IAEA Safety Standards

for protecting people and the environment

Safety of Research Reactors

Safety Requirements

No. NS-R-4

IAEA
International Atomic Energy Agency

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
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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

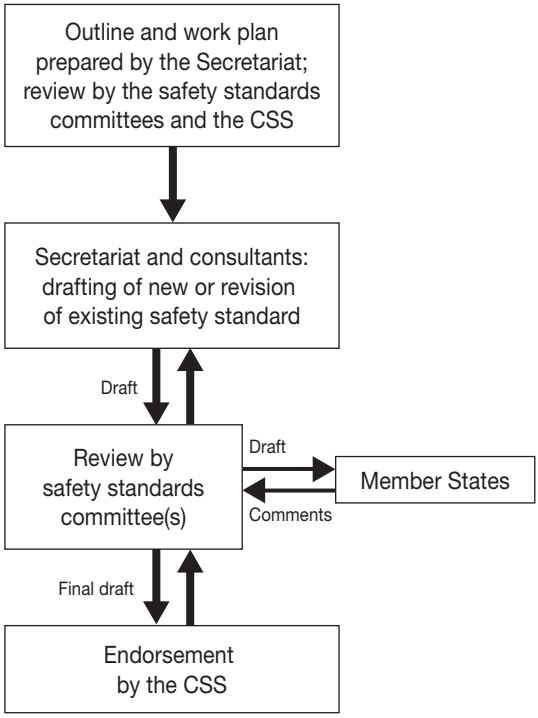
As well as by regulatory bodies and governmental departments, authorities and agencies, the standards are used by authorities and operating organizations in the nuclear industry; by organizations that design, manufacture and apply nuclear and radiation related technologies, including operating organizations 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 devices[5] 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 modification[6], 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 DEPTH[8]

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 licensee[10] (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 reviews[11] 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) E*ngineering 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 personnel[13], 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 ASSURANCE[14]

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.

VERIFICATION OF SAFETY

Safety assessments

4.14. A comprehensive safety assessment shall be carried out by the operating organization to confirm that the design meets the safety requirements set out at the beginning of the design process. The basis for this assessment shall be the data derived from the safety analysis (see para. 2.7) as well as information from other sources such as research and previous operational experience. The safety assessment shall be part of the design process, with iterations made between the design activities and the confirmatory analytical activities and with increases in the scope and the level of detail of the safety assessment as the design progresses. Methods have been developed for assessing whether safety objectives have been met. Further guidance on meeting these requirements is provided in Ref. [7]. The safety assessment shall be continued throughout all the stages in the lifetime of the reactor and it shall be conducted in accordance with the potential magnitudes and nature of the hazard associated with the particular facility or activity (see para. 5.7 of Ref. [2]).

Safety committees

4.15. One or more reactor advisory groups or safety committees that are independent of the reactor manager[15] shall be established to advise the operating organization on: (a) relevant aspects of the safety of the reactor and the safety of its utilization and (b) on the safety assessment of design, commissioning and operational issues. One of the committees shall also advise the reactor manager (see also paras 7.25 and 7.26). Members of such a group or groups shall be experts in different fields associated with the operation and design of the research reactor. It may be advisable to include external experts (i.e. from outside the operating organization) in such committees. Depending on the complexity of the operations carried out at the research reactor, one of the advisory groups could be external to the operating organization. The functions, authority, composition and terms of reference of such committees shall be documented and, if required, submitted to the regulatory body. The list

[15] The reactor manager is the member of the reactor management to whom the direct responsibility and authority for the safe operation of the research reactor is assigned by the operating organization and whose primary duties comprise the discharge of this responsibility (see paras 7.2 and 7.11).

of items that the safety committee is required to review shall also be established. Such a list shall include, among other things, the following data:

(a) Proposed changes in the OLCs in the licence for the facility;
(b) Proposed new tests, experiments, equipment, systems or procedures that have significance for safety;
(c) Proposed modifications to items important to safety and changes in experiments that have implications for safety;
(d) Violations of the OLCs, of the licence and of procedures that are significant to safety;
(e) The design, including the chemical composition, of the nuclear fuel elements[16] and the reactivity control elements;
(f) Events that are required to be reported or that have been reported to the regulatory body;
(g) Periodic reviews of the operational performance and safety performance of the facility;
(h) Reports on routine releases of radioactive material to the environment;
(i) Reports on radiation doses to the personnel at the facility and on any doses to the public.

Self-assessment and peer reviews

4.16. In order to apply the principles for the verification of safety (see paras 2.15–2.16), the operating organization shall carry out comprehensive periodic reviews of operational issues and safety related activities. The reviewing strategy and the safety factors to be evaluated shall be approved or agreed to by the regulatory body. These reviews will mainly be for identifying and solving problems concerning safety and performance and for improving safety if necessary (see also paras 7.108–7.110).

[16] The nuclear fuel elements are the elements containing fissionable and fissile nuclear material that are used in the core of a research reactor for the purpose of generating neutrons.

5. SITE EVALUATION

INITIAL EVALUATION AND SELECTION OF A SITE

Objective

5.1. The main safety objective in evaluating the site for a research reactor is the protection of the public and the environment against the radiological consequences of normal and accidental releases of radioactive material. Information shall be collected in sufficient detail to support the safety analysis to demonstrate that the research reactor facility can be safely operated at the proposed site. For low power reactors, the amount of detail to be provided can be substantially reduced below that required for a medium or high power reactor (see also paras 1.11–1.14). The results of the site evaluation shall be documented and presented in sufficient detail to permit an independent review by the regulatory body. This may constitute the first part of the development of the SAR for the research reactor.

5.2. The site evaluation shall establish the boundaries of the site area (see Glossary), which is under the control of the operating organization, and its legal rights within the area. Any activities that are unrelated to the operation of the research reactor but which will be permitted within these boundaries shall be evaluated and justified. In the evaluation of the suitability of a particular site for a research reactor, the characteristics of the site, which may affect aspects of the safety of the research reactor, shall be investigated and assessed by the operating organization. The objective of the assessment is to demonstrate how these site characteristics will influence the design criteria and operating criteria for the facility and to demonstrate the adequacy of the site characteristics in terms of effects on safety.

5.3. In the evaluation of the suitability of a site for a research reactor, the following aspects shall be considered:

(a) The effects of external events that may occur in the region of the site (the events could be of natural or human induced origin);
(b) The characteristics of the site and its environment that could influence the transfer to humans of released radioactive material;
(c) The population density and population distribution and other characteristics of the site vicinity of relevance to possible emergency

measures and the need to evaluate the risks to individuals and the population;

(d) Any other nuclear facilities at the site;

(e) The capability for an ultimate heat sink at the site.

5.4. If the site evaluation for these five factors, including their foreseeable evolution, indicates that the site is unacceptable and these deficiencies of the site cannot be compensated for by means of design features, site protection measures or administrative procedures, the site shall be deemed unsuitable. (Design features and site protection measures are the preferred means of compensating for deficiencies.)

GENERAL CRITERIA FOR SITE EVALUATION

5.5. Site characteristics that may affect the safety aspects of the research reactor shall be investigated and assessed. Environmental characteristics in the region that may be affected by potential radiological consequences of radioactive releases from the reactor in operational states and accident conditions shall be investigated. All these characteristics shall be observed and monitored throughout the lifetime of the research reactor.

5.6. The hazards associated with external events (and combinations of events) that are to be considered in the design of the reactor shall be determined. The combination of external events with anticipated operational occurrences or DBA conditions shall be considered for those cases in which an anticipated operational occurrence or a DBA condition is caused by the external event and where there is a need to consider long lasting external events (such as flooding) or long post-event recovery times.

5.7. In the analysis of the suitability of the site, consideration shall be given to matters such as storage and transport of fresh fuel, spent fuel and radioactive waste.

5.8. The potential for interaction between nuclear and non-nuclear effluents, such as the action of heat or chemicals on radioactive material in liquid effluents, should be considered.

5.9. For each proposed site the potential radiological consequences shall be evaluated for people in the region when the reactor is in operational states and

in accident conditions, including states that could lead to emergency measures being taken.

5.10. Proposed sites shall be adequately investigated with regard to all the characteristics that could affect safety in natural and human induced events.

5.11. Prehistorical, historical and instrumental information and records, as applicable, of the occurrences and severity of important natural phenomena or human induced events or activities shall be collected for the region and carefully analysed for reliability, accuracy and completeness.

5.12. In the evaluation of a site for its possible radiological consequences in the region for operational states and for accident conditions at the reactor that could lead to emergency measures being taken, appropriate estimates shall be made of expected and potential releases of radioactive material, with account taken of the design of the installation and its safety features. These estimates shall be confirmed once the design and its safety features have been established.

5.13. The region in which it is proposed to site the reactor shall be studied to evaluate the present and projected population distributions, which may influence the possible consequences of radioactive releases for individuals and the population as a whole (see also para. 5.37). If necessary, appropriate measures shall be taken to ensure that the overall risk associated with the proposed research reactor at the site remains acceptably low.

5.14. It shall be confirmed before the start of construction of the research reactor that no major problems are to be anticipated in the development of an off-site emergency plan prior to the start of its operation (see also the Appendix).

Earthquakes

5.15. The hazard for the site due to earthquake induced ground motion shall be assessed, with account taken of the seismotectonic characteristics of the region and specific site conditions. Various methods may be used to determine the earthquake hazard. The uncertainties in the methods shall be taken into consideration in deriving ground motion parameters for the design basis.

5.16. The extent and the level of detail of site investigations to determine the ground motion parameters for the design basis will depend on the installation

under consideration. For smaller installations with minimal potential for radiological consequences for people, it may be preferable (and cost-effective) to limit the site investigations and instead to use conservative values for the design basis parameters. The conservatism is necessary because in general more uncertainties will persist when the investigations are not as detailed.

Surface faulting

5.17. If there is evidence for surface faulting or if there is inadequate evidence that surface faulting has not occurred in the region, this phenomenon shall be investigated. If the site is within a zone of surface faulting that has a significant potential for relative displacement at or near the ground surface (i.e. if the fault is capable), the site shall be deemed unsuitable unless a detailed analysis proves that engineering solutions would be practicable.

EXTREME AND RARE METEOROLOGICAL EVENTS

Extreme values of meteorological phenomena

5.18. The following meteorological phenomena shall be documented for an appropriate period of time to evaluate their possible extreme values: wind, precipitation, snow, high and low temperatures and storm surges. The output of the site evaluation shall be described in a suitable way for design purposes.

Rare meteorological events

Tornadoes

5.19. The potential for tornadoes and associated missiles shall be evaluated for the region of interest, together with the hazard posed by these phenomena.

Tropical cyclones

5.20. The potential for tropical cyclones and associated missiles shall be evaluated for the region of interest, together with the hazard posed by these phenomena.

FLOODING

Floods due to precipitation and other causes

5.21. The potential for flooding due to precipitation and high water that may affect the safety of the research reactor shall be evaluated for the region.

5.22. For coastal sites and sites on estuaries, the potential shall be evaluated for flooding due to a combination of high tides, very low atmospheric pressure, wind effects on bodies of water and wave actions such as those caused by cyclones.

Water waves

5.23. The potential for tsunamis or seiches that could affect the safety of the research reactor shall be evaluated for the region.

Floods and waves caused by the failure of structures for water control

5.24. Information relating to upstream structures for water control shall be evaluated to determine whether the research reactor would be able to withstand the effects of their failure.

GEOTECHNICAL HAZARDS

Slope instability

5.25. The potential for slope instability (such as landslides, rock slides and snow avalanches) that could affect the safety of the research reactor shall be evaluated for the site and its vicinity.

Collapse, subsidence or uplift of the site surface

5.26. The potential shall be evaluated for collapse, subsidence or uplift of the site surface.

Soil liquefaction

5.27. The potential shall be evaluated for liquefaction of the subsurface materials at the proposed site.

Behaviour of foundation materials

5.28. The geotechnical characteristics of the subsurface materials and their uncertainties shall be investigated and a soil profile for the site shall be produced in a form suitable for design purposes.

Other important natural phenomena and extreme conditions

5.29. Historical data shall be collected and evaluated on phenomena that have the potential to affect the safety of the research reactor, such as data on volcanism, strong winds, the frequency and severity of lightning strikes, sand storms, severe precipitation, snow, ice, hail and subsurface freezing of subcooled water (frazil).

EXTERNAL HUMAN INDUCED HAZARDS

Aircraft crashes

5.30. The potential for aircraft crashes shall be evaluated, including impacts, fire and explosions on the site, with account taken of present and future characteristics for air traffic, the locations and types of airports, and aircraft characteristics, including aircraft with special permission to fly over or close to the facility such as fire fighting aircraft and helicopters.

Chemical explosions

5.31. Activities in the region shall be identified that involve the handling, processing, transport and storage of chemicals with a potential for causing explosions or the production of gas clouds capable of deflagration or detonation.

Other important human induced events

5.32. The vicinity of the site shall be investigated for any facilities where inflammable, toxic, corrosive or radioactive material that could affect safety may be stored, processed, transported or otherwise handled.

SPECIFIC REQUIREMENTS FOR THE CHARACTERIZATION OF THE REGION UNDER CONSIDERATION

Atmospheric dispersion of radioactive material

5.33. A meteorological description of the region, including the basic meteorological parameters and phenomena, shall be prepared. Data for at least one representative year should be presented, together with any other data that may be available from other sources. Data should be collected that adequately represent local meteorological conditions. The extent to which these data represent the long term meteorological characteristics of the site should be indicated. This information may be obtained by comparing the data for the site with concurrent and long term data from surrounding synoptic meteorological stations.

5.34. On the basis of the data obtained from the investigation of the region, the possible atmospheric dispersion of any radioactive material released shall be assessed.

Dispersion of radioactive material through surface water

5.35. A description shall be prepared of the surface hydrological characteristics of the region, including the main characteristics of water bodies, both natural and artificial, and data on water uses in the region. An evaluation shall be performed of the possible impact of the contamination of surface water on the critical group.

Dispersion of radioactive material through groundwater

5.36. A description shall be prepared of the groundwater hydrology of the region, including the main characteristics of the water bearing formations, their interactions with surface waters and data on the uses of groundwater in the region. An evaluation shall be performed of the possible impact of the contamination of groundwater on the critical group.

Population distribution

5.37. The distribution of the population within the region shall be determined. In particular, information shall be collected on the distributions of present and projected populations, including both resident and transient populations, in the vicinity of the site, and the information shall be kept up to date over the

lifetime of the research reactor. The population distribution should be used in the evaluation for the site of the possible impacts on the public of any releases of radioactive material.

Uses of land and water in the region

5.38. The uses of land and water bodies in the region shall be identified so as to assess the possible regional effects of the proposed research reactor, and in particular for the purpose of preparing emergency plans. The assessment should include land and water bodies that the population may use or which may serve as a habitat for organisms in food chains.

Ambient radioactivity

5.39. Before the commissioning of the research reactor, the ambient radioactivity of the atmosphere, hydrosphere, lithosphere and biota shall be determined for the vicinity of the site, as necessary, to permit the subsequent evaluation of the effects of the research reactor on radioactivity in the environment.

MONITORING OF HAZARDS

5.40. The characteristics of natural and human induced hazards as well as the demographic, meteorological and hydrological conditions of relevance to the research reactor shall be monitored throughout its lifetime, commencing no later than the start of construction and continuing through to decommissioning.

6. DESIGN

PHILOSOPHY OF DESIGN

General

6.1. The research reactor shall be designed in such a way that the safety objectives (see para. 2.2) are achieved. The general design requirements in this section shall be applied in the design of all types of research reactor.

Additionally, a set of specific design requirements shall be applied as appropriate to the design of SSCs for particular reactor types.

6.2. Application of these requirements is an interactive process and shall be implemented in all phases of design, with account taken of the results of an accompanying safety analysis (see also paras 2.7 and 6.72–6.78).

6.3. The reactor designer shall consider not only the reactor itself but also any associated facilities that may affect safety. In addition, the reactor designer shall also consider the effects of the reactor as designed on the associated facilities and the implications of the design in all the stages of the reactor's lifetime (e.g. in terms of service conditions, electromagnetic fields and other interferences).

6.4. The achievement of a safe design requires that a close liaison be maintained between the reactor designer and the operating organization. The designer shall arrange for the orderly preparation, presentation and submission of design documents to the operating organization for use in the preparation of the SAR. The design should be developed in parallel with the development of the SAR (see paras 3.6–3.10).

6.5. The mode of operation (e.g. operation on demand rather than continuous operation, operation at different power levels, operation with different core configurations and operation with different nuclear fuels) and the stability of the reactor at different levels of operating power should be given due consideration in the design of the safety systems.

Defence in depth

6.6. The defence in depth concept (see paras 2.5–2.7) shall be applied in the design to provide graded ('enveloped') protection against various reactor transients, including transients resulting from equipment failure and human error and from internal or external events that could lead to a DBA. In particular, the following aspects shall be considered in the design:

(a) The use of conservative design margins, the implementation of a quality assurance programme (see footnote 14) and the organization of surveillance activities.
(b) The provision of successive physical barriers to the release of radioactive material from the reactor. Examples of such barriers are the fuel matrix, the fuel cladding, the primary heat transport system, the pool and the

reactor building. Also, provision, as appropriate, for ensuring the effectiveness of these barriers, and for their surveillance and protection.

(c) Application of the single failure criterion by ensuring the fulfilment of each of the following basic safety functions:
   — shutting down the reactor and maintaining it in a safe shutdown state for all operational states or DBAs;
   — providing for adequate removal of heat after shutdown, in particular from the core (see para. 6.131), including in DBAs;
   — confining radioactive material in order to prevent or mitigate its unplanned release to the environment.

(d) The use of on-site and off-site emergency plans aimed at mitigating the consequences for the public and the environment in the event of a substantial release of radioactive effluents to the environment.[17]

6.7. Application of the defence in depth concept requires the inclusion of equipment, consisting of safety systems and safety related items or systems, and procedures to prevent and control deviations from operational states and to prevent and mitigate accident conditions, or to ensure appropriate protection in the event that prevention fails. This equipment, and in particular the equipment used to implement levels 2–4 of para. 2.6, which usually consists of safety systems and engineered safety features, shall be subject to special design requirements.

6.8. The three basic safety functions mentioned in para. 6.6(c) — essentially, shutting down the reactor, cooling, in particular the reactor core, and confining radioactive material — shall be met by incorporating into the design an appropriate combination of inherent and passive safety features, safety systems and engineered safety features, and by applying administrative procedures over the lifetime of the reactor. An example of an inherent safety feature is the appropriate choice of materials and geometries to provide prompt negative coefficients of reactivity.

[17] The implementation of an emergency response plan may require the designer to make appropriate design provisions (see paras 6.30 and 6.31).

Safety functions

6.9. Safety functions are the essential characteristic functions associated with SSCs that ensure the safety of the reactor, as mentioned in para. 6.6(c). Safety functions shall be appropriate for the particular reactor design. In normal operation, the equipment needed to perform safety functions will be the operating systems. In general, these systems will have to be supplemented by other engineered safety features to perform their functions for anticipated operational occurrences and in DBAs.

6.10. In the design of the safety systems, including engineered safety features, that are used to achieve the three basic safety functions — shutting down the reactor, cooling, in particular the reactor core, and confining radioactive material — the single failure criterion shall be applied, high reliability shall be ensured and provisions shall be included to facilitate regular inspection, testing and maintenance.

Acceptance criteria and design rules

6.11. In accordance with para. 3.13, acceptance criteria shall be established for operational states and for DBAs. In particular, the DBAs considered in the design of the research reactor and selected BDBAs shall be identified for the purposes of establishing acceptance criteria. For the design of SSCs, acceptance criteria may be used in the form of engineering design rules. These rules may include requirements in relevant codes and standards established in the State or internationally. The regulatory body shall review the acceptance criteria.

GENERAL REQUIREMENTS FOR DESIGN

Classification of SSCs[18]

6.12. SSCs and software for instrumentation and control that are important to safety shall be first specified and then classified according to their function and

[18] This classification reflects the significance for nuclear safety of the SSCs. Its purpose is to establish a gradation in the application of the requirements for design and the requirements for quality assurance. There are other possible classifications or categorizations of SSCs according to other aspects (e.g. the seismic categorization of SSCs).

significance for safety. The basis of the safety classification of the SSCs, including software, shall be stated and the design requirements shall be applied in accordance with their safety classification.

6.13. The method for classifying the safety significance of SSCs, including software, shall be based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgement, in which account is taken of their safety function and the consequences of failure to perform their functions. Appropriate design interfaces between SSCs of different classes shall be provided to ensure that the failure of any item of a lower safety class will not cause the failure of an item of a higher safety class.

Codes and standards

6.14. Codes and standards applicable to SSCs shall be identified and their use shall be in accordance with their classification (see paras 6.12 and 6.13). In particular, if different codes and standards are used for different types of items (e.g. for piping and for electrical systems), consistency between them shall be demonstrated.

6.15. In the case of SSCs for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment may be applied, or, in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these may be applied, and this results based approach shall be justified.

Design basis

6.16. All the challenges that the reactor may be expected to face during its operational lifetime shall be taken into consideration in the design process. These challenges include all the foreseeable conditions and events relating to stages in the operational lifetime of the reactor and to operational states and accident conditions, site characteristics, design requirements and the limits of parameters, modes of operation and so on. The demands imposed on the design of the reactor by these challenges and conditions shall determine the design basis of the research reactor facility. The capabilities that the research reactor facility will need in order to withstand these challenges without authorized limits being exceeded shall be specified in the design basis.

Postulated initiating events and DBAs

6.17. Challenges may occur at all levels of defence in depth. This possibility shall be recognized in the design and design measures shall be provided to ensure that the safety functions are achieved and the safety objectives can be met. These challenges to defence in depth will stem from postulated initiating events. Postulated initiating events shall be selected appropriately for the purpose of analysis (see the Appendix). It shall be shown that the set of postulated initiating events selected covers all credible accidents that may affect the safety of the research reactor. In particular, the DBAs shall be identified.

Site related characteristics

6.18. The various possible interactions between the research reactor facility and the environment shall be considered in the design, including aspects relating to the population, meteorology, hydrology, geology and seismology. Off-site services upon which the safety of the facility and the protection of the public depend, such as communications, electrical and water supplies and fire and police services, shall be taken into account.

Internal events

6.19. An analysis of the postulated initiating events shall be made to establish all those internal events that could affect the safety of the research reactor facility. These events may include equipment failures or malfunctions.

6.20. The potential for internal hazards such as fire, flooding, missile generation, pipe whip, jet impact or the release of fluid from failed systems or from other installations on the site shall be taken into account in the design of the research reactor facility. Appropriate preventive and mitigatory measures shall be taken to ensure that nuclear safety is not compromised. Some external events could initiate internal fires or floods or lead to the generation of missiles. Such interrelation of external and internal events shall also be considered in the design, where appropriate.

External events

6.21. The design basis for natural and human induced external events shall be determined. The events to be considered shall include those that have been identified in the site evaluation (see Section 5). Consideration shall also be

given to earthquake hazards (see paras 5.15, 5.16 and 6.17), including the possibility of equipping the research reactor facility with seismic detection systems that actuate the automatic shutdown systems of the reactor if a specified threshold value is exceeded.

Fires and explosions

6.22. SSCs important to safety shall be designed and located, subject to compliance with other safety requirements, so as to minimize the effects of fires and explosions. A fire hazard analysis and an explosion hazard analysis shall be carried out for the research reactor facility to determine the necessary ratings of the fire barriers and means of passive protection and physical separation against fires and explosions. The design shall include provisions to prevent or limit the formation of explosive atmospheres. Fire detection systems and fire fighting systems of the necessary capability shall be provided.

6.23. Fire fighting systems shall be automatically initiated where necessary. Fire fighting systems shall be designed and located so as to ensure that their rupture or spurious or inadvertent operation would not significantly impair the capability of SSCs important to safety, and would not simultaneously affect redundant safety groups and thereby render ineffective the measures taken to comply with the single failure criterion (see paras 6.36–6.38).

6.24. Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the research reactor facility, in particular in locations such as the reactor building and the control room. Flammable gases and liquids and combustible materials that could produce or contribute to explosive mixtures shall be kept to minimum necessary amounts and shall be stored in adequate facilities to keep reacting substances segregated.

6.25. The capability shall be maintained for shutting down the reactor, removing residual heat, confining radioactive material and monitoring the status of the facility. These capabilities shall be maintained by means of the appropriate incorporation of redundant parts, diverse systems, physical separation and design for fail-safe operation such that the following objectives are achieved:

(a) To prevent fires and explosions;
(b) To detect and extinguish quickly those fires that do start, thus limiting the damage caused;

(c) To prevent the spread of those fires that are not extinguished, and of fire induced explosions, thus minimizing their effects on the performance of essential functions of the facility.

Design limits of parameters

6.26. Design limits for all relevant parameters shall be specified for each operational state of the reactor and for DBAs.

6.27. A comparison of event sequences shall be performed to identify the most challenging parameter values. The resulting limiting parameter values, with a reasonable margin, shall be used in the design of individual systems and components, including experimental devices.

Design for operational states

6.28. The research reactor shall be designed to operate safely within predefined ranges of values for various parameters, and subject to requirements and constraints in all operational states, while meeting the radiation protection objective. The requirements relating to the anticipated utilization of the reactor, including the requirements for power stability, shall be taken into account in the design. The design shall be such that the response of the reactor and its associated systems to a wide range of events, including anticipated operational occurrences, will allow its safe operation or power reduction, if necessary, without the need to invoke provisions beyond the first, or at the most the second, level of defence in depth.

6.29. The requirements and limitations set out in para. 6.28 shall form the basis for the OLCs. The design shall be such as to facilitate the setting of a practicable set of OLCs for reactor operation.

Design for accident conditions

6.30. Where prompt reliable action is required in response to postulated initiating events, the design of the reactor shall include means of automatically initiating the operation of the necessary safety systems. It may be necessary following DBAs in some cases for the operator to place the reactor in a stable long term state and to take actions to limit the release of radioactive material. The design should be such as to reduce demands on the operator as far as practicable, in particular during and following a DBA.

6.31. The items important to safety shall be designed to withstand the effects of extreme loading and environmental conditions (e.g. extremes of temperature, humidity, radiation levels) arising from DBAs. The stable long term shutdown condition following an accident can differ from the initial shutdown condition. The design shall incorporate provisions, including a negative power coefficient, for bringing the reactor into a stable long term condition.

Engineered safety features

6.32. Engineered safety features are safety systems that are provided mainly to limit or to mitigate the consequences of anticipated operational occurrences and DBAs. Examples of engineered safety features are an emergency core cooling system and means of confinement (in particular, an emergency ventilation system). Specific requirements for these systems and their supplementary features are established in paras 6.115–6.130. Other engineered safety features, such as a second shutdown system, a containment structure or other systems, shall also be designed in accordance with these requirements.

6.33. The necessity for engineered safety features shall be determined from the safety analysis. The accidents with which these systems must be able to cope shall be specified and analyses shall be provided to demonstrate that the systems fulfil the requirements. Those systems and subsystems that are essential for the proper operation of the engineered safety features shall be provided (e.g. the emergency electrical power supply for the emergency core cooling system).

6.34. The design basis and the various modes of operation of an engineered safety feature shall be determined in detail, including the extent to which the engineered safety feature is automated and the conditions for which its manual overriding is warranted. The following shall be considered in the design of engineered safety features:

(a) Component reliability, system independence, redundancy, fail-safe characteristics, diversity and physical separation of redundant systems;
(b) The use of material to withstand the postulated DBAs (e.g. in relation to radiation levels or radiolytic decomposition);
(c) Provisions for inspection, periodic testing and maintenance (including under simulated DBA conditions where possible) to verify that the engineered safety features continue to function or are in a state of readiness to perform their functions and will be reliable and effective upon demand.

Design for reliability

6.35. Maximum authorized unavailability limits for operation of the research reactor shall be established for certain safety systems or components to ensure the required reliability in the performance of safety functions. The following measures shall be used, if necessary in combination, to achieve and maintain the required reliability, in accordance with the importance of the safety functions to be performed by the SSCs. Consideration shall be given to software systems as well as to hardware systems.

Redundancy and the single failure criterion

6.36. The principle of redundancy shall be applied as an important design principle for improving the reliability of systems important to safety. The design shall be such as to ensure, on the basis of analysis, that no single failure could result in a loss of the capability of a system to perform its intended safety function.

6.37. Multiple sets of equipment that cannot be tested individually shall not be considered redundant.

6.38. The degree of redundancy adopted shall reflect the potential for undetected failures that could degrade reliability. Possible failures shall be considered undetectable if there is no test or method of inspection by which they could be found. For undetected failures, either the failure shall be considered to occur at any time or other methods shall be applied, such as the surveillance of reference items, validated methods of calculation and the use of conservative safety margins[19].

Diversity

6.39. Diversity is applied to redundant systems or components that perform the same safety function by incorporating into the systems or components different attributes, such as:

(a) Different principles of operation;
(b) Different operating conditions;
(c) Production by different manufacturers.

[19] The safety margin is the difference between the safety limit and the operational limit. It is sometimes expressed as the ratio of these two values.

6.40. The principle of diversity can be applied to enhance reliability and to reduce the potential for common cause failures. The principle of diversity shall be adopted wherever practicable, after consideration of its possible disadvantages in terms of complications in operating, maintaining and testing the diverse equipment.

Independence

6.41. The principle of independence (e.g. functional isolation and physical separation by means of distance, barriers or a special layout for reactor components) shall be applied, as appropriate, to enhance the reliability of systems, in particular with respect to common cause failures.

Fail-safe design

6.42. The principle of fail-safe design shall be considered and shall be adopted in the design of systems and components important to safety, as appropriate: systems at research reactor facilities shall be designed to pass into a safe state, with no necessity for any action to be initiated, if a system or component fails.

Ease of testing and maintenance

6.43. Reactor items important to safety shall be designed and arranged so that they can be adequately inspected, tested and maintained as appropriate, before commissioning and at regular intervals thereafter, in accordance with their importance to safety. The layout of the reactor shall be such that these activities are facilitated and can be performed without undue exposure to radiation of the operating personnel. If it is not practicable to provide adequate accessibility of a component for testing, the possibility of its undetected failure shall be taken into account in the safety analysis.

Design for commissioning

6.44. The design shall include design features as necessary to facilitate the commissioning process for the reactor. These design features may include provisions to operate with transition cores of different geometries, which may need forced circulation cooling.

Provision for inspection, testing and maintenance

6.45. The design of the reactor shall be such as to allow for appropriate functional testing and inspection of items important to safety to ensure that systems will perform their safety functions with the required reliability. This is particularly important for passive components and for systems whose ability to function is not normally verified by routine operations. Important factors that shall be considered are the ease of performing the tests and inspections, the degree to which the tests and inspections represent real conditions, and the need to maintain the performance of the safety function during the tests. Where possible and appropriate, self-testing circuits should be installed in electrical and electronic systems.

6.46. Provisions for appropriate accessibility, shielding, remote handling, post-irradiation radiation levels and decontamination shall be made in the design to keep radiation doses and uptakes of radioactive material as low as reasonably achievable during maintenance. Materials shall be selected to minimize activation levels in items exposed to high neutron fluxes.

6.47. Provision shall be made in the design of the reactor to facilitate routine in-service inspection with the aid of appropriate non-destructive testing techniques for determining the conditions of SSCs subject to corrosion, erosion, fatigue or other ageing effects.

Design for emergency planning[20]

6.48. The inclusion of specific design features for facilitating emergency planning shall be considered, depending on the potential hazard deriving from the reactor. The need for such design features may be determined by means of analyses of BDBAs. Acceptable measures shall be based where possible on realistic or best estimate assumptions, methods and analytical criteria. They need not necessarily involve the use of conservative engineering practices. The research reactor facility shall be provided with a sufficient number of safe escape routes, clearly and durably marked, with reliable emergency lighting, ventilation and other building services essential to their safe use. The escape routes shall meet the relevant international requirements for radiation zoning

[20] For further discussion of the conduct of sequences in safety analysis, see paras 7.72–7.78.

and fire protection and the relevant national requirements for industrial safety and physical protection of the facility.

6.49. Suitable alarm systems and means of communication shall be provided so that all persons present at the research reactor facility and on the site can be warned and instructed, even under accident conditions. The availability of the means of communication necessary for safety within the research reactor facility shall be ensured at all times. Means of communication shall be available in the control room and also in the supplementary control room if there is one[21]. This requirement shall be taken into account in the design and in the diversity of the means of communication selected for use.

Design for decommissioning

6.50. In the design of the reactor and its experimental devices, consideration shall be given to facilitating its ultimate decommissioning. In this connection, attention shall be directed to keeping the radiation exposure of personnel and of the public during decommissioning as low as reasonably achievable and to ensuring adequate protection of the environment from undue radioactive contamination. In accomplishing this in the design, the following points shall be considered:

(a) The selection of materials to minimize activation and to provide for easy decontamination;
(b) Optimization of the facility's layout and access routes to facilitate the removal of large components and the detachment and handling (remotely where required) of activated components;
(c) The processing and storage of radioactive waste.

6.51. In addition, full details shall be retained of the design requirements and of information relating to the site and its final design and construction, such as the 'baseline' background radiological characterization as-built drawings relating to the facility's layout, piping and cable penetrations, as necessary information for decommissioning. Further guidance supporting these requirements is provided in Ref. [16].

[21] For further discussion of the supplementary control room, see para. 6.144.

Design for radiation protection

6.52. For all operational states and DBAs, adequate provision shall be made in the design, on the basis of a consistent radiation protection programme and in accordance with the radiation protection objective (see para. 205 of Ref. [1], quoted in para. 2.2), for shielding, ventilation, filtration and decay systems for radioactive material (such as delay tanks), and for monitoring instrumentation for radiation and airborne radioactive material inside and outside the controlled area.

6.53. The dose values used for design purposes shall be set with a sufficient margin to ensure that the authorized limits will not be exceeded. The shielding, ventilation, filtration and decay systems of the reactor and its associated facilities shall be designed to allow for uncertainties in operating practices and in all operational states and DBAs.

6.54. Structural materials (such as core supports, grids and guide tubes), in particular those used near the core, shall be carefully chosen to limit the dose to personnel during operation, inspection, testing and maintenance, and decommissioning, as well as to fulfil their other functions. The effects of radionuclides (e.g. ^{16}N, ^{3}H, ^{41}Ar, ^{24}Na and ^{60}Co) produced by neutron activation in reactor process systems shall be given due consideration in the provision of radiation protection for people on and off the site.

6.55. The design shall include any necessary provisions to segregate materials according to their radiological, physical and chemical characteristics, to facilitate their handling and to protect workers and the public by means of access control. This shall be accomplished by establishing zones within the facility (in supervised and controlled areas) (see Glossary) that are classified according to their potential for hazard. Zones shall be clearly delineated and designated. Where necessary, surfaces shall be appropriately designed to facilitate decontamination.

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

6.57. Ventilation systems with appropriate filtration shall be provided for use in operational states and DBAs. For many research reactors filtered ventilation systems are essential to fulfil the function of confining radioactive material (see paras 6.120–6.130).

6.58. Protection and safety shall be optimized by means of suitable provision in the design and layout of the reactor and its experimental devices and facilities to limit exposure and contamination from all sources. Such provision shall include the adequate design of SSCs to limit exposure during inspection, testing and maintenance, to provide shielding from direct and scattered radiation, and to provide means of monitoring and controlling access to the reactor and its experimental devices and facilities.

6.59. Provision shall be made in the design for handling the radioactive waste generated by the research reactor. Provision shall be made for appropriate decontamination facilities for both personnel and equipment and for handling the radioactive waste arising from decontamination activities.

Design for physical protection

6.60. Provision shall be made in the design to prevent any unauthorized entry to the site or to buildings on the site, for the main purposes of preventing the theft or unauthorized removal of nuclear material and sabotage.

Human factors and ergonomic considerations

6.61. Human factors are an important aspect in the safety of research reactors as the state of the reactor changes frequently and the operator has easy access to the reactor core and to experiments. Human factors and human–machine interfaces shall be given systematic consideration at an early stage of the design and throughout the entire design process.

6.62. Because of the flexibility required in operating a research reactor, it may be necessary to rely for safety in certain activities on administrative controls and procedures. Special consideration shall be given in design to ensuring that, if reliance on administrative controls and procedures is necessary, such controls are feasible. Administrative procedures may include operating rules in the form of OLCs, which are derived from the design of the reactor and the safety analysis.

6.63. Special consideration shall be given to human factors and the application of ergonomic principles in the design of the control room and reactor systems as appropriate. The operator shall be provided with clear displays and audible signals for those parameters that are important to safety. Safety actions shall be automated so that immediate operator action is not required. The design shall be such as to minimize the demands on the operator so as to reduce the burden on the operator and reduce the scope for human error. The need for interlocks and hierarchical access controls (e.g. keys and passwords) shall be taken into consideration in the design in the light of such human factors.

6.64. With regard to the presentation of information visually and on instruments and alarms, the design shall be such as to promote the success of operator actions under the constraints of the time available, the physical environmental conditions expected and the possible psychological pressure on the operator.

Provision for utilization and modification

6.65. Research reactors are flexible in nature and they may be in various different states. Special precautions shall be taken in the design in relation to the utilization and modification of the research reactor to ensure that the configuration of the reactor is known at all times. In particular, special consideration shall be given to experimental equipment since:

(a) It can cause hazards directly if it fails;
(b) It can cause hazards indirectly by affecting the safe operation of the reactor;
(c) It can increase the hazard due to an initiating event by its consequent failure and the effects of this on the event sequence.

6.66. Every proposed modification to an experiment or to a reactor that may have a major significance for safety shall be designed in accordance with the same principles as apply for the reactor itself (see para. 7.88 and Ref. [15]). In particular, all experimental devices shall be designed to standards equivalent to those applied for the reactor itself and shall be fully compatible in terms of the materials used, the structural integrity and the provision for radiation protection. The radioactive inventory and the generation and release of energy shall be considered in the design of all experimental devices.

6.67. Where experimental devices penetrate the reactor boundaries, they shall be designed to preserve the means of confinement and shielding of the reactor.

Protection systems for experimental devices shall be designed to protect both the device and the reactor.

Selection and ageing of materials

6.68. At the design stage, an appropriate safety margin shall be adopted to allow for the anticipated properties of materials at the end of their useful lifetime. Where no data are available on materials, a suitable programme of inspection and periodic testing of materials shall be put in place and the results that are obtained in this programme shall be used in reviewing the adequacy of the design at appropriate intervals. This may require provisions in the design for monitoring materials whose mechanical properties may change in service owing to such factors as stress corrosion or radiation induced changes. Improved safety factors may be achieved by the selection of materials of high strength or high melting point.

6.69. To ensure the capability of all items important to safety to perform their safety functions, appropriate margins shall be provided in the design to take account of relevant ageing effects and potential ageing related degradation. Ageing effects shall be taken into account for all operational states, including periods of maintenance and shutdown.

6.70. Provisions shall also be made for the necessary monitoring, testing, sampling and inspection for the detection, assessment, prevention and mitigation of ageing effects.

Provision for extended shutdown

6.71. Many research reactors are shut down for extended periods for various purposes, such as for modifications or for preparing for decommissioning. Provision shall be made in the design to meet the needs arising in long shutdown periods, such as the needs for maintaining the conditions of the nuclear fuel, the coolant or the moderator, for the inspection, periodic testing and maintenance of the relevant SSCs of the facility, and for providing physical protection. Special consideration shall be given to long lived neutron poisons, which may affect the restarting of the reactor.

Safety analysis

6.72. A safety analysis shall be conducted of the design of the research reactor. The safety analysis shall include analyses of the response of the reactor to a

range of postulated initiating events (such as malfunctions or failures of equipment, operator errors or external events) that could lead either to anticipated operational occurrences or to accident conditions (see also Ref. [7]). These analyses shall be used as the basis for the design of items important to safety and the selection of the OLCs for the reactor. The analyses shall also be used as appropriate in the development of operating procedures, periodic testing and inspection programmes, record keeping practices, maintenance schedules, proposals for modifications and emergency planning.

6.73. The scope of the safety analysis shall include:

(a) Characterization of the postulated initiating events that are appropriate;
(b) Analysis of event sequences and evaluation of the consequences of the postulated initiating events;
(c) Comparison of the results of the analysis with radiological acceptance criteria and design limits;
(d) Demonstration that the management of anticipated operational occurrences and DBAs is possible by means of an automatic response of safety systems in combination with prescribed operator actions;
(e) Determination of the OLCs for normal operation;
(f) The analysis of safety systems and the engineered safety features;
(g) The analysis of the means of confinement.

6.74. For each postulated initiating event, qualitative and quantitative information about the following aspects shall be considered in the evaluation:

(a) The input parameters, initial conditions, boundary conditions, assumptions, models and codes used;
(b) The sequence of events and the performance of reactor systems;
(c) The sensitivity to single failure modes and common cause failures;
(d) The sensitivity to human factors;
(e) Analysis of transients;
(f) The identification of damage states;
(g) The potential for releases of fission products and radiation exposures;
(h) The derivation of source terms;
(i) The evaluation of radiological consequences.

6.75. For each accident sequence considered, the extent to which the safety systems and any operable process systems are required to function under DBA conditions shall be indicated. These events are usually evaluated by deterministic methods. Probabilistic techniques can be used to complement the

evaluation. The results of these complementary analyses provide input to the design of the safety systems and the definition of their functions.

6.76. Where applicable, the analysis shall include consideration of the experimental devices with regard to both their own safety aspects and their effects on the reactor (see Ref. [15]).

6.77. The applicability of the methods of analysis shall be verified.

6.78. The results of the safety analysis of the reactor, including the effects of anticipated process disturbances and postulated component failures and human errors (postulated initiating events) and their consequences, shall be reflected in the SAR to evaluate the capability of the reactor to control or to accommodate such situations and failures.

SPECIFIC REQUIREMENTS FOR DESIGN

Reactor core and reactivity control system

Reactor core and fuel design

6.79. Appropriate neutronic, thermal-hydraulic, mechanical, material, chemical and irradiation related considerations associated with the reactor as a whole shall be taken into account in the design of fuel elements and assemblies, the reflectors and other core components.

6.80. Analyses shall be performed to show that the intended irradiation conditions and limits (such as fission density, total fissions at the end of lifetime and neutron fluence) are acceptable and will not lead to undue deformation or swelling of the fuel elements. The anticipated upper limit of possible deformation shall be evaluated. These analyses shall be supported by data from experiments and from experience with irradiation. Consideration should be given in the design of the fuel elements to the requirements relating to the long term management of irradiated elements.

6.81. All foreseeable reactor core configurations from the initial core through to the equilibrium core for various appropriate operating schedules shall be considered in the core design.

6.82. The reactor core (i.e. the fuel elements, reflectors, cooling channel geometry, irradiation devices and structural parts) shall be designed to maintain the relevant parameters within specified limits in all operational states. There shall be provisions in the design to monitor the integrity of the fuel. In the event of the detection of fuel failure, an investigation shall be conducted to identify the failed fuel element. Authorized limits shall not be exceeded (see also paras 7.96–7.102) and if necessary the reactor shall be shut down and the failed fuel element shall be unloaded from the core.

6.83. The reactor core shall be designed so that fuel damage in DBAs would be kept within acceptable limits.

6.84. The reactor core, including fuel elements, reactivity control mechanisms[22] and experimental devices, shall be designed and constructed so that the permissible design limits that are specified for all operational states are not exceeded. A suitable margin, including margins for uncertainties and engineering tolerances, shall be incorporated in setting these limits.

6.85. The reactor core shall be designed so that the reactor can be shut down, cooled and held subcritical with an adequate margin for all operational states and for DBAs. The state of the reactor shall be assessed for selected BDBAs.

6.86. Wherever possible, the design of the reactor core should make use of inherent safety characteristics to minimize the consequences of accident conditions (those that are produced by transients and instabilities).

Reactivity control system

6.87. Sufficient negative reactivity shall be available in the reactivity control devices(s) in order that the reactor can be brought into a subcritical condition and maintained subcritical in all operational states and in DBA conditions, with account taken of the experimental arrangements with the highest positive reactivity contribution. In the design of reactivity control devices, account shall be taken of wear-out and the effects of irradiation, such as burnup, changes in physical properties and the production of gas.

[22] Reactivity control mechanisms are devices of all kinds for controlling the reactivity, including regulating rods, control rods, shutdown rods or blades, and devices for controlling the moderator level.

6.88. The maximum rate of addition of positive reactivity allowed by the reactivity control system or by an experiment shall be specified and shall be limited to values justified in the SAR.

6.89. It shall be demonstrated in the design that the reactivity control system will function properly under all operational states of the reactor and will maintain its reactor shutdown capability under all DBAs also, including failures of the control system itself.

Reactor shutdown system

6.90. At least one automatic shutdown system shall be incorporated into the design. The provision of a second independent shutdown system may be necessary, depending on the characteristics of the reactor, and this shall be given due consideration.

6.91. The effectiveness, speed of action and shutdown margin[23] of the reactor shutdown system shall be such that the specified limits and conditions are met.

6.92. No single failure in the shutdown system shall be capable of preventing the system from fulfilling its safety function when required (e.g. with the most reactive shutdown rod stuck in the out position).

6.93. One or more manual initiations suitable for emergency shutdown may be necessary and this shall be given due consideration.

6.94. Instrumentation shall be provided and tests shall be specified to be performed to ensure that the means of shutdown are always in the state stipulated for the given condition of the reactor. For computer based digital reactivity control systems, verification and validation of software shall be performed.

[23] The shutdown margin is the negative reactivity provided in addition to the negative reactivity necessary to maintain the reactor in a subcritical condition without time limit, with the most reactive control device removed from the core and with all experiments that can be moved or changed during operation in their most reactive condition.

Reactor protection system

6.95. The reactor protection system shall be automatic and independent of other systems. In addition, a manual reactor trip signal shall be provided as an input to the reactor protection system.

6.96. The reactor protection system shall be capable of automatically initiating the required protective actions for the full range of postulated initiating events to terminate the event safely. The possible malfunction (single failure) of parts of the system should be taken into account in providing this capability. In some cases, manual operator action may be considered to be sufficiently reliable provided that:

(a) Adequate time is available;
(b) Information is suitably processed and presented;
(c) Diagnosis is simple and action is clearly defined;
(d) The demands imposed on the operator are not excessive.

6.97. Consideration shall be given to the provision of the capability to initiate reactor shutdown from a remote location.

6.98. The reactor protection system shall be designed in such a way that necessary automatic actions, once initiated, cannot be impeded or prevented by manual actions and that no manual actions are necessary within a short period of time following an accident. Protective actions, once initiated automatically by the reactor protection system, shall be designed to proceed to completion. Such automatic actions by the reactor protection system shall not be self-resetting and a return to operation shall require deliberate operator action.

6.99. The possibility of bypassing interlocks and trips of the reactor protection system shall be carefully evaluated and appropriate means of protecting interlocks and trips that are important to safety from being inadvertently bypassed shall be incorporated into the reactor protection system.

6.100. The design of the reactor protection system shall employ redundancy and independence sufficient to ensure that no single failure could result in the loss of automatic protective actions. Design techniques such as the use of fail-safe behaviour and diversity shall be used to the extent practicable to prevent the loss of the reactor protection function. The appropriate protective actions shall be designed to be initiated automatically.

6.101. The reactor protection system shall be designed to bring the reactor into a safe condition and to maintain it in a safe condition even if the reactor protection system is subjected to a feasible common cause failure (e.g. hardware failure or failure due to ageing or human factors).

6.102. All components of the reactor protection system shall be capable of being functionally tested.

6.103. It shall be ensured in the design that the set points can be established with a margin between the initiation point and the safety limits such that the action initiated by the reactor protection system will be able to control the process before the safety limit is reached. Some of the factors in establishing this margin are:

(a) The accuracy of the instrumentation;
(b) Uncertainties in calibration;
(c) Instrument drift;
(d) Instrument and system response times.

6.104. Where a computer based system is intended to be used in a reactor protection system, the following requirements shall apply in addition to those of paras 6.138–6.140:

(a) Hardware and software of high quality and best practices shall be used;
(b) The whole development process, including control, testing and commissioning of the design changes, shall be systematically documented and reviewable;
(c) To confirm the reliability of the computer based systems, an assessment of the computer based systems shall be undertaken by expert personnel who are independent of the designers and the suppliers.

6.105. Where the necessary integrity of a computer based system that is intended for use in a reactor protection system cannot be demonstrated with a high level of confidence, diverse means of ensuring fulfilment of the protection functions (e.g. hard wired systems) shall be provided.

Reactor coolant system and related systems

Reactor coolant system

6.106. The reactor coolant system shall be designed to provide adequate cooling to the reactor core with an acceptable and demonstrated margin.

6.107. Systems containing reactor coolant shall be designed to allow tests and inspections so that the possible occurrence of leaks, fast growing cracks and brittle fractures could be detected. Consideration shall be given in the design to obtaining characteristics that ensure the slow propagation of any flaw. A multiple barrier concept may be adopted as appropriate (e.g. the primary cooling system may be fully contained within the pool block or in a special design to cope with possible breaches).

6.108. In the design of water cooled reactors particular attention shall be paid to preventing the uncovering of the core. Special features, such as penetrations over the core, whenever feasible, siphon breaks and suitable isolation devices shall be used. High quality design and fabrication together with the characteristics of ease of inspection and testing and redundancy, where appropriate, shall be ensured.

6.109. The reactor coolant boundary shall be designed to facilitate pre-service and in-service inspection and testing.

6.110. Where a separate system is required for cooling the core after shutdown, an adequate and reliable system, in addition to the primary cooling system, shall be provided for the removal of residual heat.

6.111. For reactor systems that use flappers[24] or equivalent systems for natural circulation cooling, and for which this mode is part of the safety system (or is considered an engineered safety feature), an appropriate number of redundant devices shall be used (in application of the single failure criterion), including devices to verify the functioning and to provide signals to the reactor protection system.

[24] A flapper is a passive valve that opens when the flow is below a set value to allow for the creation of a natural circulation circuit on the loss of forced flow.

6.112. The reactor coolant system shall provide long term, reliable heat transfer from the fuel to the ultimate heat sink.

6.113. If two fluid systems that are operating at different pressures are interconnected, either the systems shall both be designed to withstand the higher pressure, or provision shall be made to preclude the design pressure of the system operating at the lower pressure from being exceeded, on the assumption that a single failure occurs.

6.114. Provision shall be made to monitor and control the properties (e.g. the pH and conductivity of the water) of the reactor coolant and/or the moderator, and to remove radioactive substances, including fission products, from the coolant.

Emergency core cooling system

6.115. Where required, an emergency core cooling system shall be provided to prevent damage to the fuel in the event of a loss of coolant accident. The accidents with which the system must cope shall be identified and analyses shall be performed to show that the system fulfils the requirements.

6.116. The emergency core cooling system shall be capable of keeping core temperatures within specified safety limits for a sufficient period of time.

6.117. The emergency core cooling system shall be capable of preventing significant failure of fuel for the range of loss of coolant accidents specified in the design basis (i.e. under DBAs, damage to the fuel and the releases of radioactive material shall be kept within authorized limits). Special procedures for cooling the core shall be considered in the case of selected BDBAs.

6.118. The emergency core cooling system shall be designed with sufficient reliability to meet the requirements of paras 6.35–6.43. The system shall be designed to perform its intended function in the event of any single failure in the system.

6.119. The emergency core cooling system shall be designed to permit the periodic inspection of components and shall be designed for appropriate periodic functional testing for the verification of performance.

Means of confinement

6.120. Where required, means of confinement[25] shall be designed to ensure that a release of radioactive material (fission products and activation products) following an accident involving disruption of the core does not exceed acceptable limits. The means of confinement may include physical barriers surrounding the main parts of the research reactor that contain radioactive material. Such barriers shall be designed to prevent or mitigate an unplanned release of radioactive material in operational states or in DBAs. The barriers for confinement usually comprise the reactor building together with other items. The other items may be: sumps and tanks for collecting and containing spills; an emergency ventilation system, usually with filtration; isolation devices on barrier penetrations; and a point of release which is usually elevated.

6.121. The means of confinement shall be designed for sufficient reliability to meet the requirements established in paras 6.32–6.34.

6.122. For the proper functioning of the means of confinement, the pressure within a barrier shall be set at such a level as to prevent the uncontrolled release of radioactive material to the environment from the barrier. In setting this pressure, variations in atmospheric conditions (e.g. wind speed and atmospheric pressure) shall be taken into account.

6.123. In the design of the means of confinement, the effects of extreme conditions (e.g. explosions within the barrier) and environmental conditions

[25] Confinement is the function of containing radioactive material within a nuclear reactor so as to prevent or mitigate its unplanned release. Confinement is a basic safety function that is required to be fulfilled in normal operational modes, for anticipated operational occurrences, in design basis accidents and, to the extent practicable, in selected beyond design basis accidents (see Ref. [22], para. 4.6). The function of confinement is usually fulfilled by means of several barriers surrounding the main parts of a nuclear reactor that contain radioactive material (see paras 2.19, 6.6). For a research reactor, the reactor building is the ultimate barrier for ensuring confinement. Consideration may be given to the use of other structures (e.g. the reactor block in a fully enclosed research reactor) for providing confinement where this is technically feasible. For most designs of large nuclear reactor, a strong structure housing the reactor is the ultimate barrier providing confinement. Such a structure is called the containment structure or simply the containment. The containment also protects the reactor against external events and provides radiation shielding in operational states and in accident conditions.

due to accidents, including conditions arising from the external and internal events listed in the Appendix, as relevant (e.g. fire conditions and the associated increases in local pressures), shall be taken into account, in accordance with the design basis.

6.124. The barriers shall be designed with suitable margins for the highest calculated pressure and temperature loads expected in DBA conditions.

6.125. The acceptable release rate under DBA conditions shall be determined with account taken of the source term and other parameters such as filtration, the point of release, environmental conditions, and the pressure and temperature under DBA conditions.

6.126. Each penetration of the barriers shall be capable of being automatically and reliably sealed in the event of DBA conditions arising (including those that may produce increases in pressure), in which the control of leakage from the barrier is essential to prevent the release of radioactive material to the environment in excess of acceptable limits.

6.127. Provisions to enable initial and periodic performance tests to check air leakage rates and the operational performance of the ventilation system shall be included in the design.

6.128. Where confinement is dependent on the efficiency of filters, provision shall be made as appropriate for in situ periodic testing of the efficiency of the filters.

6.129. For structures and components performing the function of confinement, coverings and coatings shall be carefully selected and their methods of application shall be specified so as to ensure the fulfilment of their safety functions and to minimize interference with other safety functions in the event of their deterioration.

6.130. For research reactors that have greater potential hazards associated with them, consideration shall be given to the provision of a containment structure to ensure that in DBAs, including both internal and external events, any release of radioactive material will be kept below authorized limits. Specific procedures shall be put in place for mitigating the consequences of selected BDBAs.

Experimental devices

6.131. Experimental devices shall be so designed that they will not adversely affect the safety of the reactor in any operational states. In particular, experimental equipment shall be so designed that neither its operation nor its failure will result in an unacceptable change in reactivity for the reactor, in a reduction of cooling capacity or in an unacceptable radiation exposure.

6.132. A design basis shall be established for each experimental device associated directly or indirectly with the reactor. The radioactive inventory of the experimental device as well as the potential for the generation or release of energy shall be taken into consideration. A safety analysis shall also be performed, including an analysis of the damage that would be caused to the experimental devices by the postulated initiating events of the reactor.

6.133. If safety devices are interconnected with the reactor protection system, they shall be designed to maintain the quality of the reactor protection system. The possibility of deleterious interactions with the reactor protection system shall be assessed.

6.134. Where necessary for the safety of the reactor and the safety of the experiment, the design shall provide appropriate monitoring of the parameters for experiments in the reactor control room and shall include specific safety features, if necessary, for the reactor systems, for the experimental devices and for any other related facility, such as for bunkers that contain experimental devices with stored energy.

6.135. Requirements for the safe utilization of experimental devices and requirements for deciding which devices and experiments shall be referred to the regulatory body shall be included in the OLCs. OLCs and limiting conditions for safe operations (see para. 7.35) shall be prepared for the device and incorporated as appropriate into the OLCs of the research reactor. A preliminary decommissioning plan shall be prepared for the device. Further guidance on the safety of experimental devices is provided in Ref. [15].

Instrumentation and control

6.136. The reactor shall be provided with sufficient instrumentation for monitoring its operation and process systems in normal operation and for recording all variables important to safety. The reactor shall be provided with appropriate controls, both manual and automatic, to maintain parameters

within specified operating ranges. The reactor shall be provided with sufficient indicators and recording instrumentation to monitor important reactor parameters during and following anticipated operational occurrences and DBAs. This instrumentation shall be adequate for the purposes of emergency response.

6.137. The selection and arrangement of the instrumentation and the means of display shall be planned with account taken of ergonomic principles, to allow for the operator to assimilate the information and to take appropriate safety related actions, thus reducing the possibility of operator errors. The arrangement is usually centralized in an adequately equipped reactor control room. Appropriate measures shall be taken to protect the occupants of this control room during anticipated operational occurrences and accidents.

6.138. If the design is such that a system important to safety is dependent upon the reliable performance of a computer based system, appropriate standards and practices for the development and testing of computer hardware and software shall be established and adopted throughout the lifetime of the system. For computer based digital instrumentation and control systems, verification, validation and testing of software shall be provided.

6.139. The level of reliability required shall be commensurate with the safety importance of the system. The required level of reliability shall be achieved by means of a comprehensive strategy that uses various complementary means (including an effective regime of analysis and testing) at each phase of development of the system and a validation strategy to confirm that the design requirements for the system have been fulfilled. The conditions in which equipment is to be used and stored and the effects of possible environmental factors (e.g. humidity, extreme temperature, and electromagnetic fields) shall be taken into account in the reliability analysis.

6.140. The level of reliability assumed in the safety analysis for a computer based system shall include a specified conservatism to compensate for the inherent complexity of the technology and the consequent difficulty of analysis.

6.141. In the design of the instrumentation and control systems, provision shall be made for startup neutron sources and dedicated startup instrumentation for conditions in which they are needed. This requirement shall be fulfilled for all commissioning and after long shutdowns.

6.142. Audio and visual alarm systems shall be provided for the early indication of changes in the operating conditions of the reactor that could affect its safety.

6.143. The design shall include adequate provision for the inspection, testing and maintenance of safety related instrumentation.

6.144. Where necessary, a supplementary control room, separated and functionally independent from the main control room, shall be provided where the staff could operate in the event of an emergency. Information on important parameters and the radiological conditions in the facility and its surroundings shall be made available in the supplementary control room. Systems designed for this purpose shall be considered safety related systems.

Radiation protection systems

6.145. Radiation protection systems shall be provided for research reactors to ensure adequate monitoring for radiation protection purposes in operational states, DBAs and, as practicable, BDBAs, including:

(a) Stationary dose rate meters for monitoring the local radiation dose rate at places routinely occupied by operating personnel and at other places (e.g. beam tube areas) where changes in radiation levels may occur;
(b) Stationary dose rate meters to indicate the general radiation levels at appropriate locations in the event of anticipated operational occurrences, DBAs and, as practicable, BDBAs;
(c) Monitors for measuring the activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel and where the levels of airborne activity may be expected to be such as to require protective measures;
(d) Stationary equipment and laboratories for determining the concentrations of selected radionuclides in fluid process systems and in gas and liquid samples taken from the research reactor facility or the environment in operational states, DBAs and, as practicable, BDBAs;
(e) Stationary equipment for monitoring effluents prior to or during their discharge to the environment;
(f) Devices for measuring radioactive surface contamination;
(g) Installations and equipment needed for measuring doses to and contamination of personnel;

(h) Radiation monitoring at gates and other possible points of egress from the facility for radioactive material being removed from the reactor building without permission or by unnoticed contamination.

6.146. As required, the instruments mentioned above shall be used to provide an indication in the control room and other appropriate control positions in all operational states, DBAs and, as practicable, BDBAs.

6.147. Measures shall be taken to prevent the spread of radioactive contamination by means of adequate monitoring systems (see also paras 7.72–7.78).

6.148. In addition to monitoring within the facility, arrangements shall also be made to determine the radiological consequences of the facility in the vicinity, where necessary.

Fuel handling and storage systems

6.149. The design shall include provisions for the safe handling and storage of fresh and irradiated fuel.

6.150. The design shall include provisions for storing a sufficient number of spent fuel elements. These provisions shall be in accordance with the programmes for core management and for removing fuel elements from the facility, and shall be in compliance with the requirement established in para. 6.154 and the documented limiting conditions for safe operation and requirements for periodic testing as specified in the operational limits and conditions and outlined in the SAR (see para. 7.35).

6.151. The design shall include provisions to unload the core safely at all times.

6.152. The implications of the storage of irradiated fuel over an extended time period shall be considered in the design, where applicable.

6.153. The handling and storage systems for fresh and irradiated fuel shall be designed:

(a) To prevent inadvertent criticality by physical means such as the use of an appropriate geometry and fixed absorbers;
(b) To permit periodic inspection and testing;
(c) To minimize the probability of loss of or damage to the fuel;

(d) To prevent the inadvertent dropping of heavy objects on the fuel;
(e) To permit the storage of suspect or damaged fuel elements;
(f) To provide for radiation protection;
(g) To provide a means for controlling the chemistry and activity of the storage medium;
(h) To provide physical protection against theft and sabotage;
(i) To prevent unacceptable levels of stress in the fuel elements;
(j) To identify individual fuel elements.

6.154. Handling and storage systems for irradiated fuel shall be designed to permit adequate heat removal in operational states and DBAs.

Electrical power supply systems

6.155. The basis for the design of normal and emergency electrical power systems shall be specified. The availability of reliable electrical power supplies for essential functions (e.g. the reactor protection system, cooling systems, radiation protection systems, communications, physical protection, instrumentation, emergency lighting and emergency ventilation) in DBAs shall be included in this design basis.

6.156. Consideration shall be given to the need for uninterruptible power supplies.

6.157. The provision of an emergency electrical power system with adequate reliability to ensure the availability of emergency electrical power when it is required for systems important to safety shall be considered.

6.158. The maximum period for the interruption of AC and DC electrical power supplies shall be specified and shall be demonstrated to be acceptable.

6.159. In the design of an emergency electrical power system, the starting load requirements of the various items of equipment served by the system shall be taken into account.

6.160. Appropriate means of testing the functional capability of the emergency power supply system shall be provided in the design.

6.161. In the selection and routing of electrical and signal cables, common cause failure mechanisms such as electrical interference and fire shall be

considered, and appropriate solutions (e.g. separation, redundancy or the use of suitable materials) shall be adopted.

Radioactive waste systems

6.162. The design and the operation (see para. 7.104) of the research reactor shall be such as to minimize the generation of radioactive waste. Treatment systems for radioactive waste shall include adequate provisions for control and monitoring to keep releases as low as reasonably achievable and below authorized limits.

6.163. Appropriate means, such as shielding and decay systems, to reduce the exposure of personnel and radioactive releases to the environment shall be considered in the design.

6.164. Suitable means of measuring discharges to the environment, such as by sampling and monitoring of discharges of radioactive effluents, shall be provided in the design.

6.165. Means shall be provided in the design, as necessary, for the handling, collection, processing, storage, removal from the site and disposal of radioactive waste. Where liquid radioactive waste is to be handled, provision shall be made for the detection of leakage and the recovery of waste, if appropriate.

6.166. Systems shall be provided for the handling of solid or concentrated radioactive waste and for its storage on the site for a reasonable period of time.

Buildings and structures

6.167. The buildings and structures important to safety shall be designed for all operational states, DBAs and, as far as practicable, BDBAs. However, these items may constitute engineered safety features, for which specific design requirements are established in paras 6.32–6.34.

6.168. The buildings and structures important to safety shall be designed to keep radiation levels and radioactive releases on and off the site as low as reasonably achievable and below authorized limits in all operational states and DBAs.

6.169. The required degree of leaktightness of the reactor building or of other buildings and structures containing radioactive material and the requirements for the ventilation system shall be determined in accordance with the safety analysis of the reactor and its utilization.

Auxiliary systems

6.170. The failure of any auxiliary system, irrespective of its importance to safety, shall not be able to jeopardize the safety of the reactor. Adequate measures shall be taken to prevent the release of radioactive material to the environment in the event of the failure of an auxiliary system containing radioactive material.

6.171. Where necessary for the safety of the research reactor and its associated facilities, adequate provision for communication systems shall be made.

7. OPERATION[26]

ORGANIZATIONAL PROVISIONS

Structure and responsibilities of the operating organization

7.1. The operating organization shall establish an appropriate management structure for the research reactor and shall provide for all necessary infrastructures for the conduct of reactor operations. The organization for reactor operation (the reactor management[27]) shall include the reactor

[26] Operation includes all activities performed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified. This covers: maintenance, testing and inspection; fuel handling and handling of radioactive material, including the production of radioisotopes; installation, testing and operation of experimental devices; the use of neutron beams; use of the research reactor systems for the purposes of research and development and education and training; and other associated activities.

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

manager and the operating personnel. The operating organization shall ensure that adequate provision is made for all functions relating to the safe operation and utilization of the research reactor facility, such as inspection, periodic testing and maintenance, radiation protection, quality assurance and relevant support services.

7.2. The operating organization shall have the overall responsibility for the safety of the research reactor, which shall not be delegated. The reactor manager shall have the direct responsibility and the necessary authority for the safe operation of the research reactor. However, the regulatory body shall retain the authority to prohibit certain activities or to require their reconsideration if it so considers. A system for reviewing and reporting abnormal occurrences shall be established.

7.3. The operating organization shall establish the functions and responsibilities for the key positions in the organization for reactor operation. In particular, the operating organization shall clearly establish lines of authority and communications between the reactor manager, the safety committee(s), the radiation protection group, maintenance groups, the quality assurance personnel and the experimenters.

7.4. The operating organization shall determine the staff positions that require a licence or certificate and shall provide for adequate training in accordance with the requirements of the regulatory body (see also paras 7.11–7.27). In particular, the reactor manager, the shift supervisors and the reactor operators shall hold a licence or certification issued by an appropriate authority.

7.5. The operating organization shall establish and implement a radiation protection programme to ensure that all activities involving radiation exposure or potential exposure are planned, supervised and executed to achieve the aims stated in paras 7.93–7.107. In particular, the operating organization shall ensure that adequate measures are in place to provide protection against radiological hazards arising from utilization and modification projects for the reactor (see also paras 7.85–7.92).

7.6. The operating organization shall have overall responsibility for the preparation and satisfactory completion of the commissioning programme (see paras 7.42–7.50).

7.7. The operating organization shall prepare and issue specifications and procedures, in particular for the procurement, loading, utilization, unloading, storage, movement and testing of fuel, core components and other fresh or irradiated fissile material.

7.8. In the operational stage of the research reactor, the operating organization shall become familiar with decommissioning projects at similar research reactors to facilitate the assessment of the complexity and costs of the ultimate decommissioning of its own reactor. Before decommissioning, the operating organization shall prepare a detailed plan to ensure safety throughout decommissioning.

7.9. The operating organization shall prepare periodic summary reports on matters relating to safety as required by the regulatory body and shall submit these reports to the safety committee and to the regulatory body if so required.

7.10. It shall be the responsibility of the operating organization to ensure that:

(a) The design enables the reactor to be operated safely and the reactor is constructed in accordance with the approved design;
(b) An adequate SAR is prepared and kept up to date;
(c) The commissioning process demonstrates that the design requirements have been met and that the reactor can be operated in accordance with the design assumptions;
(d) A radiation protection programme is developed and implemented;
(e) Emergency procedures are established and implemented;
(f) The research reactor is being operated and maintained in accordance with the safety requirements by suitably qualified and experienced personnel;
(g) Personnel with responsibilities relating to safe operation are adequately trained, and a training and retraining programme is established, implemented and kept up to date and periodically reviewed to verify its effectiveness (see also paras 7.27 and 7.28);
(h) Adequate facilities and services are available during operation;
(i) Information on reportable incidents, including any assessments of such events and the corrective actions intended, is submitted to the regulatory body;
(j) Safety culture is fostered in the organization to ensure that the attitudes of personnel and the actions and interactions of all individuals and organizations are conducive to safe operation (see paras 2.11–2.14);

(k) An appropriate quality assurance programme (see footnote 14) is established and implemented (see paras 2.21 and 4.5–4.13);
(l) The reactor management is provided with sufficient authority and resources to enable it to fulfil its duties effectively;
(m) The research reactor is operated and maintained in accordance with the OLCs and operating procedures (see paras 7.29–7.41 and 7.51–7.55);
(n) The fissile and radioactive materials that are utilized or generated are controlled;
(o) Operational experience, including information on operating experience at similar research reactors, is carefully examined for any precursor signs of tendencies adverse to safety, so that corrective actions can be taken before serious adverse conditions arise and recurrences can be prevented.

Operating personnel

7.11. The operating organization shall assign direct responsibility and authority for the safe operation of the reactor to the reactor manager. The primary duties of the reactor manager shall comprise the discharge of this responsibility (see para. 7.2) The reactor manager shall have overall responsibility for all aspects of operation, inspection, periodic testing and maintenance, and utilization and modification of the reactor.

7.12. The reactor manager shall clearly document the duties, the responsibilities, the necessary experience and the training requirements of operating personnel, and their lines of communication. Other personnel involved in the operation or use of the reactor (e.g. technical support personnel and experimenters) shall also have their duties, responsibilities and lines of communication clearly documented.

7.13. The reactor manager shall specify the minimum staffing requirements for the various disciplines required to ensure safe operation for all operational states of the research reactor. These requirements include both the number of personnel and the duties for which they are required to be authorized. The person with responsibility for the direct supervision of the operation of the reactor shall be clearly identified at all times. The availability of the staff who would be required to deal with accident conditions shall also be specified.

7.14. The reactor manager shall be responsible for ensuring that the staff selected for reactor operation are given the training and retraining necessary for the safe and efficient operation of the reactor and that this training and retraining is appropriately evaluated. There shall be adequate training in the

procedures to be followed in both operational states and accident conditions (see paras 7.51–7.55).

7.15. Notwithstanding the presence of independent radiation protection personnel (see para. 7.22), the operating personnel, including technical support personnel and experimenters, shall be given suitable training in radiation protection.

7.16. The detailed programme for the operation and experimental use of the research reactor shall be prepared in advance and shall be subject to the approval of the reactor manager.

7.17. The reactor manager shall be responsible for and shall make arrangements for all the activities associated with core management and fuel handling and the handling of any other fissile material.

7.18. The reactor manager shall periodically review the operation of the research reactor, including experiments, and shall take appropriate corrective actions in regard of any problems identified. The reactor manager shall seek the advice of the safety committee or shall call upon advisers to review important safety issues arising in the commissioning, operation, inspection, periodic testing and maintenance, and modification of the reactor and experiments.

7.19. The operating personnel shall operate the facility in accordance with the approved OLCs and operating procedures (see paras 7.29–7.41 and 7.51–7.55). The number and the type of operating personnel required will depend on design aspects of the reactor, such as the power level, the duty cycle and the utilization.

7.20. Every licensed or authorized reactor operator shall have the authority to shut down the reactor in the interest of safety.

7.21. A maintenance group shall be established by the operating organization to implement the programmes for inspection, periodic testing and maintenance as discussed in paras 7.56–7.64. At some research reactors, the shift supervisor and the reactor operators are trained to perform these tasks.

Radiation protection personnel

7.22. A radiation protection group shall be established to prepare and implement a radiation protection programme and to advise the reactor management and the operating organization on matters relating to radiation protection. This is discussed in paras 7.93–7.107.

Additional support personnel

7.23. The operating organization shall make provision for additional technical personnel such as training officers, safety officers and reactor chemists.

7.24. The operating organization shall arrange for the provision of assistance by contractor personnel as required.

Safety committee

7.25. The safety committee advising the reactor manager (see para. 4.15) shall provide judgements on the safety issues submitted by the reactor manager. In particular, the safety committee shall review the adequacy and safety of proposed experiments and modifications and shall provide the reactor manager with recommendations for action. (See also paras 4.15 and 7.18.)

7.26. Notwithstanding the judgement of the safety committee, the reactor manager (see para. 7.15) shall have the authority to refuse or delay the performance of an experiment or a modification that he or she considers is not safe and shall refer such a proposal to higher authority for additional review.

TRAINING, RETRAINING AND QUALIFICATION

7.27. Training and retraining programmes shall be established for the operating personnel, including the reactor manager, the shift supervisors, the reactor operators, the radiation protection staff, the maintenance personnel, the quality assurance personnel and others working at the research reactor facility. Regular training and retraining shall be provided to enhance the knowledge and abilities of personnel continually.

7.28. Procedures shall be put in place for the validation of the training to verify its effectiveness and the qualification of the staff.

OPERATIONAL LIMITS AND CONDITIONS

General

7.29. A set of OLCs important to reactor safety, including safety limits, safety system settings, limiting conditions for safe operation, requirements for inspection, periodic testing and maintenance and administrative requirements, shall be established and submitted to the regulatory body for review and assessment.

7.30. The OLCs shall be used to provide the framework for the safe operation of the research reactor. OLCs shall be prepared for each stage in the lifetime of the reactor (e.g. commissioning and operation). The operating staff shall adhere to the OLCs throughout the lifetime of the reactor.

7.31. The OLCs shall be adequately selected, clearly established and appropriately substantiated (e.g. by clearly stating for each OLC its object, its applicability and its specification; i.e. its specified limit and its basis). The selection of and the values for the OLCs shall be based on the SAR, on the reactor design or on aspects relating to the conduct of operations, and shall be demonstrably consistent with the SAR, which reflects the present status of the reactor.

Safety limits

7.32. Safety limits shall be set to protect the integrity of the physical barriers that protect against the uncontrolled release of radioactive material. For many research reactors, the first and principal physical barrier is the cladding of the fuel material. For others, the principal physical barrier is the primary coolant boundary.

7.33. Safety limits shall be set on such important parameters as the temperature and other measured process variables that may affect the integrity of the barrier and which can be readily measured and controlled.

Safety system settings

7.34. For each parameter for which a safety limit is required and for other important safety related parameters, there shall be a system that monitors the parameter and provides a signal that can be utilized in an automatic mode to prevent that parameter from exceeding the set limit. The point for this

protective action that will provide the minimal acceptable safety margin is the safety system setting. This safety margin will allow for, among other things, behaviour in system transients, the equipment response time and inaccuracy of the measuring devices.

Limiting conditions for safe operation

7.35. Limiting conditions for safe operation are conditions established to ensure that there are acceptable margins between normal operating values and the safety system settings. The setting of limiting conditions for safe operations is aimed at avoiding the undesirably frequent actuation of safety systems. Limiting conditions for safe operations shall include limits on operating parameters, requirements relating to minimum operable equipment and minimal staffing levels, and prescribed actions to be taken by operating personnel to preserve the settings of the safety system.

Requirements for inspection, periodic testing and maintenance

7.36. Requirements shall be established for the frequency and scope of inspection, periodic testing and maintenance, operability checks and calibrations of all items important to safety to ensure compliance with safety system settings and limiting conditions for safe operation.

7.37. The requirements for inspection, periodic testing and maintenance shall include a specification that clearly states the applicability, the frequency of performance and the acceptable deviation. In order to provide operational flexibility, the specification concerning frequency shall state average intervals with a maximum that is not to be exceeded.

Administrative requirements

7.38. The OLCs shall include administrative requirements or controls concerning organizational structure and the responsibilities for key positions in the safe operation of the reactor, staffing, the training and retraining of facility personnel, review and audit procedures, modifications, experiments, records and reports, and required actions following a violation of an OLC.

Violations of OLCs

7.39. In the event that the operation of the reactor deviates from one or more OLCs, remedial actions shall be taken and the regulatory body shall be notified.

7.40. Actions shall be prescribed to be taken by the operating staff within an allowed time if a limiting condition for safe operation is violated. The reactor management shall conduct an investigation of the cause and the consequences and shall take appropriate actions to prevent a recurrence. The regulatory body shall be notified in due time.

7.41. If a safety limit is not observed, the reactor shall be shut down and maintained in a safe condition. Under such circumstances, the regulatory body shall be promptly notified, an investigation of the cause shall be carried out by the operating organization and a report shall be submitted to the regulatory body for assessment before the reactor is returned to operation.

COMMISSIONING

Commissioning programme

7.42. An adequate commissioning programme shall be prepared for the testing of reactor components and systems after their construction or modification to demonstrate that they are in accordance with the design objective and meet the performance criteria. The commissioning programme shall establish the organization and responsibilities for commissioning, the commissioning stages, the suitable testing of SSCs on the basis of their importance to safety, the test schedule, the commissioning procedures and reports, the methods of review and verification, the treatment of deficiencies and deviations, and the requirements for documentation.

7.43. Experimental devices shall be given adequate consideration during the commissioning of the reactor.

7.44. The commissioning programme shall be submitted to the safety committee and the regulatory body and shall be subjected to an appropriate review and assessment before being implemented.

Organization and responsibilities

7.45. The operating organization, designers and manufacturers shall be involved in the preparation and execution of the commissioning programme. The commissioning process shall involve co-operation between the operating organization and the supplier to ensure an effective means of familiarizing the operating organization with the characteristics of the particular reactor. Close liaison shall be maintained between the regulatory body and the operating organization throughout the commissioning process. In particular, the results and analyses of tests directly affecting safety shall be made available to the safety committee and the regulatory body for review and approval as appropriate.

Commissioning tests and stages

7.46. Commissioning tests shall be arranged in functional groups and in a logical sequence. This sequence includes pre-operational tests, initial criticality tests, low power tests and power ascension and power tests. No test sequence shall proceed unless the required previous steps have been successfully completed. The commissioning programme shall therefore be divided into stages which are usually arranged according to the following sequences:

— Stage A: tests prior to fuel loading;
— Stage B: fuel loading tests, initial criticality tests and low power tests;
— Stage C: power ascension tests and power tests.

Commissioning procedures and reports

7.47. Procedures shall be prepared, reviewed and approved for each commissioning stage prior to the commencement of tests for that stage. Commissioning activities shall be performed in accordance with approved written procedures. If necessary, the procedures shall include hold points for the notification and involvement of the safety committee, outside agencies, manufacturers and the regulatory body.

7.48. The commissioning programme shall include provisions and procedures for audits, reviews and verifications intended to ensure that the programmes have been conducted as planned and that its objectives have been fully achieved. Provisions shall also be included for resolving any deviation or deficiency that is discovered during the commissioning tests.

7.49. Reports covering the scope, sequence and expected results of these tests shall be prepared in appropriate detail and in accordance with the quality assurance requirements. The reports shall cover:

(a) The purpose of the tests and expected results;
(b) The safety provisions required to be in force during the tests;
(c) Precautions and prerequisites;
(d) The test procedures;
(e) The test reports, including a summary of the data collected and their analysis, an evaluation of the results, the identification of deficiencies, if any, and any necessary corrective actions.

7.50. The results of all commissioning tests, whether conducted by a member of the operating organization or a supplier, shall be made available to the operating organization and shall be maintained for the lifetime of the facility.

OPERATING PROCEDURES

7.51. Operating procedures shall be developed for all safety related operations that may be conducted over the entire lifetime of the facility, including:

(a) Commissioning;
(b) Operation in all operational states and, where appropriate, the loading, unloading and movement within the reactor of fuel elements and assemblies or other core and reflector components, including experimental devices;
(c) The maintenance of major components or systems that could affect reactor safety;
(d) Periodic inspections, calibrations and tests of SSCs that are essential for the safe operation of the reactor;
(e) Radiation protection activities;
(f) The review and approval process for operation and maintenance and the conduct of irradiations and experiments that could affect reactor safety or the reactivity of the core;
(g) The reactor operator's response to anticipated operational occurrences and DBAs and, to the extent feasible, to BDBAs;
(h) Emergencies[28];

[28] In many cases emergency procedures are developed as an element of a separate emergency plan (see paras 7.72–7.78).

(i) Physical protection;
(j) Handling of radioactive waste and monitoring and control of radioactive releases;
(k) Inspection, periodic testing and maintenance, as required, of the reactor and its auxiliary systems during extended periods of shutdown of the reactor;
(l) Utilization;
(m) Modifications;
(n) Activities of an administrative nature with a possible effect on safety (e.g. the control of visitors);
(o) Quality assurance.

7.52. Operating procedures shall be developed by the reactor operating personnel, in co-operation whenever possible with the designer and manufacturer and with other staff of the operating organization, including radiation protection staff. Operating procedures shall be consistent with and useful in the observance of the OLCs and shall be prepared in accordance with a general quality assurance procedure that governs the format, development, review and control of such procedures. They shall be reviewed independently (e.g. by the safety committee) and they shall be subject to the approval of the reactor manager.

7.53. The operating procedures shall be reviewed and updated periodically on the basis of the lessons learned in using the procedure or, if the need arises, in accordance with predetermined internal procedures. They shall be available as relevant for the particular operation of the reactor.

7.54. All personnel involved in the operation and use of the reactor shall be adequately trained in the use of these procedures, as relevant.

7.55. When activities that are not covered by existing procedures are planned, an appropriate procedure shall be prepared and reviewed and shall be subject to appropriate approval before the operation is started. Additional training of relevant staff in these procedures shall be provided.

INSPECTION, PERIODIC TESTING AND MAINTENANCE

7.56. Inspection, periodic testing and maintenance shall be conducted to ensure that SSCs are able to function in accordance with the design intent and with requirements, in compliance with the OLCs and in accordance with the

long term safety of the reactor. In this context, the term 'maintenance' includes both preventive and corrective actions.

7.57. There shall be documented programmes based on the SAR for the inspection, periodic testing and maintenance of the reactor equipment, especially all items important to safety. It shall be ensured by means of these programmes that the level of safety is not reduced during their execution. The inspection, periodic testing and maintenance programmes shall be reviewed at regular intervals to incorporate lessons learned from experience. All inspection, periodic testing and maintenance of systems or items important to safety shall be performed by following approved, written procedures. The procedures shall specify the measures to be taken for any changes from the normal reactor configuration and shall include provisions for the restoration of the normal configuration on the completion of the activity. A system of work permits in accordance with the quality assurance requirements shall be used for inspection, periodic testing and maintenance, including appropriate procedures for checking off before and after the conduct of the work. These procedures shall include acceptance criteria. There shall be a clearly defined structure of review and approval for the performance of the work.

7.58. Non-routine inspections or corrective maintenance of systems or items important to safety shall be performed in accordance with to a specially prepared plan and procedures. In-service inspections conducted for safety purposes and on a programmatic basis shall be performed in a similar manner.

7.59. The decision to carry out maintenance work on installed equipment, to remove equipment from operation for maintenance purposes, or to reinstall equipment after maintenance:

(a) Shall be the overall responsibility of the reactor manager;
(b) Shall be in accordance with the objective of maintaining the level of safety of the reactor as specified in the OLCs.

7.60. The frequency of inspection, periodic testing and maintenance of individual SSCs shall be adjusted on the basis of experience and shall be such as to ensure adequate reliability, in accordance with the requirements established in para. 6.35.

7.61. Equipment and items used for periodic testing and maintenance shall be identified and controlled to ensure their proper use.

7.62. Maintenance shall not be performed in such a way as to result in either deliberate or unintentional design changes to the system being maintained. If a maintenance activity requires a design change, procedures for the implementation of a modification shall be followed.

7.63. The results of inspection, periodic testing and maintenance shall be assessed by properly qualified personnel, who shall verify that the activities have been accomplished as specified in the appropriate procedure and shall verify compliance with the OLCs.

7.64. The regulatory body shall be informed of any non-conformance that is significant to safety. A maintenance assessment shall be made and the co-ordinator of maintenance activities shall review its results. The resumption of operation shall be subject to the approval of the co-ordinator of maintenance activities.

CORE MANAGEMENT AND FUEL HANDLING

7.65. Core management shall be used to produce safe operational cores consistent with the needs of the experimental programme. The basic activities for core management are:

(a) To determine by calculation, using validated methods and codes, the appropriate locations for fuel, reflectors, safety devices (such as neutron absorbing rods and valves for dumping the moderator and burnable poisons), experimental devices and moderators in the core;
(b) To keep and update baseline information on the parameters for the fuel and core configurations;
(c) To procure fuel on the basis of specifications in accordance with the design intent and the requirements of the OLCs;
(d) To load the fuel following the procedures for fuel handling;
(e) To utilize (burn up) the reactor core while ensuring the integrity of the fuel by maintaining the relevant parameters for the core configuration in accordance with the design intent and the assumptions as specified in the OLCs for the reactor, and by detecting, identifying and unloading failed fuel;
(f) To unload the irradiated fuel when appropriate.

7.66. In addition to the above activities, other activities shall be undertaken in the core management programme to ensure the safe use of the fuel in the core or to facilitate the basic activities for core management, such as:

(a) The assessment of the safety implications for any core component or material proposed for irradiation;
(b) The conduct of investigations into the causes of fuel failures and means of avoiding such failures;
(c) The assessment of the effects of irradiation on core components and core material.

7.67. Fuel handling comprises the movement, storage, transfer, packaging and transport of fresh and irradiated fuel. Applicable safety requirements shall be complied with in these processes.

7.68. Procedures shall be prepared for the handling of fuel elements and core components to ensure their quality, safety and physical protection and to avoid damage or degradation. In addition, OLCs shall be established and procedures shall be prepared for dealing with failures of fuel elements and control rods so as to minimize the amounts of radioactive products released. The integrity of the reactor core and the fuel shall be continuously monitored by a cladding failure detection system, not necessarily on-line. If a failure of fuel is detected, an investigation shall be conducted to identify the failed fuel element. Authorized limits shall not be exceeded and if necessary the reactor shall be shut down and the failed fuel element shall be unloaded (see also paras 7.96–7.102).

7.69. The packaging and transport of fuel assemblies with fresh and irradiated fuel shall be carried out in accordance with national and international requirements and, as appropriate, in accordance with Ref. [18].

7.70. A comprehensive record system shall be maintained in compliance with the quality assurance programme to cover core management, handling activities for fuel and core components and fuel storage.

FIRE SAFETY

7.71. The operating organization shall conduct periodic fire safety analyses. These analyses shall include: assessments of the vulnerability of safety systems to fire; modifications to the application of defence in depth; modifications to

fire fighting capabilities; the control of inflammables; the control of ignition sources; maintenance; testing; and the readiness of personnel.

EMERGENCY PLANNING

7.72. Emergency plans shall be prepared for a research reactor facility to cover all activities planned to be carried out in an emergency. Emergency procedures shall be prepared by the operating organization, in accordance with the requirements of the regulatory body, and in co-operation, where necessary, with the appropriate governmental and local authorities or other bodies, to ensure the effective co-ordination of all site services and of external aid in an emergency. Emergency procedures shall be based on the accidents analysed in the SAR as well as those additionally postulated for the purposes of emergency planning. Requirements for emergency planning are established in Ref. [19].

7.73. The emergency plan and arrangements prepared by the operating organization shall include, as necessary:

(a) The identification of the emergency organizations (for preparedness and response), including the authorities and responsibilities of key individuals;
(b) The identification and classification of emergencies;
(c) The conditions under which an emergency should be declared, a list of persons empowered to declare an emergency and a description of suitable warning procedures or devices;
(d) The arrangements for initial and subsequent assessment, including environmental monitoring of the radiological conditions;
(e) Agreements with off-site agencies that will help in an emergency, including letters of agreement and details of contact points;
(f) Protective measures for minimizing the exposure of persons to radiation and measures for ensuring the medical treatment of any casualties;
(g) Guidance on limits on the doses due to exposure of personnel performing rescue missions or missions to mitigate the consequences of an emergency;
(h) Action at the facility to limit the extent of any radioactive release and the spread of contamination;
(i) The chain of command and communication, clearly defining the responsibilities and duties of the persons and organizations concerned;
(j) Provisions to ensure the reliability of communications between the emergency control centre and internal and external locations;

(k) A description of facilities, equipment and procedures for emergencies;
(l) The inventory of the equipment for emergencies to be kept in readiness at specified locations;
(m) Notification requirements for informing the authorities;
(n) Notification requirements for requesting additional resources;
(o) The actions to be taken by persons and bodies involved in the implementation of the plan;
(p) Provisions for informing the public;
(q) Provisions for the training of personnel, including specification of the frequency and scope of drills;
(r) Provisions for the termination of and recovery from the emergency.

7.74. The emergency plan shall be implemented by means of emergency procedures in the form of documents and instructions detailing the implementation actions and the arrangements required to mitigate the consequences of the emergency. The emergency plan and procedures shall be reviewed at specified periods and shall be amended as necessary to ensure that lessons learned are incorporated.

7.75. The operating personnel shall take appropriate action in accordance with established emergency procedures in response to an emergency. Other on-site support service groups and off-site agencies shall be involved as specified in the emergency plan, depending on the nature and the extent of the emergency.

7.76. The emergency response team shall include persons with up to date knowledge of the operations of the research reactor, and it should normally be led by the reactor manager or a delegate. All personnel involved in responding to the emergency shall be instructed, trained and retrained periodically as necessary in the performance of their duties in an emergency. All persons on the site shall receive instruction on the steps to take in an emergency. Instructions shall be prominently displayed.

7.77. Exercises shall be conducted at suitable intervals and shall involve, to the extent practicable, all those persons with duties in responding to the emergency. The results of the exercise shall be reviewed and, where necessary, the lessons learned shall be incorporated into revisions of the emergency plan.

7.78. Facilities, instruments, tools, equipment, documentation and communication systems to be used in emergencies shall be kept available and shall be maintained in such conditions that it is unlikely that they would be affected or made unavailable by the accidents postulated to happen.

PHYSICAL PROTECTION

7.79. Appropriate measures shall be taken, in accordance with national laws and regulations, to prevent unauthorized actions, including acts of sabotage, that could jeopardize safety at research reactors and their associated facilities, and respond to such actions should they occur.

7.80. International recommendations on the physical protection of nuclear facilities and nuclear material are provided in Ref. [24].

RECORDS AND REPORTS

7.81. For the safe operation of the reactor, the operating organization shall retain all essential information concerning the design, construction, commissioning, current configuration and operation of the reactor. This information shall be maintained up to date throughout the operational stage of the reactor and shall be kept available during decommissioning. Such information includes site data and environmental data, design specifications, details of the equipment and material supplied, as-built drawings, information on the cumulative effects of modifications, logbooks, operating and maintenance manuals and quality assurance documents.

7.82. Administrative procedures consistent with the quality assurance programme shall be developed for the generation, collection, retention and archiving of records and reports. Information entries in logbooks, checklists and other appropriate records shall be properly dated and signed.

7.83. Records of non-compliance and the measures taken to return the research reactor to compliance shall be prepared and retained and shall be made available to the regulatory body. The operating organization shall specify the records to be retained and their retention periods.

7.84. The arrangements made for storing and maintaining records and reports shall be in accordance with the quality assurance programme. The document management system shall be designed to ensure that obsolete documents are archived and that personnel use only the latest version of each document. The off-site storage (e.g. in the emergency control centre) of documents for access in an emergency shall be considered.

UTILIZATION AND MODIFICATION OF THE REACTOR

7.85. The operating organization shall be responsible for all safety aspects of the preparation and performance of a modification or experiment. It may assign or subcontract the execution of certain tasks to other organizations but it shall not delegate its responsibilities. In particular, the operating organization shall be responsible for the management of the proposed utilization or modification project, in which the reactor manager shall participate according to established procedures. For major projects this shall include the setting of the objectives and the structure of the project, the appointment of a project manager, the specification of responsibilities and the allocation of adequate resources. In addition, before the project commences, it shall establish and follow approved procedures for controlling utilization and modification projects.

7.86. The operating organization shall be responsible for ensuring that:

(a) Safety analyses of the proposed utilization or modification are conducted.
(b) The approved categorization criteria are applied (see para. 7.87 and Ref. [15]).
(c) The relevant safety documentation is followed.
(d) The associated requirements for review and approval are met. These may include the requirement to obtain the approval of the regulatory body before proceeding or the establishment of a formal licensing process.
(e) Proper safety precautions and controls are applied with regard to all persons involved in the performance of the modification or experiments, and with regard to the public and the environment.
(f) Quality assurance is applied at all stages in the preparation and performance of the experiment or modification to ascertain whether all applicable safety requirements and criteria have been satisfied.
(g) All personnel who will be involved in making a proposed modification or in conducting the proposed utilization are suitably trained, qualified and experienced for the task and, if necessary, trained in advance in the effect of this modification or utilization on reactor operation and the safety characteristics of the reactor.
(h) All documents that relate to the safety characteristics of the reactor, such as the SARs, the OLCs and the relevant procedures for operation, maintenance and emergencies, are promptly updated as necessary.

7.87. Proposals for the utilization and modification of the research reactor shall be categorized and relevant criteria for this categorization shall be established.

Proposals for utilization and modification shall be categorized (see paras 305–326 of Ref. [15]) either according to the safety significance of the proposal or on the basis of a statement of whether or not the proposed change will put the operation of the reactor outside the OLCs.

7.88. Utilization and modification projects having a major safety significance (see para. 310 of Ref. [15]) shall be subject to safety analyses and to procedures for design, construction and commissioning that are equivalent to those described in paras 6.72 and 6.78 for the reactor itself.

7.89. In implementing utilization and modification projects for research reactors, the radiation exposure of the workers involved shall be kept as low as reasonably achievable.

7.90. The reactor manager shall establish a procedure for the review and approval of proposals for experiments and modifications and for the control of their performance. This procedure shall include all relevant information such as:

(a) A description of the purpose of the experiment or modification;
(b) A justification for the necessity of the experiment or modification;
(c) The requirements and criteria for design, including their safety assessment;
(d) A description of the manufacturing processes involved;
(e) A description of the installation procedures involved;
(f) A description of the commissioning process;
(g) A review of the operational procedures and emergency procedures;
(h) A description of the possible radiation hazards to experimenters;
(i) A description of the radiation safety measures necessary to prevent accidental exposure (including the restriction of access to the irradiation facility and to radioactive sources and/or neutron beams);
(j) A description of the radiation shielding required around the facility to prevent an increase in radiation (direct or scattered) generated in normal and abnormal conditions;
(k) A description of the need for the disposal of radioactive waste generated in the experiment or modification;
(l) A list of the relevant documentation that needs to be updated;
(m) Any special requirements for the training and, if necessary, relicensing of reactor operators;
(n) The quality assurance requirements.

7.91. The use and handling of experimental devices shall be controlled by means of written procedures. The possible effects on the reactor, particularly changes in reactivity, shall be taken into account in these procedures.

7.92. Any modifications made to experimental devices shall be subject to the same procedures for design, operation and approval as were followed for the original experimental device.

RADIATION PROTECTION

General

7.93. Radiation exposures at the research reactor facility shall be subject to dose constraints that are set or approved by the regulatory body or another competent authority for the purpose of ensuring that the relevant dose limits are not exceeded. In all operational states, the main aims of radiation protection shall be to avoid unnecessary exposure to radiation and to keep doses below the dose constraints and as low as reasonably achievable, social and economic factors being taken into account.

7.94. For accident conditions, the radiological consequences shall be kept low by means of appropriate engineered safety features and the measures provided for in the emergency plan.

7.95. All documentation and activities for radiation protection shall conform to the quality assurance requirements for operation.

Radiation protection programme

7.96. A radiation protection programme shall be established by the operating organization in accordance with regulatory requirements. This programme shall include a policy statement from the operating organization that includes the radiation protection objective (see para. 3.2. of Ref. [20]) and a statement of the operating organization's commitment to the principle of optimization of protection (see paras 4.9–4.12 of Ref. [20]). The radiation protection programme is subject to the requirements of the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources [12] and is subject to the approval of the regulatory body.

7.97. The radiation protection programme is subject to the requirements for occupational radiation protection (see Refs [12, 25]) and in particular shall include measures for:

(a) Ensuring that there is cooperation between the radiation protection staff and the operating staff in establishing operating procedures and maintenance procedures when radiation hazards are anticipated, and ensuring that direct assistance is provided when required;
(b) Providing for the decontamination of personnel, equipment and structures;
(c) Controlling compliance with applicable regulations for the transport of radioactive material [18];
(d) Detecting and recording any releases of radioactive material;
(e) Recording the inventory of radiation sources;
(f) Providing adequate training in practices for radiation protection;
(g) Providing for the review and update of the programme in the light of experience.

Radiation protection personnel

7.98. The radiation protection programme shall include the appointment of qualified personnel with responsibility for radiation protection who are knowledgeable about the radiological aspects of the design and operation of the reactor. These individuals shall work in cooperation with the group that operates the reactor, but they shall have reporting lines to the operating organization that are independent of the reactor management.

7.99. A qualified expert[29] shall be identified who shall be made available to the reactor manager for the purpose of advising on the observance of the radiation protection programme and its compliance with the requirements established in Ref. [12], and shall have access to the managers in the operating organization who have the authority to establish and enforce operational procedures.

7.100. All personnel at the facility shall be individually responsible for putting into practice the measures for exposure control in their areas of activity that are specified in the radiation protection programme. Consequently, particular emphasis shall be given to training all the facility's personnel to ensure that

[29] See paras 2.31 and 2.32 of Ref. [12].

they are fully aware of both the radiological hazards and the protective measures available. Special attention shall be paid to the possibility that the personnel at the research reactor facility may include persons not permanently working there (e.g. experimenters, trainees, visitors and contractors).

Reference levels

7.101. To assist the reactor management in ensuring that radiation doses are kept as low as reasonably achievable and that the dose constraints are not exceeded, the operating organization shall set reference levels for doses and/or dose rates and reference levels for radioactive releases that are below the authorized limits on releases. These reference levels shall be included in the OLCs and shall be set to comply with the radiation protection objective (see para. 205 of Ref. [1]). If the reference levels are exceeded, the operating organization shall investigate the matter for the purpose of taking corrective action.

7.102. If the applicable dose limits for occupational or public exposure or the authorized limits for radioactive releases are exceeded, the regulatory body and other competent authorities shall be informed in accordance with the requirements.

Control of occupational exposure

7.103. All personnel who may be occupationally exposed to radiation at significant levels shall have their doses measured, recorded and assessed, as required by the regulatory body or other competent authority, and these records shall be made available to the regulatory body and other competent authorities as designated in the national regulations. Detailed requirements on occupational exposure are established in Appendix I of Ref. [12].

Radioactive waste management

7.104. The reactor and its experimental devices shall be operated to minimize the production of radioactive waste of all kinds, to ensure that releases of radioactive material to the environment are kept as low as reasonably achievable and to facilitate the handling and disposal of waste. Arrangements shall be put in place for the management of solid, liquid and gaseous radioactive waste in the research reactor facility and its ultimate removal from the facility. All activities concerning radioactive effluents and waste shall be

conducted in accordance with the quality assurance programme (see footnote 14). Further requirements on the subject are established in Ref. [14].

7.105. Releases of radioactive effluents shall be monitored and the results recorded in order to verify compliance with the applicable regulatory requirements. They shall also be reported periodically to the regulatory body or another competent authority in accordance with its requirements.

7.106. Written procedures shall be followed for the handling, collection, processing, storage and disposal of radioactive waste. These activities shall be carried out in accordance with the requirements of the regulatory body or other competent authority.

7.107. An appropriate record shall be kept of the quantities, types and characteristics of the radioactive waste stored and disposed of or removed from the reactor site.

SAFETY ASSESSMENTS AND AGEING RELATED ASPECTS

7.108. The operating organization shall conduct safety assessments throughout the operational lifetime of the reactor (see paras 2.15 and 2.16). The scope of the assessments shall cover all safety related aspects of operation, including radiation protection, site re-evaluation, physical protection and emergency planning. In conducting the safety assessments, the operating organization shall give due consideration to information drawn from operating experience and other relevant sources. A programme of comprehensive periodic review will fulfil this requirement for safety assessments. On the basis of the results of the safety assessments, the operating organization shall implement any necessary corrective actions and shall consider making justified modifications to enhance safety.

7.109. The programme of periodic review should cover aspects of the programme for the management of ageing to demonstrate the status of the facility with regard to ageing and to provide a basis for taking actions in relation to ageing. Thus, periodic reviews are operational tools to prevent and mitigate the effects of ageing and of modifications made around the site. Reviews of reactor SSCs carried out by using non-destructive techniques are called in-service inspections. In-service inspections shall be conducted by the operating organization under its programme for the management of ageing (see paras 6.68–6.70).

Peer reviews

7.110. Some reviews of research reactors shall be performed as peer reviews, i.e. by reviewers from other research reactors which are performing well. Such peer reviews will provide access to the practices and programmes at other research reactors (see paras 2.16 and 4.16).

EXTENDED SHUTDOWN

7.111. A research reactor facility may have a period of extended shutdown pending decisions on its future, owing to budgetary considerations, a lack of utilization or equipment failure, for example. While an extended shutdown may be planned, more often it will be unanticipated. The operating organization shall take appropriate measures during an extended shutdown to ensure that materials and components do not seriously degrade. The following measures shall be considered:

(a) Unloading the fuel elements from the reactor core to the storage racks;
(b) Changing the OLCs in accordance with the requirements for the shut down reactor;
(c) Removing components for protective storage;
(d) Taking measures to prevent accelerated corrosion and ageing;
(e) Retaining adequate staff in the facility for the purposes of performing the necessary inspection, periodic testing and maintenance.

7.112. The operating organization shall take the necessary decisions as soon as possible to reduce the period of extended shutdown to a minimum. During a period of extended shutdown, the operating organization shall consider the consequences of the shutdown for the fulfilment of the licence conditions (e.g. for the physical protection of the fuel) and for the qualification of the operating staff.

8. DECOMMISSIONING

8.1. For some operating research reactors, the need for their ultimate decommissioning was not taken into account in their design. Nevertheless, all operational activities at research reactors, including inspection, periodic testing

and maintenance, modification and experiments, shall be conducted in a way that will facilitate their decommissioning. Documentation of the reactor shall be kept up to date and information on experience with the handling of contaminated or irradiated SSCs in the maintenance or modification of the reactor shall be recorded to facilitate the planning of decommissioning.

8.2. A decommissioning plan shall be prepared to ensure safety throughout the decommissioning process. The decommissioning plan shall be submitted for review and approval by the safety committee and the regulatory body before decommissioning activities are commenced. Guidance on the decommissioning of research reactors is provided in Ref. [16].

8.3. The decommissioning plan shall include an evaluation of one or more approaches to decommissioning that are appropriate for the reactor concerned and are in compliance with the requirements of the regulatory body. The following are examples of approaches to decommissioning:

(a) Protective storage of the reactor in an intact condition after the removal of all fuel assemblies and of all readily removable activated and radioactively contaminated components and radioactive waste;
(b) The entombment of activated structures and large components after the removal of all fuel assemblies and of all readily removable activated and radioactively contaminated components and radioactive waste from the reactor;
(c) The removal of all radioactive material and all removable activated and radioactively contaminated components from the reactor and the thorough decontamination of the remaining structures to permit the unrestricted use of the facility.

8.4. In developing the decommissioning plan, aspects of the reactor's design to facilitate decommissioning shall be reviewed, such as the selection of materials to reduce activation and to facilitate decontamination, the installation of remote handling capabilities for the removal of activated components and the incorporation of facilities for the processing of radioactive waste. In addition, aspects of the facility's operation that are important in relation to decommissioning, such as any unintentional contamination whose cleanup has been deferred until the reactor's decommissioning, and any modifications that may not have been fully documented, shall also be reviewed. The decommissioning plan shall include all the steps that lead to the ultimate completion of decommissioning to the point that safety can be ensured with minimum or no surveillance. These stages may include storage and surveillance,

restricted site use and unrestricted site use. Guidance on decommissioning is provided in Ref. [16].

8.5. The decision to decommission a reactor is often made after a period of extended shutdown. Occurrences at the reactor over this period shall be taken into consideration in developing the decommissioning plan.

8.6. All activities conducted during the decommissioning process shall be subject to a quality assurance programme (see footnote 14).

8.7. The responsibility of the operating organization shall be terminated only with the approval of the regulatory body.

8.8. Procedures for the handling, dismantling and disposal of experimental devices and other irradiated equipment that require storage and eventual disposal shall be established in advance, or as early as possible if the equipment concerned has already been constructed and these procedures are not in place. For guidance on this matter, see paras 901–908 of Ref. [15].

Appendix

SELECTED POSTULATED INITIATING EVENTS FOR RESEARCH REACTORS

(l) Loss of electrical power supplies:
- Loss of normal electrical power[30].

(2) Insertion of excess reactivity:
- Criticality during fuel handling (due to an error in fuel insertion);
- Startup accident;
- Control rod failure or control rod follower failure;
- Control drive failure or system failure;
- Failure of other reactivity control devices (such as a moderator or reflector);
- Unbalanced rod positions;
- Failure or collapse of structural components;
- Insertion of cold water;
- Changes in the moderator (e.g. voids or leakage of D_2O into H_2O systems);
- Influence by experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material);
- Insufficient shutdown reactivity;
- Inadvertent ejections of control rods;
- Maintenance errors with reactivity devices;
- Spurious control system signals.

(3) Loss of flow:
- Primary pump failure;
- Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger);
- Influence of the failure or mishandling of an experiment;
- Rupture of the primary coolant boundary leading to a loss of flow;
- Fuel channel blockage;

[30] Although the loss of normal electrical power is not considered an initiating event, consideration should be given to the loss of normal electrical power followed by the loss of emergency power to ensure that the consequences would be acceptable under emergency conditions (for example, a drop in voltage may cause devices to fail at different times).

— Improper power distribution due, for example, to unbalanced rod positions in core experiments or fuel loading (power–flow mismatch);
— Reduction in coolant flow due to bypassing of the core;
— Deviation of system pressure from the specified limits;
— Loss of heat sink (e.g. due to the failure of a valve or pump or a system rupture).

(4) Loss of coolant:
— Rupture of the primary coolant boundary;
— Damaged pool;
— Pump-down of the pool;
— Failure of beam tubes or other penetrations.

(5) Erroneous handling or failure of equipment or components:
— Failure of the cladding of a fuel element;
— Mechanical damage to core or fuel (e.g. mishandling of fuel and dropping of a transfer flask onto the fuel);
— Failure of a emergency cooling system;
— Malfunction of the reactor power control;
— Criticality in fuel in storage;
— Failure of means of confinement, including the ventilation system;
— Loss of coolant to fuel during transfer or storage;
— Loss or reduction of proper shielding;
— Failure of experimental apparatus or material (e.g. loop rupture);
— Exceeding of fuel ratings.

(6) Special internal events:
— Internal fires or explosions;
— Internal flooding;
— Loss of support systems;
— Security related incidents;
— Malfunctions in reactor experiments;
— Improper access by persons to restricted areas;
— Fluid jets and pipe whip;
— Exothermic chemical reactions.

(7) External events:
— Earthquakes (including seismically induced faulting and landslides);
— Flooding (including failure of an upstream dam and blockage of a river);
— Tornadoes and tornado missiles;
— Sandstorms;
— Hurricanes, storms and lightning;
— Tropical cyclones;
— Explosions;

— Aircraft crashes;
— Fires;
— Toxic spills;
— Accidents on transport routes;
— Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);
— Biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects;
— Extreme meteorological phenomena;
— Lightning strikes;
— Power or voltage surges on the external supply line.

(8) Human errors.

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Annex I

SELECTED SAFETY FUNCTIONS FOR RESEARCH REACTORS

I-1. Selected safety functions for research reactors are shown in Table I-1. Safety functions are the essential characteristic functions associated with SSCs for ensuring the safety of the reactor. The safety functions are appropriate for the particular design of reactor. Some safety functions are not relevant for some types of research reactor. The safety functions are one of the key elements in grading the application of requirements to SSCs. The safety functions that each SSC fulfils have to be identified. The selected safety functions presented in Table I-1 are for consideration by the operating organization for a research reactor. A justification needs to be made for not providing for the fulfilment of any of these safety functions for a particular reactor.

TABLE I-1. SELECTED SAFETY FUNCTIONS FOR RESEARCH REACTORS

Items important to safety	Safety functions
Buildings and structures	(a) To form a barrier to the uncontrolled release of radioactive material to the environment (b) To provide protection against external and internal events for the enclosed safety systems (c) To provide shielding against radiation
Reactor core	(a) To maintain the fuel geometry and the necessary coolant flow path so as to ensure the possibility of shutdown and heat removal in all operational states of the reactor and in DBAs (b) To provide negative feedback of reactivity (c) To provide a means of moderating and controlling neutron fluxes
Fuel matrix and cladding	(a) To form a barrier to the release of fission products and other radioactive material from the fuel (b) To provide a constant configuration
Reactivity control system (including the reactor shutdown system)	To control the reactivity of the reactor core to ensure that the reactor can be safely shut down and to ensure that the fuel design limits and other limits will not be exceeded in any operational state of the reactor or in DBAs

TABLE I-1. SELECTED SAFETY FUNCTIONS FOR RESEARCH REACTORS (cont.)

Items important to safety	Safety functions
Reactor coolant primary circuit	To provide adequate core cooling and to ensure that the specified limits for the fuel and the coolant will not be exceeded in any operational state of the reactor or in DBAs
Emergency core cooling system	To transfer heat from the reactor core following a loss of coolant accident at an adequate rate to prevent significant damage to the fuel
Reactor protection system	(a) To take protective actions to shut down the reactor, to cool and contain radioactive material and to mitigate the consequences of accidents (b) To control interlocks to protect against operational errors if the required conditions have not been met
Other safety related instrumentation and control systems	(a) To keep reactor parameters within operational limits without reaching safety limits (b) To provide and present to the reactor operator enough information to determine readily the status of the reactor protection system and to take the correct safety related actions
Electrical power supply	To provide sufficient power of suitable quality to systems and equipment to ensure their capability to perform their safety functions when required
Fuel handling and storage system	(a) To minimize radiation exposure (b) To prevent inadvertent criticality (c) To limit any rise in fuel temperature (d) To store fresh and irradiated fuel (e) To prevent mechanical or corrosive damage of fuel
Radiation monitoring system	To provide measurements and warnings to minimize the radiation exposure of operating and research personnel
Fire protection system	To ensure that the adverse effects of fire or fire induced explosions do not prevent items important to safety from performing their safety function when required to do so

Annex II

OPERATIONAL ASPECTS OF RESEARCH REACTORS WARRANTING PARTICULAR ATTENTION

II-1. Annex II highlights operational aspects of research reactors that warrant special consideration.

REACTIVITY AND CRITICALITY MANAGEMENT

II-2. Core configurations are frequently changed in research reactors and these changes involve the manipulation of components such as fuel assemblies, control rods and experimental devices, many of which represent considerable reactivity value. Care has to be taken to ensure that the relevant subcriticality limits and reactivity limits for fuel storage and core loading are not exceeded at any time.

CORE THERMAL SAFETY

II-3. The frequent changes in core loading referred to above affect the nuclear and thermal characteristics of the core. Care has to be taken to ensure that, in each case, these characteristics are correctly determined and that they are checked against the relevant conditions for nuclear and thermal safety before the reactor is put into operation.

SAFETY OF EXPERIMENTAL DEVICES

II-4. Experimental devices used in research reactors may, by virtue of their technical, nuclear or operational characteristics, significantly affect the safety of the reactor. Care has to be taken to ensure that the technical, nuclear and operational characteristics of experimental devices are adequately assessed for their safety implications and that suitable documentation is made available.

MODIFICATION OF REACTORS

II-5. Research reactors and their associated experimental devices are often modified in order to adapt their operational and experimental capabilities to changing requirements for their utilization. Special assurance is needed to verify that every modification has been properly assessed, documented and reported in terms of its potential effects on safety, and that the reactor is not restarted without formal approval after the completion of modifications with major implications for safety.

MANIPULATIONS OF COMPONENTS AND MATERIAL

II-6. In pool type research reactors in particular, components, experimental devices and material are frequently manipulated in the vicinity of the reactor core. Special assurance is needed that the persons carrying out these manipulations will adhere strictly to the procedures and restrictions established to prevent any nuclear or mechanical interference with the reactor, to minimize the probability of a blockage in the fuel cooling system by uncontrolled foreign objects, and to prevent radioactive releases and undue radiation exposures.

SAFETY MEASURES FOR VISITORS

II-7. Guest scientists, trainees, students and other persons who visit research reactors may have access to controlled areas and may be actively involved in the operation or utilization of the reactor. Care has to be taken to ensure that all procedures, restrictions and controls that are aimed at verifying that such visitors have safe working conditions and that their activities will not affect the safety of the reactor are strictly observed.

GLOSSARY

acceptable limit.
See limit.

applicant. A legal person who applies to a regulatory body for authorization to undertake specified activities.

area.

controlled area. A defined area in which specific protection measures and safety provisions are or could be required for controlling normal exposures or preventing the spread of contamination during normal working conditions, and preventing or limiting the extent of potential exposures. A controlled area is often within a supervised area, but need not be.

operations area. A geographical area that contains an authorized facility. It is enclosed by a physical barrier (the operations boundary), to prevent unauthorized access and by means of which the management of the authorized facility can exercise direct authority.

site area. A geographical area that contains an authorized facility, and within which the management of the authorized facility may directly initiate emergency actions. This area is often identical to the operations area, except in situations (e.g. research reactors, irradiation installations) where the authorized facility is on a site where other activities are being carried out beyond the operations area, but where the management of the authorized facility can be given some degree of authority over the whole site area. The site boundary is the boundary of the site area.

supervised area. A defined area not designated a controlled area but for which occupational exposure conditions are kept under review, even though specific protection measures and safety provisions are not normally needed.

authorization. The granting by a regulatory body or other governmental body of written permission for an operator to perform specified activities. Authorization could include, for example, licensing, certification, registration, etc. The term authorization is also sometimes used to

describe the document granting such permission. Authorization is normally a more formal process than approval.

authorized limit.
See limit.

commissioning. The process during which systems and components of facilities and activities, having been constructed, are made operational and verified to be in accordance with the design and to have met the required performance criteria.

common cause failure. Failure of two or more structures, systems or components due to a single specific event or cause.

containment. Methods or physical structures designed to prevent the dispersion of radioactive substances. Containment is normally used to refer to methods or structures to prevent radioactive substances from being dispersed in the environment if confinement fails.

critical assembly. An assembly containing fissile material intended to sustain a controlled fission chain reaction at a low power level, used to investigate reactor core geometry and composition.

critical group. A group of members of the public which is reasonably homogeneous with respect to its exposure for a given radiation source and is typical of individuals receiving the highest effective dose or equivalent dose (as applicable) from the given source.

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository, which is closed and not decommissioned).

design basis. The range of conditions and events taken explicitly into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.

disposal. Emplacement of waste in an appropriate facility without the intention of retrieval.

diversity. The presence of two or more redundant systems or components to perform an identified function, where the different systems or components have different attributes so as to reduce the possibility of common cause failure. Examples of such attributes are: different operating conditions, different working principles or different design teams (which provide functional diversity), and different sizes of equipment, different manufacturers, and types of equipment that use different physical methods (which provide physical diversity).

dose constraint. A prospective restriction on the individual dose delivered by a source, which serves as an upper bound on the dose in optimization of protection and safety for the source.

dose limit
See limit.

facilities and activities. A general term encompassing nuclear facilities, uses of all sources of ionizing radiation, all radioactive waste management activities, transport of radioactive material and any other practice or circumstances in which people may be exposed to radiation from naturally occurring or artificial sources. Facilities include nuclear facilities, irradiation installations, mining and milling facilities, waste management facilities and any other place where radioactive materials are produced, processed, used, handled, stored or disposed of — or where radiation generators are installed — on such a scale that consideration of protection and safety is required. Activities include the production, use, import and export of radiation sources for industrial, research and medical purposes, the transport of radioactive material, the mining and processing of radioactive ores and closeout of associated facilities, cleanup of sites affected by residues from past activities and radioactive waste management activities such as the discharge of effluents.

fuel assembly. A set of fuel elements and associated components which are loaded into and subsequently removed from a reactor core as a single unit.

fuel element. A rod [or other form] of nuclear fuel, its cladding and any associated components necessary to form a structural entity.

level.

action level. The level of dose rate or activity concentration above which remedial actions or protective actions should be carried out in chronic exposure or emergency exposure situations.

intervention level. The level of avertable dose at which a specific protective action or remedial action is taken in an emergency exposure situation or a chronic exposure situation.

investigation level. The value of a quantity such as effective dose, intake, or contamination per unit area or volume at or above which an investigation should be conducted.

recording level. A level of dose, exposure or intake specified by the regulatory body at or above which values of dose, exposure or intake received by workers are to be entered in their individual exposure records.

reference level. An action level, intervention level, investigation level or recording level.

licence. A legal document issued by the regulatory body granting authorization to perform specified activities related to a facility or activity. The holder of a current licence is termed a licensee.

limit. The value of a quantity used in certain specified activities or circumstances that must not be exceeded. The term limit should only be used for a criterion that must not be exceeded, e.g. where exceeding the limit would cause some form of legal sanction to be invoked. Criteria used for other purposes — e.g. to indicate a need for closer investigation or a review of procedures, or as a threshold for reporting to a regulatory body — should be described using other terms, such as reference level.

acceptable limit. A limit acceptable to the regulatory body. The term acceptable limit is usually used to refer to a limit on the predicted radiological consequences of an accident (or on potential exposures if they occur) that is acceptable to the relevant regulatory body when the probability of occurrence of the accident or potential exposures has been taken into account (i.e. on the basis that it is unlikely to occur). The term authorized limit should be used to refer to limits on doses or risks, or on

releases of radionuclides, which are acceptable to the regulatory body on the assumption that they are likely to occur.

authorized limit. A limit on a measurable quantity, established or formally accepted by a regulatory body.

dose limit. The value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded.

operational limits and conditions. A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.

safety limits. Limits on operational parameters within which an authorized facility has been shown to be safe. Safety limits are operational limits and conditions beyond those for normal operation.

maintenance. The organized activity, both administrative and technical, of keeping structures, systems and components in good operating condition, including both preventive and corrective (or repair) aspects.

monitoring. Continuous or periodic measurement of radiological or other parameters or determination of the status of a system. Sampling may be involved as a preliminary step to measurement.

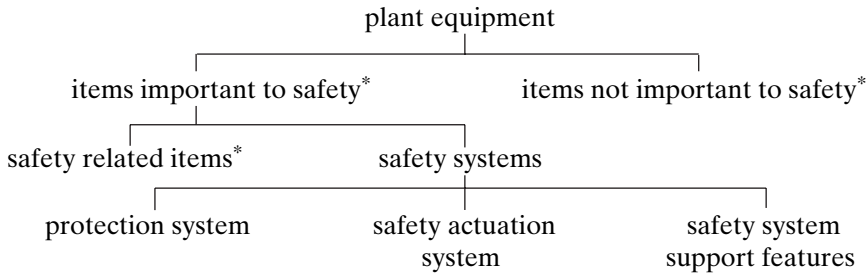
nuclear safety (or safety)**.** The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers (and other site personnel), the public and the environment from undue radiation hazards.

— Often abbreviated to safety in IAEA publications on nuclear safety, particularly when other types of safety (e.g. fire safety, conventional industrial safety) are being discussed.

operating organization. The organization authorized by the regulatory body to operate a facility.

operational limits and conditions.
See limit.

plant equipment (reactor equipment).



* In this context, an 'item' is a structure, system or component

item important to safety. An item that is part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public. Items important to safety include:

— those structures, systems and components whose malfunction or failure could lead to undue radiation exposure of site personnel or members of the public;
— those structures, systems and components which prevent anticipated operational occurrences from leading to accident conditions; and
— those features which are provided to mitigate the consequences of malfunction or failure of structures, systems or components.

protection system. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition. The 'system' in this case encompasses all electrical and mechanical devices and circuitry, from sensors to actuation device input terminals.

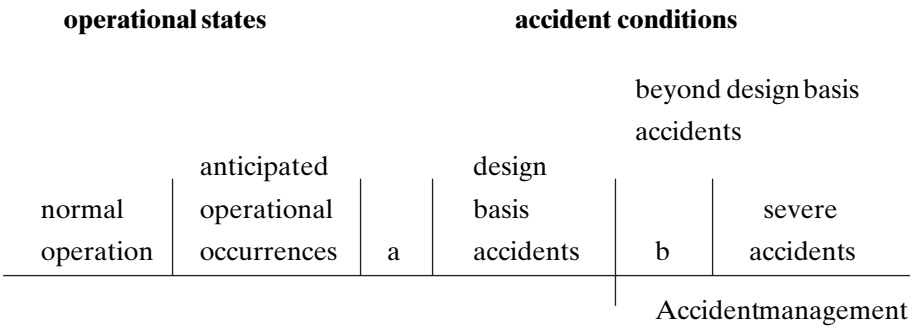
safety actuation system. The collection of equipment required to accomplish the necessary safety actions when initiated by the protection system.

safety related item. An item important to safety which is not part of a safety system.

safety system.[31] A system important to safety, provided to ensure the safe shutdown of the reactor or residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents. Safety systems consist of the protection system, the safety actuation systems and the safety system support features. Components of safety systems may be provided solely to perform safety functions or may perform safety functions in some plant operational states and non-safety functions in other operational states.

safety system support features. The collection of equipment that provides services such as cooling, lubrication and energy supply required by the protection system and the safety actuation systems.

plant states (reactor states).



a: Accident conditions which are not explicitly considered design basis accidents but are encompassed by them.

b: Beyond design basis accidents without significant core degradation.

accident conditions. Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

[31] Safety systems may be of the active or passive type. Active systems or components are those that will initiate performance of their designated functions on receiving an input signal from the protection system or on receiving a manual signal. Passive systems or components are those that do not need an input signal to initiate their designated functions. There is a recognized degree of passivity for safety systems that allows for a definition (not universally recognized) of three categories. The highest category is the one in which all the components necessary for safety are passive.

accident management. The taking of a set of actions during the evolution of a beyond design basis accident:

—to prevent the escalation of the event into a severe accident;
—to mitigate the consequences of a severe accident; and
—to achieve a long term safe stable state.

anticipated operational occurrence. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

beyond design basis accident. Accident conditions more severe than a design basis accident.

design basis accident. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

normal operation. Operation within specified operational limits and conditions.

operational states (or operating conditions)**.** States defined under normal operation and anticipated operational occurrences.

severe accident. Accident conditions more severe than a design basis accident and involving significant core degradation.

postulated initiating event. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions

protection (or radiation protection). The protection of people from the effects of exposure to ionizing radiation, and the means for achieving this.

protective action. A protection system action calling for the operation of a particular safety actuation device.

qualified expert. An individual who, by virtue of certification by appropriate boards or societies, professional licences or academic qualifications and

experience, is duly recognized as having expertise in a relevant field of specialization, e.g. medical physics, radiation protection, occupational health, fire safety, quality assurance or any relevant engineering or safety speciality.

quality assurance. Planned and systematic actions necessary to provide adequate confidence that an item, process or service will satisfy given requirements for quality, for example, those specified in the licence.

redundancy. Provision of alternative (identical or diverse) structures, systems or components, so that any one can perform the required function regardless of the state of operation or failure of any other.

regulatory body. An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

safety culture. The assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

safety function. A specific purpose that must be accomplished for safety.

safety group. The assembly of equipment designated to perform all actions required for a particular postulated initiating event to ensure that the limits specified in the design basis for anticipated operational occurrences and design basis accidents are not exceeded.

safety limits
See limit.

safety system settings. The levels at which protective devices are automatically actuated in the event of anticipated operational occurrences or accident conditions, to prevent safety limits being exceeded.

self-assessment. A routine and continuing process conducted by management at all levels to evaluate the effectiveness of performance in all areas of their responsibility. Self-assessment activities include review, surveillance and discrete checks, which are focused on preventing, or identifying and

correcting, management problems that hinder the achievement of the organization's objectives, particularly safety objectives.

shutdown reactivity. The reactivity when all control devices are introducing their maximum negative reactivity.

single failure. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

single failure criterion. A criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure.

siting. The process of selecting a suitable site for a facility, including appropriate assessment and definition of the related design bases.

source term. The amount and isotopic composition of material released (or postulated to be released) from a facility. Used in modelling releases of radionuclides to the environment, particularly in the context of accidents at nuclear installations or releases from radioactive waste in repositories.

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