraphs 6.72–6.78 establish requirements for the scope, factors and process to be considered in the safety analysis.

1.14. The factors to be considered in deciding whether certain requirements established here may be waived in applying a graded approach include:

- (a) The reactor power;
- (b) The source term;
- (c) The amount and enrichment of fissile and fissionable material;
- (d) Spent fuel elements, high pressure systems, heating systems and the storage of flammables, which may affect the safety of the reactor;
- (e) The type of fuel elements;
- (f) The type and the mass of moderator, reflector and coolant;
- (g) The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and additional safety features;
- (h) The quality of the containment structure or other means of confinement;
- (i) The utilization of the reactor (experimental devices, tests and reactor physics experiments);
- (j) Siting;
- (k) Proximity to population groups.

STRUCTURE

1.15. This Safety Requirements publication covers all the important stages in the lifetime of research reactor facilities, from site evaluation to design and construction, commissioning, operation, including utilization and modification⁶, and decommissioning. It consists of eight sections, an appendix and two annexes.

1.16. Section 2 introduces the general safety objectives, concepts and principles for the safety of nuclear installations with emphasis on the radiation safety and nuclear safety aspects of research reactors. This section draws on Ref. [1].

1.17. Section 3 deals with the general requirements for regulatory control as far as these are relevant for research reactors, including the corresponding steps in the licensing process for research reactors. This section draws upon other Safety Requirements publications and Safety Guides [2–7].

1.18. Section 4 deals with requirements on topics relating to the management of safety, including quality assurance and verification of safety. This section covers the general safety aspects of nuclear installations and is based on IAEA safety standards and safety related publications [1, 7–10].

1.19. Section 5 establishes requirements regarding the evaluation and selection of the reactor site and deals with the evaluation of new sites and the sites of existing reactors. This section is based on the Safety Requirements publication on Site Evaluation for Nuclear Installations [11].

1.20. Section 6 establishes requirements for the safe design of all types of research reactors with account taken of the considerations mentioned in para. 1.9.⁷

1.21. Section 7 establishes requirements for safe operation of research reactors, including commissioning, maintenance, utilization and modification. The

⁶ Modification is the deliberate changing of or addition to an existing reactor configuration, with possible implications for safety, intended to permit the continuation of the reactor's operation. It may involve safety systems, safety related items or systems, procedures, documentation or operating conditions.

⁷ This section is based on the superseded safety standard Code on the Safety of Research Reactors: Design, Safety Series No. 35-S1, IAEA, Vienna (1992).

requirements in this section are presented in greater detail in view of the conditions of operation of research reactors and the interest of operating organizations and regulatory bodies. The section is based on Refs [12–19] and the superseded Code on the Safety of Research Reactors: Operation, Safety Series No. 35-S2, IAEA, Vienna (1993).

1.22. Section 8 establishes requirements for the safe decommissioning of research reactors on the basis of Ref. [16].

1.23. The appendix provides a list of the selected postulated initiating events to be considered in the safety analysis for a research reactor.

1.24. Finally, the annexes provide a list of the safety functions of the safety systems and of other safety related items usually included in research reactor design, and examples of operational aspects that require particular attention.

2. SAFETY OBJECTIVES, CONCEPTS AND PRINCIPLES

GENERAL

2.1. The Safety Fundamentals publications on The Safety of Nuclear Installations [1] and Radiation Protection and the Safety of Radiation Sources [20] present the objectives, concepts and principles on which the requirements for minimizing the risks associated with nuclear installations are based.

SAFETY OBJECTIVES

2.2. There are three safety objectives: the first is general in nature. The other two are complementary and deal with radiation protection and the technical aspects of safety. The following paragraphs are reproduced directly from Ref. [1]:

“203. **General Nuclear Safety Objective:** *To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.*

“204. This General Nuclear Safety Objective is supported by two complementary Safety Objectives dealing with radiation protection and technical aspects. They are interdependent: the technical aspects in conjunction with administrative and procedural measures ensure defence against hazards due to ionizing radiation.

“205. **Radiation Protection Objective:** *To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.*

“206. **Technical Safety Objective:** *To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.*

“207. Safety Objectives require that nuclear installations are designed and operated so as to keep all sources of radiation exposure under strict technical and administrative control. However, the Radiation Protection Objective does not preclude limited exposure of people or the release of legally authorized quantities of radioactive materials to the environment from installations in operational states. Such exposures and releases, however, must be strictly controlled and must be in compliance with operational limits and radiation protection standards”.

2.3. Although measures are taken to limit radiation exposure in all operational states to levels that are as low as reasonably achievable and to minimize the likelihood of an accident that could lead to the loss of normal control over the source of radiation, there will remain a probability, albeit very low, that an accident might happen. Measures are therefore taken to ensure that the radiological consequences of any accident that might happen are mitigated. Such measures include: engineered safety features; on-site procedures established by the operating organization; and possibly also off-site intervention measures put in place by the appropriate authorities to mitigate radiation exposures if an accident does occur.

SAFETY CONCEPTS AND PRINCIPLES

2.4. The safety philosophy that is followed to fulfil the objectives stated in paras 203–205 of Ref. [1] relies on the defence in depth concept and on safety principles, as set out in Refs [1, 20, 21]. The safety principles encompass three areas: defence in depth, management issues and technical issues. They envisage the implementation of the *defence in depth concept*, the establishment of a *legislative and regulatory infrastructure*, the adoption of measures for the *management and verification of safety*, and the application of technical principles (*technical aspects of safety*) in the design and over the lifetime of the installation. What follows is a summary of these safety concepts and principles, which form the basis for the requirements for ensuring safety in nuclear installations, and an introduction to the sections of this publication in which the safety requirements for research reactors are established.

CONCEPT OF DEFENCE IN DEPTH⁸

2.5. The concept of defence in depth, as applied to all activities for safety, whether organizational, behavioural or design related, ensures that they are subject to overlapping provisions so that if a failure were to occur, it would be detected and compensated for or corrected by means of appropriate measures. The concept has been further elaborated in Refs [21, 23]. Application of the concept of defence in depth throughout design and operation provides a graded protection against a wide variety of transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or human action within the installation, and events that originate outside the installation.

2.6. Application of the concept of defence in depth in the design of the research reactor provides a series of levels of defence (inherent features, equipment and procedures) which are aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails. However, defence in depth shall be applied with account taken of the graded approach as mentioned in Section 1 and of the fact that many low power research reactors do not qualify for the fifth level of defence or even for the fourth level.

- (1) The aim of the first level of defence is to prevent deviations from normal operation and to prevent system failures. This leads to the requirement

⁸ The concept is adapted for research reactors from Ref. [22].

that the nuclear installation shall be soundly and conservatively designed, constructed, maintained and operated in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity. To meet this objective, careful attention is paid to the selection of appropriate design codes and materials, and to control of the fabrication of components and control of the construction, operation and maintenance of the nuclear installation.

- (2) The aim of the second level of defence is to control (by detection and intervention) deviations from operational states so as to prevent anticipated operational occurrences from escalating to accident conditions. This aim is framed in recognition of the likelihood that some postulated initiating events may occur at some point during the lifetime of the reactor, despite the precautions taken to prevent them. This level of defence necessitates the provision of specific systems, as determined in the safety analysis, and the definition of operating procedures to prevent or minimize damage resulting from such postulated initiating events.
- (3) For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events may not be arrested by a preceding level of defence and a more serious event may develop. These unlikely events are anticipated in the design basis for the research reactor, and inherent safety features, fail-safe design, additional equipment and procedures are provided to control their consequences and to achieve stable and acceptable states of the nuclear installation following such events. This leads to the requirement that engineered safety features shall be provided that are capable of transferring the research reactor first to a controlled state and subsequently to a safe shutdown state, and of maintaining at least one barrier for the confinement of radioactive material.
- (4) The aim of the fourth level of defence is to address beyond design basis accidents (BDBAs) in which the design basis may be exceeded and to ensure that radioactive releases are kept as low as practicable. The most important objective for this level is the protection of the confinement function. This may be achieved by complementary measures and procedures to prevent accident progression, and by mitigation of the consequences of selected BDBAs,⁹ in addition to emergency procedures and intervention measures. The protection provided by the means of confinement may be demonstrated by using best estimate methods.

⁹ The terms ‘severe accident’ and ‘accident management’, as defined in Ref. [22], are not used in the present Safety Requirements publication.

- (5) The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive material that may result from accident conditions. This requires the provision of an adequately equipped emergency control centre and plans for the on-site and off-site emergency response.

2.7. The defence in depth concept is applied mainly through the safety analysis and the use of sound engineering practices based on research and operational experience. This analysis is carried out in the design to ensure that the safety objectives are met. It includes a systematic critical review of the ways in which the nuclear installation's structures, systems and components (SSCs) could fail and identifies the consequences of such failures. The safety analysis therefore examines: (1) all planned normal operational modes of the nuclear installation; and its performance in (2) anticipated operational occurrences, (3) design basis accident (DBA) conditions and (4) event sequences that may lead to BDBAs. Requirements for the safety analysis in design are presented in paras 6.72–6.78. These analyses are independently assessed by the operating organization and by the regulatory body (paras 2.8–2.10).

LEGISLATIVE AND REGULATORY INFRASTRUCTURE

2.8. For a nuclear installation that is built, is in operation or is to be built (or to undergo a major modification), a legal infrastructure is required to be established that provides for the regulation of nuclear activities and for the clear assignment of responsibilities for safety. Government is responsible for the adoption of legislation that assigns the prime responsibility for safety to the operating organization and establishes a regulatory body responsible for a system of licensing (see Glossary), for the regulatory control of nuclear activities and for the enforcement of the regulations. These principles are established in Section 3 (Principles 1–3) of The Safety of Nuclear Installations (Ref. [1]) and are reproduced below:

- “(1) *The government shall establish a legislative and statutory framework for the regulation of nuclear installations. There shall be a clear separation of responsibilities between the regulatory body and the operating organization.*
- “(2) *The prime responsibility for safety shall be assigned to the operating organization.*
- “(3) *The regulatory body shall be effectively independent of the organization or body charged with the promotion or utilization of nuclear energy. It shall*

have licensing, inspection and enforcement responsibilities and shall have adequate authority, competence and resources to fulfil its assigned responsibilities. No other responsibility shall jeopardize or conflict with its responsibility for safety.”

2.9. General requirements to fulfil these principles are presented in Ref. [2]. This Safety Requirements publication establishes requirements for the development of the legal infrastructure for establishing a regulatory body and other actions to achieve effective regulatory control over facilities and activities. These facilities and activities include nuclear power plants and other nuclear reactors such as research reactors (see footnote 4). These requirements therefore also apply to the general legal and governmental infrastructure for the safety of research reactors during site selection, design, construction, commissioning, operation, utilization, modification and decommissioning.

2.10. Regulatory control over safety is maintained primarily through the issue of governmental licences that authorize, usually in stages, the development of the research reactor project and place conditions on the licensee¹⁰ (see the Glossary). A primary task of the regulatory body is therefore to decide whether or not to approve the application for a licence within the framework of a licensing process on the basis of its review and assessment of the proposals submitted by the operating organization. One of the ways in which the operating organization demonstrates that it has achieved adequate safety for the research reactor is through the information normally incorporated into a safety analysis report (SAR). The information in the SAR also constitutes the prime basis for the regulatory decision on licensing the nuclear installation and the requirements against which it is licensed and inspected. The content of the SAR may differ between Member States, depending upon their particular legal and regulatory system. Section 3 establishes requirements to be met in the preparation, submission and evaluation of the information included in a SAR. It is recognized in these requirements that the depth of information in the SAR should be commensurate with the potential hazard associated with the nuclear installation under consideration and the particular stage of the licensing process. Guidance on meeting these requirements is provided in Ref. [7].

¹⁰ The licensee is the holder of a current licence issued by the regulatory body granting authorization to perform specified activities relating to the research reactor facility. The applicant becomes the licensee on receipt of a licence issued by the regulatory body.

MANAGEMENT OF SAFETY

2.11. Management of safety encompasses all the principles pertaining to general management, including management of personnel, that form the basis for the measures required to ensure that an acceptable level of safety is maintained throughout the lifetime of the installation, including decommissioning. The starting point for the management of safety is the senior managers of all organizations concerned. “The principles of safety management broadly apply to all organizations. Thus, the practices described for the operating organization apply, where relevant, to other organizations with safety responsibilities” (Ref. [1], para. 402). The principles for the management of safety are established in Section 4 (Principles 4–8) of Ref. [1] and are reproduced below:

- “(4) *Organizations engaged in activities important to safety shall establish policies that give safety matters the highest priority, and shall ensure that these policies are implemented within a managerial structure having clear divisions of responsibility and clear lines of communication.*
- “(5) *Organizations engaged in activities important to safety shall establish and implement appropriate quality assurance programmes [see footnote 14] which extend throughout the life of the installation, from siting and design through to decommissioning.*
- “(6) *Organizations engaged in activities important to safety shall ensure that there are sufficient numbers of adequately trained and authorized staff working in accordance with approved and validated procedures.*
- “(7) *The capabilities and limitations of human performance shall be taken into account at all stages in the life of the installation.*
- “(8) *Emergency plans for accident situations shall be prepared and appropriately exercised by all organizations concerned. The capability to implement emergency plans shall be in place before an installation commences operation.”*

2.12. The management of safety at the installation will be effective if the operating organization develops a safety culture to a high level. The safety culture will influence the actions and interactions of all individuals and organizations engaged in activities relating to nuclear technology. The concept of safety culture is described in Ref. [8], which sets conditions at three levels: (a) at the policy level; (b) for managers; and (c) for individuals. Other principles in para. 2.11 refer to other responsibilities of the operating organization to ensure safety. General and specific requirements in respect of organization and responsibilities, the training of personnel, human factors and

emergency preparedness for research reactors are established in Sections 4 and 7.

2.13. General requirements to fulfil the principle concerning quality assurance programmes are established in the IAEA Code and Safety Guides on Quality Assurance for Safety in Nuclear Power Plants and Other Nuclear Installations [9] (see footnote 14). While some of these requirements are quoted in Section 4, the present Safety Requirements publication also includes specific requirements in respect of quality assurance for nuclear research reactors.

2.14. Accident prevention is the first priority of the reactor designer and of the operating organization. Nevertheless, accidents can happen, even though the probability that they will occur is very low. The operating organization therefore needs to make arrangements for effective procedures and emergency planning and preparedness to cope with accidents. The capability to implement emergency plans needs to be exercised regularly to the level necessary to ensure the preparedness of the operating organization. Requirements for emergency planning are presented in Section 7.

VERIFICATION OF SAFETY

2.15. The principles for the verification of safety are set out in Ref. [1] (Principles 24 and 25) and are reproduced below:

“(24) *The operating organization shall verify by analysis, surveillance, testing and inspection that the physical state of the installation and its operation continue in accordance with operational limits and conditions, safety requirements and the safety analysis.*

“(25) *Systematic safety reassessments of the installation in accordance with the regulatory requirement shall be performed throughout its operational lifetime, with account taken of operating experience and significant new safety information from all relevant sources.*”

2.16. Activities for systematic periodic assessments include, among others, periodic reviews such as self-assessment reviews and peer reviews¹¹ to confirm that the SAR and other selected documents (such as documentation for operational limits and conditions (OLCs), maintenance and training) for the installation remain valid or, if necessary, to make improvements. In such reviews, the cumulative effects of modifications, changes to procedures, the ageing of components, the use of feedback from operating experience and technical developments need to be considered, and it is necessary to verify that selected SSCs and software comply with the design requirements. Specific requirements on these topics for nuclear research reactors are established in Sections 4 (for general purpose and scope) and 7 (for operational issues).

TECHNICAL ASPECTS OF SAFETY

2.17. There are several underlying technical principles that are essential to the successful application of safety technology for nuclear installations. They are established in Section 5 (Principles 9–23) of Ref. [1] and relate to: site evaluation and selection (Principle 9); design and construction (Principles 10–15); commissioning (Principle 16); operation and maintenance (Principles 17–21); and radioactive waste management for and decommissioning of nuclear installations (Principles 22–23). The following paragraphs summarize these principles.

2.18. From Section 5 of Ref. [1]:

“(9) *The site selection shall take into account relevant features that might affect the safety of the installation, or be affected by the installation, and the feasibility of carrying out emergency plans. All aspects shall be evaluated for the projected lifetime of the installation and re-evaluated as necessary to ensure the continued acceptability for safety of site related factors.*”

¹¹ A peer review is a review conducted by a team of independent experts with technical competence and experience in the areas of evaluation. Judgements are based on the combined expertise of the team members. The objectives, scope and size of the review team are tailored to the review that is to be conducted. A review is neither an inspection nor an audit against specific standards. Instead, it consists of a comprehensive comparison of the practices applied by organizations with internationally accepted good practices, and an exchange of expert judgement.

Potential sites shall be evaluated for human made and natural factors that could adversely affect the safety of the installation. The effects that the installation may have on the surrounding population and on the environment, for example through use of the land and water, shall also be evaluated. The basis for the selection of a site for a research reactor will vary, depending on a number of factors, including the design of the reactor and its intended uses. Certain low power research reactors may impose minimal siting constraints. Research reactors designed to achieve significant power levels and to be used for extensive experimental testing will necessitate more stringent requirements for siting and design, which are established in Ref. [11]. General and specific requirements to fulfil the above principles are established in Section 5 of this publication.

2.19. The principles for the design and construction of nuclear installations are established in Section 5 of Ref. [1] and are reproduced below:

- “(10) *The design shall ensure that the nuclear installation is suited for reliable, stable and easily manageable operation. The prime goal shall be the prevention of accidents.*
- “(11) *The design shall include the appropriate application of the defence in depth principle so that there are several levels of protection and multiple barriers to prevent releases of radioactive materials, and to ensure that failures or combinations of failures that might lead to significant radiological consequences are of very low probability.*
- “(12) *Technologies incorporated in a design shall be proven or qualified by experience or testing or both.*
- “(13) *The systematic consideration of the human-machine interface and human factors shall be included in all stages of design and in the associated development of operational requirements.*
- “(14) *The exposure to radiation of site personnel and releases of radioactive materials to the environment shall be made by design as low as reasonably achievable.*
- “(15) *A comprehensive safety assessment and independent verification shall be carried out to confirm that the design of the installation will fulfil the safety objectives and requirements, before the operating organization completes its submission to the regulatory body.”*

2.20. To comply with the safety objectives set out in para. 2.2, the design and construction of the nuclear installation shall ensure: (a) the limitation of radiation exposures, radioactive releases and the generation of radioactive waste in all operational states, as far as is reasonably achievable; (b) the

prevention of accidents that could affect site personnel, the public and the environment; and (c) the limitation and mitigation of the consequences of accidents if they do occur. Consequently, the design shall use or apply:

- (a) Components, systems and structures with high reliability;
- (b) Specific considerations in design to minimize personnel exposures;
- (c) The appropriate classification of SSCs, including software, that are items important to safety, on the basis of their safety significance;
- (d) The single failure criterion, to ensure that no single failure or single maintenance action or any other single human action could disable a safety function;
- (e) Features to minimize the possibility of failures due to a common cause by means of the independence, physical separation and diversity of equipment;
- (f) Technology that is proven, or qualified by experience or testing or both, and that meets conservative regulations or criteria with appropriate safety margins;
- (g) Appropriate inherent and engineered safety features;
- (h) Fail-safe design concepts where practicable.

Some of the above items, such as (e), (f), (g) and (h), may not apply to experimental devices. The design shall also take into account the capabilities for performance of the operating and maintenance personnel. Attention to human factors will ensure that the installation is tolerant of human errors. Among the appropriate elements in minimizing human errors are: the systematic application of ergonomic principles to the relevant engineered systems; the provision of automatic control, protection and alarm systems; the elimination of human actions that jeopardize safety; the clear presentation of data; and reliable communications (see also para. 2.23).

2.21. The construction of an installation shall start only after the operating organization has satisfied itself by means of verification that the main safety issues in the design have been resolved; and after the regulatory body has satisfied itself, by means of review and assessment, of the adequacy of the safety analysis submitted, and the adequacy of the proposed arrangements, procedures and quality assurance programmes for implementing the design throughout construction. In this regard, the responsibility for ensuring that the construction is in accordance with the design and with quality assurance programmes lies with the operating organization. General and specific requirements for technical aspects of the design and construction of research reactors are included in Section 6.

2.22. The operating organization shall establish an adequate and appropriate organization for the operation of the nuclear installation, which shall undertake an appropriate and adequate commissioning process. The purpose of commissioning is to demonstrate that the design specifications of the installation have been met and that the completed installation is satisfactory for service. From Section 5 of Ref. [1]:

“(16) Specific approval by the regulatory body shall be required before the start of normal operation on the basis of an appropriate safety analysis and a commissioning programme. The commissioning programme shall provide evidence that the installation as constructed is consistent with design and safety requirements. Operating procedures shall be validated to the extent practicable as part of the commissioning programme, with the participation of the future operating staff.”

Requirements in respect of the commissioning of research reactors are established in Section 7.

2.23. The principles for the operation and maintenance of a nuclear installation are established in Section 5 of Ref. [1] and are reproduced below:

“(17) A set of operational limits and conditions derived from the safety analysis, tests and subsequent operational experience shall be defined to identify safe boundaries for operation. The safety analysis, operating limits and procedures shall be revised as necessary if the installation is modified.

“(18) Operation, inspection, testing and maintenance and supporting functions shall be conducted by sufficient numbers of adequately trained and authorized personnel in accordance with approved procedures.

“(19) Engineering and technical support, with competence in all disciplines important for safety, shall be available throughout the lifetime of the installation.

“(20) The operating organization shall establish documented and approved procedures as a basis for operator response to anticipated operational occurrences and accidents.

“(21) The operating organization shall report incidents significant to safety to the regulatory body. The operating organization and the regulatory body shall establish complementary programmes to analyse operating experience to ensure that lessons are learned and acted upon. Such experience shall be shared with relevant national and international bodies.”

The operation of the installation shall be controlled in accordance with a set of OLCs, derived from the safety analysis, that identify safe boundaries of operation. Competent technical support for the operation of the installation shall be made available. Operations shall be carried out by adequately trained and authorized personnel in accordance with written and validated operating procedures for normal operation and anticipated operational occurrences. A quality assurance programme (see footnote 14) shall be established. Procedures to manage accident conditions shall be in place. The installation shall be regularly inspected, tested and maintained in accordance with an approved programme, which is implemented with procedures to ensure that the SSCs continue to be available and to operate as intended, and that they retain their capability to meet the design objectives and the requirements of the safety analysis. A programme for the safe utilization and modification of the installation shall be in place. Periodic reviews shall be conducted to ensure that the safety analysis report, the OLCs and operating procedures remain valid, with account taken of current operational issues, such as those relating to ageing, operating experience and currently applicable safety standards. Exposures of site personnel to radiation and releases of radioactive material shall be minimized and controlled as far as is reasonably achievable. The operating organization shall establish a programme for the collection and analysis of operating experience. Safety significant information shall be disseminated to all those concerned. General and specific requirements for operation and maintenance of nuclear research reactors are established in Section 7.

2.24. The principles for radioactive waste management and for the decommissioning of nuclear installations are established in Section 5 of Ref. [1] and are reproduced below:

- “(22) The generation of radioactive waste, in terms of both activity and volume, shall be kept to the minimum practicable by appropriate design measures and operating practices. Waste treatment and interim storage shall be strictly controlled in a manner consistent with the requirements for safe final disposal.*
- “(23) The design of an installation and the decommissioning programme shall take into account the need to limit exposures during decommissioning to as low as is reasonably achievable. Prior to the initiation of decommissioning activities, the decommissioning programme shall be approved by the regulatory body.”*

General requirements and guidance for waste management and the decommissioning of nuclear installations are established in several IAEA

safety standards. Principles, concepts and objectives of radioactive waste management are set out in Ref. [17]. Requirements for discharges of radioactive material and disposal of radioactive waste, including decommissioning, are established in Ref. [14]. Supporting guidance is provided in Refs [13, 16]. Specific requirements for the management of radioactive waste and for dealing with the decommissioning of research reactors are included in Sections 7 and 8.

3. REGULATORY SUPERVISION

GENERAL

3.1. This section establishes requirements relating to general aspects of the legal and governmental infrastructure for the safety of research reactors. Requirements that apply to the regulatory supervision of nuclear facilities are established in Ref. [2]. Guidance on how to meet these requirements is provided in the associated Safety Guides [3–6].

LEGAL INFRASTRUCTURE

3.2. The government shall ensure that an adequate legal infrastructure and regulatory basis for assessing the safety of the research reactor is available. The government is responsible for adopting the necessary legislation, which shall assign the prime responsibility for safety to the operating organization. “The regulatory regime shall be structured and resourced in a manner commensurate with the potential magnitude and nature of the hazard to be controlled” (Ref. [2], para. 2.1). This legislation shall provide for the establishment and maintenance of a regulatory body “which shall be effectively independent of organizations or bodies charged with the promotion of nuclear technologies or responsible for facilities or activities” (Ref. [2], para. 2.2(2)).

REGULATORY BODY

3.3. To be effective, the regulatory body shall be provided with the legal powers and statutory authority necessary to ensure that it can discharge its

responsibilities and fulfil its functions. Such powers usually include the authority to review and assess safety related information submitted by the operating organization during the licensing process and to apply the relevant regulations (e.g. by issuing, amending or revoking licences or licence conditions), including carrying out compliance inspections and audits, taking enforcement action and providing other competent authorities or the public with information, as appropriate.

LICENSING PROCESS

General

3.4. The licensing process may vary among Member States but in all cases the major stages of the licensing process for nuclear research reactors shall include the regulation of:

- (a) Site evaluation;
- (b) Design and construction;
- (c) Commissioning;
- (d) Operation, including utilization and modification;¹²
- (e) Decommissioning.

3.5. The licensing process is ongoing, starting at the site evaluation stage and continuing up to and including the decommissioning of the research reactor. While licensing steps and procedures vary among Member States, the first formal licensing action will be the authorization of the safety concept and the design and the issuing of a construction licence for an evaluated site. In some cases, only a single licence is issued for the project, but conditions are attached to it to effect control over subsequent stages (see the Appendix of Ref. [6]). Despite these differences between national practices, a detailed demonstration of safety in the form of a SAR, which includes an adequate safety analysis, shall be submitted by the operating organization to the regulatory body. The SAR shall be reviewed and assessed by the regulatory body before the project is

¹² Although the utilization and modification of research reactors are activities that are normally included under operation, they may be considered separate stages in the licensing process since their safety implications give rise to a large number of review and assessment activities that are repeated many times over the reactor's lifetime (see paras 7.87–7.94).

authorized to progress to the next stage. Close liaison shall be maintained between the regulatory body and the operating organization throughout the entire process of regulatory supervision of the installation.

Safety analysis report

3.6. The SAR shall be prepared by the operating organization for the justification of the site and design and shall be the basis for the safe operation of the research reactor. The SAR is an important link between the operating organization and the regulatory body since it is the main document for the licensing of the reactor. It shall be updated during the operational lifetime of the reactor on the basis of the experience and knowledge gained and in accordance with regulatory requirements. Further guidance on the preparation and assessment of the SAR is provided in Ref. [7].

3.7. The SAR shall give a detailed description of the reactor site, the reactor, experimental devices and all other facilities and activities with safety significance. It shall provide a detailed description of the general safety principles and criteria applied to the design for the protection of the reactor, the operating personnel¹³, other on-site personnel, the public and the environment. It shall analyse the potential hazards associated with the operation of the reactor. The SAR shall include safety analyses of accident sequences and shall describe the safety features incorporated in the design to avoid or to minimize the likelihood of occurrence of accidents, or to mitigate their consequences through design and operating procedures.

3.8. The SAR shall form the basis for establishing the OLCs for the reactor. It shall also provide details as to how the operating organization intends to organize and conduct operations and as to the quality assurance programme (see footnote 14) for all stages of reactor life, including design and construction. It shall also provide details of the emergency plan of the research reactor.

3.9. Beyond the items discussed in paras 3.7 and 3.8, the SAR shall include additional information as prescribed in national legislation and by the regulatory body. Guidance on the information to be included in a typical SAR is presented in Ref. [7]. The level of detail of the information to be presented in the SAR shall be determined in accordance with the type, characteristics (its

¹³ The operating personnel comprise the reactor manager, the shift supervisors, the operators, the maintenance staff and the radiation protection staff.

design, power and usage) and site of the reactor. For reactors with higher power levels, accident scenarios will usually require more details of the site and of the safety features for protecting against any significant releases of radioactive material to the environment. For some reactors (e.g. critical assemblies or low power reactors) the requirements for the safety analysis may be less extensive (see also para. 1.13). However, as the SAR may be the only comprehensive document produced concerning the safety of the facility, every topic mentioned in paras 3.6–3.8 should be considered in it.

3.10. The SAR shall cite the technical literature in the form of references that may be necessary for a thorough review and assessment process. This reference material shall be readily available to the regulatory body and shall not be subject to any classification or limitation that would prevent its adequate review and assessment.

Review and assessment by the regulatory body

3.11. A review and assessment of the information (usually in the form of a SAR) submitted by the operating organization in support of its licence application shall be performed by the regulatory body to determine whether the proposed facility can be sited, constructed, commissioned, operated, utilized, modified and decommissioned without undue radiological risks to the personnel at the site, the public and the environment. The review and assessment shall be performed in accordance with the potential magnitude of the hazard associated with the research reactor (see also paras 1.11–1.14). Within this general aim, the review and assessment shall have the following specific objectives:

- (a) To determine whether the site is adequate for the type, power and usage of the proposed research reactor facility.
- (b) To determine before construction whether the proposed design of the facility (systems or modifications) meets the regulatory body's requirements and to impose any further requirements or conditions that may be deemed necessary by the regulatory body.
- (c) To determine whether the applicant has the ability, reliability, resources, organizational structure and competent personnel to meet the regulatory requirements; in particular, whether the personnel requiring a licence at the research reactor facility have been appropriately and adequately trained and have been licensed.
- (d) To determine whether the construction remains consistent with the regulatory body's requirements.

- (e) To determine whether the commissioning programme is adequate and whether its results conform to the intentions in the design.
- (f) To determine whether the OLCs, including the actions required to be taken when a safety limit or a limiting condition is violated, are justified and consistent with the regulatory requirements and whether an adequate level of operational safety can be ensured.
- (g) To determine whether the operation, utilization and procedures for modification of the facility meet the requirements of the regulatory body.
- (h) To determine whether the proposed decommissioning process meets the regulatory requirements.
- (i) To ensure that all design and operational activities are conducted in such a way as to facilitate ultimate decommissioning.
- (j) To ensure that financial instruments for decommissioning are put in place.
- (k) To determine whether periodic summary reports and incident reports are in accordance with the regulatory requirements.
- (l) To determine whether systematic safety reassessments are sufficiently comprehensive and whether account is taken of operating experience and new safety related information.

3.12. A schedule for the submission of documents for review and assessment that sets out the appropriate stages in the licensing process shall be agreed at an early date.

Acceptance criteria

3.13. States shall develop their own approach to acceptance criteria, depending upon their particular legal and regulatory infrastructures. Acceptance criteria that are chosen on the basis of suitable principles for safe design and operation shall be made available to the operating organization.

INSPECTION AND ENFORCEMENT

3.14. Paragraphs 5.12 and 5.13 of Ref. [2] establish the general requirements for inspection and enforcement.

3.15. The regulatory body shall establish a planned and systematic inspection programme. The scope of this programme and the frequency of inspections shall be commensurate with the potential hazard posed by the research reactor.

3.16. If there is evidence of a deterioration in the level of safety, or in the event of serious violations which in the judgement of the regulatory body could pose

an imminent radiological hazard to the workers, the public or the environment, the regulatory body shall require the operating organization to curtail its activities and to take any further actions necessary to restore an adequate level of safety. In the event of continual, persistent or extremely serious non-compliance, the regulatory body shall direct the operating organization to curtail its activities and may suspend or revoke the authorization.

4. MANAGEMENT AND VERIFICATION OF SAFETY

RESPONSIBILITIES OF THE OPERATING ORGANIZATION

General

4.1. The operating organization shall have the prime responsibility for the safety of the research reactor over its lifetime, from the beginning of the project for site evaluation, design and construction, through to commissioning, operation, utilization, modification and decommissioning. In order to ensure rigour and thoroughness at all levels of the staff in the achievement and maintenance of safety, the operating organization shall:

- (a) Establish and implement safety policies and ensure that safety matters are given the highest priority;
- (b) Clearly define responsibilities and accountabilities with corresponding lines of authority and communication;
- (c) Ensure that it has sufficient staff with appropriate education and training at all levels;
- (d) Develop and strictly adhere to sound procedures for all activities that may affect safety, ensuring that managers and supervisors promote and support good safety practices while correcting poor safety practices;
- (e) Review, monitor and audit all safety related matters on a regular basis, implementing appropriate corrective actions where necessary;
- (f) Be committed to safety culture on the basis of a statement of safety policy and safety objectives which is prepared and disseminated and is understood by all staff.

The functions and responsibilities of the operating organization for ensuring safety in each of the above stages are presented in paras 2.11–2.23 as well as here in Section 4. Specific requirements are established in Section 5 (see

paras 5.2, 5.40), Section 6 (see para. 6.4) and Section 7. Requirements for preparing for decommissioning are established in Section 8 (see para. 8.7).

Interaction between the regulatory body and the operating organization

4.2. The operating organization shall demonstrate to the regulatory body that its responsibility for safety at all stages in the lifetime of the reactor will be discharged. Whenever a change of stage is initiated by the operating organization, it shall submit a detailed demonstration, which shall include an adequate safety analysis, for review and assessment by the regulatory body before the project is authorized to progress to the next stage.

4.3. The operating organization shall submit to the regulatory body in a timely manner any information that it has requested. The operating organization shall be responsible for making arrangements with the vendors to ensure the availability of any information that has been requested by the regulatory body. The operating organization shall also be responsible for apprising the regulatory body of any new information on the research reactor and of any changes to information submitted previously.

4.4. The format and content of documents submitted to the regulatory body by the operating organization in support of a licence application shall be based on the requirements established in paras 3.6–3.10. The regulatory body may request additional information, depending on the regulatory practices of the particular Member State.

QUALITY ASSURANCE¹⁴

4.5. The establishment, management, performance and evaluation of a quality assurance programme for a research reactor and its associated experiments are

¹⁴ The IAEA is at present revising the safety standards in the area of quality assurance that were issued as Safety Series No. 50-C/SG-Q (1996). The revised Safety Requirements publication will cover management systems for protection and safety in nuclear facilities and in activities involving the use of ionizing radiation. The term ‘management system’ has been adopted in the revised drafts instead of the terms ‘quality assurance’ and ‘quality assurance programme’. This development embraces all aspects of the management of a nuclear facility such as a research reactor, and brings the safety, health, environmental and quality assurance related requirements together in one coherent system.

important for ensuring safety. The operating organization shall establish and implement performance based quality assurance requirements for research reactors for the stages of site evaluation, design, construction, commissioning, operation, utilization, modification and decommissioning. In particular, all operational activities relating to safety, such as those mentioned in Annex II, including decommissioning, shall be covered by appropriate requirements for quality assurance.

4.6. The operating organization shall develop quality assurance programmes for all the stages in the lifetime of a research reactor at a time consistent with the schedule for accomplishing stage related activities. In particular, activities for site investigation, which are usually initiated long before the establishment of a project, shall be covered by a quality assurance programme.

4.7. Requirements for a quality assurance programme are established and objectives, principles and guidance are provided in Ref. [9]. The objectives, principles and guidance presented in Ref. [9] shall be taken into account in the preparation of the quality assurance programme for a research reactor by means of a graded approach on the basis of the importance to safety of each item, service or process. The graded approach shall be adopted so as to reflect planned and accepted differences in the application of specific quality assurance requirements to research reactors. The extent of the detailed quality assurance programme that is required for a particular research reactor or experiment shall be governed by the potential for hazard of the reactor and the experiment (see paras 1.11 and 1.14) and shall meet the requirements of the regulatory body. Further guidance on grading the quality assurance programme is provided in Ref. [10].

4.8. The quality assurance programme shall be reviewed and approved at the appropriate levels of management in the operating organization and shall be submitted to the regulatory body. The provisions of the programme shall be based on the following three functional principles:

- (a) Managers provide planning, direction, resources and support so as to achieve objectives;
- (b) Staff perform the work so as to achieve quality;
- (c) Independent assessments are made by staff in the operating organization or by an outside agency so as to evaluate the effectiveness of the management processes and the performance of work.

Management

4.9. Management shall provide and demonstrate support for the effective implementation of the quality assurance programme in all work areas. The management aspects of the quality assurance programme shall include:

- (a) A statement of the policy of the organization on quality assurance;
- (b) The organizational structure;
- (c) The functional responsibilities;
- (d) Requirements for training, qualification and certification;
- (e) Levels of authority and interfaces for those who manage, perform and evaluate the adequacy of the work.

Performance

4.10. At all stages in the lifetime of the research reactor, work shall be planned and performed in accordance with established codes, standards, specifications, procedures and administrative controls. Items and services important to safety shall be specified and controlled to ensure their proper use, maintenance and configuration.

4.11. It shall be ensured that items and services under procurement meet established requirements and perform as specified. Suppliers shall be evaluated and selected on the basis of specified criteria. Requirements for reporting deviations from procurement specifications shall be specified in the procurement documents. Evidence that purchased items and services meet procurement specifications shall be made available for verification before the items are used or the services are provided.

Assessment

4.12. The management at all levels shall periodically assess the processes for which it is responsible to determine its effectiveness in achieving the objectives for nuclear safety. Weaknesses in processes shall be identified and corrected.

4.13. Independent assessments shall be conducted on behalf of the management to measure the effectiveness of management processes and the adequacy of work performed, to monitor the quality of items and services and to promote improvements. The persons conducting the independent assessments shall not include anyone directly involved in the work being assessed.