# IAEA Safety Standards

**for protecting people and the environment**

# Safety of Research Reactors

## Specific Safety Requirements

## No. SSR-3

IAEA

International Atomic Energy Agency

# IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

## IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

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

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

http://www-ns.iaea.org/standards/

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

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

## RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

# 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. SSR-3

# SAFETY OF RESEARCH REACTORS

## SPECIFIC SAFETY REQUIREMENTS

This publication includes a CD-ROM containing the IAEA Safety Glossary: 2007 Edition (2007) and the Fundamental Safety Principles (2006), each in Arabic, Chinese, English, French, Russian and Spanish versions. The CD-ROM is also available for purchase separately. See: http://www-pub.iaea.org/books

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Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
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# FOREWORD

## by Yukiya Amano
## Director General

The IAEA's Statute authorizes the Agency to "establish or adopt… standards of safety for protection of health and minimization of danger to life and property" — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For 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 regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

# THE IAEA SAFETY STANDARDS

## BACKGROUND

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

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

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

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

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

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

## THE IAEA SAFETY STANDARDS

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

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

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

**Safety Fundamentals**

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

**Safety Requirements**

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

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

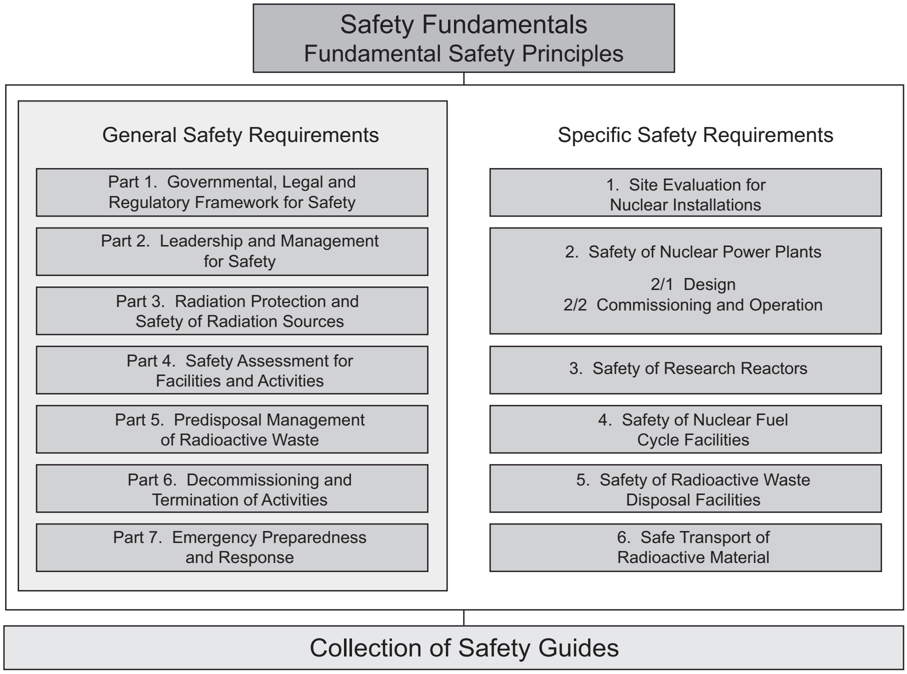


*FIG. 1. The long term structure of the IAEA Safety Standards Series.*

**Safety Guides**

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

## APPLICATION OF THE IAEA SAFETY STANDARDS

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

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

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

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

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

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

## DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of

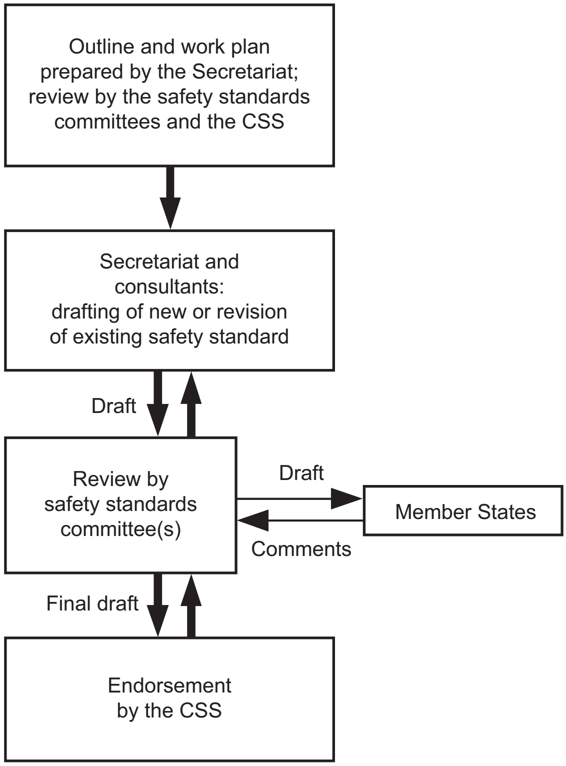


*FIG. 2. The process for developing a new safety standard or revising an existing standard.*

the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

## INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

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

## INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see 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 in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

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

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

# CONTENTS

# 1. INTRODUCTION

## BACKGROUND

1.1. This publication supersedes the Safety Requirements publication Safety of Research Reactors,[1] which was issued in 2005 as IAEA Safety Standards Series No. NS-R-4. Account has been taken of IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], published in 2006. Requirements for nuclear safety are intended to ensure the highest level of safety that can reasonably be achieved for the protection of workers and other on-site personnel, and the public, and the protection of the environment from harmful effects of ionizing radiation arising from nuclear facilities. It is recognized that technology and scientific knowledge advance, and that nuclear safety and the adequacy of protection against radiation risks need to be considered in the context of the present state of knowledge. Safety requirements will change over time; this Safety Requirements publication reflects the present international consensus.

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].

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 between the different types of research reactor,[3] including critical assemblies and

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, IAEA Safety Standards Series No. NS-R-4, IAEA, Vienna (2005).

[2] The important areas of research reactor safety include all activities performed to achieve the purpose for which the 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] A research reactor is a nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes, including experimental facilities associated with the reactor and storage, handling and treatment facilities for radioactive materials on the same site that are directly related to safe operation of the research reactor. Facilities commonly known as critical assemblies and subcritical assemblies are included.

subcritical assemblies, these requirements are to be applied in accordance with the potential hazards associated with the reactor by means of a graded approach (see paras 2.15–2.17 and IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [2]).

## OBJECTIVE

1.4. The main objective of this Safety Requirements publication is to provide a basis for safety and for safety assessment for all stages in the lifetime of a research reactor by establishing requirements on aspects relating to regulatory supervision, management for safety, site[4] evaluation, design, manufacture, construction, commissioning, operation, including utilization and modification, and planning for decommissioning.

1.5. Technical and administrative requirements for the safety of research reactors are established in accordance with this objective. This publication is intended for use by organizations involved in the design, manufacture, construction, operation, modification, maintenance and decommissioning of research reactors, in safety analysis, verification and review, and in the provision of technical support, as well as by regulatory bodies.

## SCOPE

1.6. The safety requirements established in this publication are applicable for the site evaluation, design, manufacture, construction, commissioning, operation, including utilization and modification, and decommissioning of research reactors, including critical assemblies and subcritical assemblies. The safety requirements established in this publication are also to be applied to existing research reactors to the extent practicable.

[4] Within this context, the site area is the geographical area that contains an authorized facility, authorized activity or radiation source, and within which the management of the authorized facility or authorized activity may directly initiate emergency actions. The site boundary is the perimeter of the site area.

1.7. For the purposes of this publication, a research reactor is a nuclear reactor (including critical assemblies and subcritical assemblies) used for nuclear research and for the generation and utilization of radiation for research and other purposes. This definition excludes nuclear reactors used for the production of electricity, naval propulsion, desalination or district heating. The term covers the reactor core, radioactive sources used, experimental devices[5], all systems needed for their operation, installations managed by the facility that contain nuclear material (irradiated or not), and radioactive waste management facilities and all other facilities relevant to either the reactor or its associated experimental facilities and devices located on the reactor site.

1.8. 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 and cold or hot neutron sources may require the application of supplementary measures or even the application of requirements for power reactors and/or additional safety measures (e.g. in the case of reactors used for testing hazardous material). For such facilities, the requirements (and engineering 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. Homogeneous reactors and accelerator driven systems are out of the scope of this publication.

1.9. All the requirements established here are to be applied unless it can be justified that, for a specific research reactor, critical assembly or subcritical assembly, the application of certain requirements may be graded. Each case in which the application of requirements is graded shall be identified, with account taken of the nature and possible magnitude of the hazards presented by the given facility and the activities conducted. Hereafter, subcritical assemblies will be mentioned separately only if a specific requirement is not relevant for, or is applicable only to, subcritical assemblies. Paragraph 2.17 sets out factors to be considered in deciding whether the application of certain requirements established here may be graded.

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[5] For the purposes of this safety standard, the term experimental devices includes devices 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.

1.10. This publication does not address:

(a) Requirements that are specifically covered in other IAEA Safety Requirements publications (e.g. Refs [3–7]);
(b) Matters relating to nuclear security (other than the interfaces between nuclear safety and nuclear security, addressed in Section 9) or to the State system of accounting for, and control of, nuclear material;
(c) Conventional industrial safety matters that under no circumstances could affect the safety of the research reactor;
(d) Non-radiological impacts arising from the operation of the research reactor facility.

1.11. Terms in this publication are to be understood as defined and explained in the IAEA Safety Glossary [8], unless otherwise stated (see Definitions).

## STRUCTURE

1.12. This Safety Requirements publication follows the relationship between the safety objective and safety principles, and between requirements for nuclear safety functions and design criteria and operational criteria for safety. It consists of nine sections, two appendices and two annexes. Section 2, which draws on SF-1 [1], 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. Section 3, which draws on IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [3], deals with the general requirements on legal and regulatory infrastructure as far as these are relevant for research reactors. Section 4 deals with requirements on topics relating to the management and verification of safety. This section is based on IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [4]. Section 5 establishes requirements regarding the evaluation and selection of the research reactor site and deals with the evaluation of new sites and the sites of existing research reactor facilities. This section is based on IAEA Safety Standards Series No. NS-R-3 (Rev. 1), Site Evaluation for Nuclear Installations [5]. Section 6 establishes requirements for the safe design of all types of research reactor, with account taken of the considerations mentioned in paras 1.8 and 1.9. Coherence is ensured with the Safety Requirements publication on the same subject for nuclear power plants, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [9]. Section 7 establishes requirements for the safe operation of research reactors, including commissioning, maintenance, utilization and modification. Coherence is likewise ensured with the Safety Requirements publication on the same subject for nuclear power plants, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [10]. Section 8 establishes requirements for the preparation of the safe decommissioning of research reactors on the basis of IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [11], while Section 9 establishes requirements for the interfaces between safety and security. Appendix I provides a list of the selected postulated initiating events to be considered in the safety analysis for a research reactor. Appendix II deals with the operational aspects warranting particular consideration. Annex I lists selected safety functions of the safety systems for research reactors and of other safety related items usually included in the design of research reactor. Annex II provides an overview of the application of the safety requirements to subcritical assemblies.

## 2. APPLYING THE SAFETY OBJECTIVE, CONCEPTS AND PRINCIPLES FOR RESEARCH REACTOR FACILITIES

### GENERAL

2.1. SF-1 [1] establishes the fundamental safety objective and ten fundamental safety principles that provide the basis for requirements and measures to protect people and the environment from harmful effects of ionizing radiation and for the safety of facilities and activities that give rise to radiation risks.

### FUNDAMENTAL SAFETY OBJECTIVE

2.2. The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. This fundamental safety objective has to be achieved, and the ten safety principles have to be applied, without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. To ensure that research reactors are operated and activities are conducted to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken to achieve the following (see para. 2.1 of SF-1 [1]):

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

2.3. Paragraph 2.2 of SF-1 [1] states that:

> "The fundamental safety objective applies for all facilities and activities and for all stages over the lifetime of a facility or radiation source, including planning, siting, design, manufacturing, construction, commissioning and operation [and utilization], as well as decommissioning and closure. This includes the associated transport of radioactive material and management of radioactive waste."

## FUNDAMENTAL SAFETY PRINCIPLES

2.4. Paragraph 2.3 of SF-1 [1] states that:

> "Ten safety principles have been formulated, on the basis of which safety requirements are developed and safety measures are to be implemented in order to achieve the fundamental safety objective. The safety principles form a set that is applicable in its entirety; although in practice different principles may be more or less important in relation to particular circumstances, the appropriate application of all relevant principles is required."

2.5. The requirements presented in this publication are derived from the fundamental safety objective of protecting people and the environment, and the related safety principles [1]:

Principle 1: Responsibility for safety

The prime responsibility for safety must rest with the person or organization[6] responsible for facilities and activities that give rise to radiation risks.

Principle 2: Role of government

An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.

Principle 3: Leadership and management for safety

Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.

Principle 4: Justification of facilities and activities

Facilities and activities that give rise to radiation risks must yield an overall benefit.

Principle 5: Optimization of protection

Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

Principle 6: Limitation of risks to individuals

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

Principle 7: Protection of present and future generations

People and the environment, present and future, must be protected against radiation risks.

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[6] For research reactor facilities, this is the operating organization.

Principle 8: Prevention of accidents

All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

Principle 9: Emergency preparedness and response

Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.

Principle 10: Protective actions to reduce existing or unregulated radiation risks

Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

The requirements derived from these principles must be applied to minimize and control the radiation risks to workers and other personnel, the public and the environment.

## RADIATION PROTECTION

2.6. In order to satisfy the safety principles, it is required that, for all operational states of a research reactor and for any associated activities including experiments, doses from exposure to radiation within the research reactor facility or exposure due to any planned radioactive release from the facility be kept below the dose limits and kept as low as reasonably achievable (protection and safety is required to be optimized [7]).

2.7. To apply the safety principles, it is also required that research reactors be designed and operated so as to keep all sources of radiation under strict technical and administrative control. However, these principles do not preclude limited exposures or the release of authorized amounts of radioactive substances to the environment from the research reactor facility in operational states. Such exposures and radioactive releases are required to be strictly controlled, to be recorded and to be kept as low as reasonably achievable, in compliance with regulatory and operational limits as well as radiation protection requirements.

2.8. 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 event 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 could happen. Emergency arrangements shall therefore be applied to ensure that the consequences of any accident that do occur are mitigated. Such measures and arrangements include: engineered safety features; safety features for design extension conditions; on-site emergency plans and procedures established by the operating organization; and possibly off-site emergency intervention measures put in place by the appropriate authorities in accordance with IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [6].

2.9. The safety philosophy that is followed to meet the principles stated in SF-1 [1] relies on the concept of defence in depth and on the adoption of measures for the management and verification of safety over the entire lifetime of the research reactor facility. The safety philosophy addresses the means with which the organization supports individuals and groups to perform their tasks safely, with account taken of the interactions between humans, technology and organizational aspects.

## CONCEPT OF DEFENCE IN DEPTH

2.10. The primary means of preventing accidents in a research reactor facility and mitigating the consequences of accidents if they do occur is the application of the concept of defence in depth. This concept is applied to all safety related activities, whether organizational, behavioural or design related, in all operational states.

2.11. Application of the concept of defence in depth throughout design and operation provides protection against anticipated operational occurrences and accidents, including those resulting from equipment failure or inappropriate human actions within the installation and events induced by external hazards.

2.12. Application of the concept of defence in depth in the design of a research reactor provides a series of levels of defence (based on inherent features, equipment and procedures) that are aimed at preventing accidents and ensuring adequate protection of people and the environment against harmful effects of radiation and mitigation of the consequences in the event that an accident does occur. Paragraph 3.31 of SF-1 [1] states that "The independent effectiveness of the different levels of defence is a necessary element of defence in depth."

However, the concept of defence in depth shall be applied with account taken of the graded approach. There are five levels of defence:

(1) The purpose of the first level of defence is to prevent deviations from normal operation and the failure of items important to safety. This leads to the requirement that the research reactor facility shall be soundly and conservatively sited, designed, constructed, operated and maintained, in accordance with the management system and proven 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 control of the fabrication of components and control of the construction, commissioning, operation and maintenance of the research reactor.
(2) The purpose of the second level of defence is to detect and control deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions. This is in recognition of the fact that some postulated initiating events are likely to occur at some point over the operating lifetime of the research reactor, despite the precautions taken to prevent them. This level of defence necessitates the provision of specific systems and features in the design, as determined in the safety analysis, and the establishment 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 might not be controlled at 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 research reactor facility following such events. This leads to the requirement that engineered safety features be capable of transferring the research reactor first to a controlled state and subsequently to a safe state. The radiological objective is to have no, or only a minor, off-site radiological impact.
(4) The purpose of the fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of defence. The most important objective for this level is to ensure that the confinement function is maintained, thus ensuring that radioactive releases are kept as low as reasonably achievable.

(5) The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accidents. This requires the provision of adequately equipped emergency response facilities and emergency plans and procedures for on-site and, if necessary, off-site emergency response.

2.13. A relevant aspect of the application of the concept of defence in depth for a research reactor is to include in the design a series of physical barriers, as well as a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations. The number of barriers that will be necessary will depend upon the potential source term in terms of the amount and the isotopic composition of the radionuclides, the effectiveness of the individual barriers, the possible internal and external hazards, and the potential consequences of barrier failures.

2.14. The defence in depth concept is applied mainly through the safety analysis and the use of sound engineering practices based on research and operating 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 research reactor structures, systems and components could fail and identifies the consequences of such failures. The safety analysis examines: all planned normal operational modes of the research reactor facility; and its performance in anticipated operational occurrences, design basis accident conditions and if necessary, event sequences that may lead to design extension conditions (see Requirement 22 and paras 6.64–6.68). Requirements for the safety analysis of the design are presented in paras 6.119–6.125. These analyses are independently reviewed by the operating organization and by the regulatory body (see paras 3.1–3.3).

## GRADED APPROACH

2.15. Research reactors are used for special and varied purposes, such as research, training, education, radioisotope production, neutron radiography and materials 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.

2.16. Most research reactors give rise to fewer potential hazards to the public than nuclear power plants, but they may pose greater potential hazards to operators, researchers and other users owing to the relative ease of access to radiation or radioactive materials. Qualitative categorization of the facility shall be performed on the basis of the potential hazard associated with the research reactor (see SSG-22 [2]).

2.17. The factors to be considered in deciding whether the application of certain requirements established here may be graded include:

(a) The reactor power;
(b) The potential 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 flammable materials, 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 (including those for preventing inadvertent criticality);
(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) The site evaluation, including external hazards associated with the site and the proximity to population groups;
(k) The ease or difficulty in changing the overall configuration.

# 3. REGULATORY SUPERVISION FOR RESEARCH REACTOR FACILITIES

## LEGAL AND REGULATORY INFRASTRUCTURE

3.1. GSR Part 1 (Rev. 1) [3] requires the government to ensure that an adequate legal infrastructure for a research reactor facility is established. This shall provide for the regulation of nuclear activities and for the clear assignment of responsibilities for safety in all stages in the lifetime of the facility. The government is responsible for the adoption of legislation that assigns the prime responsibility for safety to the operating organization and establishes a regulatory body. The regulatory body is responsible for the establishment of regulations that results in a system of authorization[7] for the regulatory control of nuclear activities and for the enforcement of the regulations. These principles are established in section 3 (Principles 1 and 2) of SF-1 [1].

3.2. General safety requirements to fulfil these principles are established in GSR Part 1 (Rev. 1) [3]. GSR Part 1 (Rev. 1) [3] covers the essential aspects of the governmental and legal framework for establishing a regulatory body and for taking actions necessary to ensure the effective regulatory control of facilities and activities — existing and new — utilized for peaceful purposes. Other responsibilities and functions are also covered, such as liaison within the global safety regime and liaison for providing the necessary support services for the purposes of safety (including radiation protection), emergency preparedness and response, nuclear security[8], and the State system of accounting for, and control of, nuclear material. These general safety requirements apply to the legal and governmental infrastructure for the safety of research reactors during site evaluation, design, construction, commissioning, operation, including utilization and modification, and decommissioning. The application of a graded approach that is commensurate with the potential hazards of the facility is essential and shall be used in the determination and application of adequate safety requirements (see paras 2.15–2.17).

3.3. GSR Part 1 (Rev. 1) [3] requires the government to establish and maintain an effectively independent regulatory body for the regulatory control of facilities and activities (see Requirements 3 and 4 of GSR Part 1 (Rev. 1) [3]). To be effective, the regulatory body shall be provided with the statutory legal authority and resources necessary to ensure that it can fulfil its responsibilities and fulfil its functions. This includes the authority to review and assess safety related information submitted by the operating organization during the authorization process and to apply the relevant regulations (e.g. by issuing, amending or revoking authorizations or their conditions), including carrying out compliance inspections and audits, taking enforcement actions and providing other competent authorities and the public with information, as appropriate.

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[7] Authorization to operate a facility or to conduct an activity may be granted by the regulatory body or by another governmental body to an operating organization or to a person. ‘Authorization’ includes approval, written permission, licensing, certification or registration. See Ref. [8] and Requirement 23 of GSR Part 1 (Rev. 1) [3].

[8] The IAEA issues guidance on nuclear security in the IAEA Nuclear Security Series publications.

## AUTHORIZATION PROCESS

3.4. The authorization process is ongoing, starting at the site evaluation stage and continuing up to and including the release of the facility from regulatory control. The authorization process may vary among States, but the major stages of the authorization process for nuclear research reactors shall include the following:

(a) Site evaluation;
(b) Design;
(c) Construction;
(d) Commissioning;
(e) Operation, including utilization and modification[9];
(f) Decommissioning;
(g) Release from regulatory control.

3.5. In some cases, several stages may be authorized by a single licence, but conditions are attached to it to control the subsequent stages. Despite these differences between national practices, a detailed demonstration of safety in the form of a safety analysis report that includes an adequate safety analysis shall be submitted by the operating organization to the regulatory body for review and assessment as part of the authorization process.

**Requirement 1: Safety analysis report**

**A safety analysis report shall be prepared by the operating organization for a research reactor facility. The safety analysis report shall provide a justification of the site and the design and shall provide a basis for the safe operation of the research reactor. The safety analysis report shall be reviewed and assessed by the regulatory body before the research reactor project is authorized to progress to the next stage. The safety analysis report shall be periodically updated over the research reactor's operating lifetime to reflect modifications made to the facility and on the basis of experience and in accordance with regulatory requirements.**

[9] Although the utilization and modification of research reactors are activities that are normally included under operation, they may be considered separate stages in the authorization process, since their safety implications give rise to a large number of review and assessment activities that are repeated many times over the lifetime of the reactor facility (see paras 7.98–7.106).

3.6. The safety analysis report is one of the main documents for the authorization of the research reactor facility and an important link between the operating organization and the regulatory body. The safety analysis report shall contain a detailed description of the reactor site, the reactor facility and experimental devices, and shall include all other facilities and activities with safety significance. It shall describe in detail the general safety principles and criteria applied to the design for the safety of the reactor, the protection of operating personnel[10] and the public, and the protection of the environment. The safety analysis report shall contain the analyses of the potential hazards from the operation of the reactor. The safety analysis report 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 in accordance with the defence in depth concept.

3.7. The safety analyses in the safety analysis report shall form the basis for the operational limits and conditions for the reactor. The safety analysis report shall provide details about the operating organization, the conduct of operations and the management system throughout the lifetime of the research reactor facility. The safety analysis report shall also provide information on emergency arrangements for the research reactor, although this does not preclude the need for detailed emergency arrangements, in accordance with Requirement 81.

3.8. The safety analysis report shall include information to demonstrate compliance with national legislation and requirements issued by the regulatory body. The level of detail of the information to be presented in the safety analysis report shall be determined using a graded approach. For reactors with high power levels, the safety analysis report will usually require more detail in discussions such as those of reactor design and accident scenarios. For some reactors (e.g. research reactors with low potential hazard, critical or subcritical assemblies), the requirements for the safety analysis report content may be much less extensive. However, in all cases, the safety analysis report shall cover every topic in paras 3.6–3.7.

3.9. The safety analysis report shall cite references that may be necessary for its thorough review and assessment. 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.

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

### Review and assessment by the regulatory body

3.10. A review and assessment of the information (usually in the form of a safety analysis report) submitted by the operating organization in support of its application for authorization shall be performed by the regulatory body. The specific objectives of the regulatory review and assessment are provided in GSR Part 1 (Rev. 1) [3]. The review and assessment shall be commensurate with the magnitude of the potential radiation risk associated with the research reactor facility in accordance with a graded approach. If necessary, the regulatory body may request additional information depending on national practices.

3.11. A schedule for the submission of documents for review and assessment for the stages in the authorization process shall be established early in the research reactor project and made available to the operating organization.

### Acceptance criteria

3.12. Each State shall develop its own approach to acceptance criteria depending upon its particular legal and regulatory infrastructure. Acceptance criteria based on principles for safe design and operation shall be made available to the operating organizations.

## INSPECTION AND ENFORCEMENT

3.13. Paragraph 2.5 (10) of GSR Part 1 (Rev. 1) [3] states that "an effective governmental, legal and regulatory framework for safety…shall set out…[p]rovision for the inspection of facilities and activities, and for the enforcement of regulations, in accordance with a graded approach".

3.14. Paragraph 4.50 of GSR Part 1 (Rev. 1) [3] states that:

> "The regulatory body shall develop and implement a programme of inspection of facilities and activities, to confirm compliance with regulatory requirements and with any conditions specified in the authorization. In this programme, it shall specify the types of regulatory inspection (including scheduled inspections and unannounced inspections), and shall stipulate the frequency of inspections and the areas and programmes to be inspected, in accordance with a graded approach."

3.15. Requirement 30 of GSR Part 1 (Rev. 1) [3] states that:

> "The regulatory body shall establish and implement an enforcement policy within the legal framework for responding to non-compliance by authorized parties with regulatory requirements or with any conditions specified in the authorization."

3.16. If there is evidence of a deterioration in the level of safety, or in the event of serious violations that, in the judgement of the regulatory body, could pose an imminent radiological hazard to workers and other personnel, 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 cease its activities and may suspend or revoke the authorization.

# 4. MANAGEMENT FOR SAFETY AND VERIFICATION OF SAFETY FOR RESEARCH REACTOR FACILITIES

**Requirement 2: Responsibilities in the management for safety**

**The operating organization for a research reactor facility 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, construction, commissioning, operation, including utilization and modification, and decommissioning, until its release from regulatory control.**

4.1. In order to ensure rigour and thoroughness at all levels of the staff in the achievement and maintenance of safety, the operating organization:

(a) Shall establish and implement safety policies and shall ensure that safety matters are given the highest priority;
(b) Shall clearly define responsibilities and accountabilities with corresponding lines of authority and communication;
(c) Shall ensure that it has sufficient staff with appropriate qualifications and training at all levels;

(d) Shall 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) Shall review, monitor and audit[11] all safety related matters on a regular basis, and shall take appropriate corrective actions where necessary;
(f) Shall develop and sustain a strong safety culture, and shall prepare a statement of safety policy and safety objectives, which is disseminated to and understood by all staff.

4.2. Whenever a change of stage in the lifetime of a research reactor is to be initiated by the operating organization, it shall submit a detailed demonstration of safety, 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 vendors and suppliers to ensure the availability of any information that has been requested by the regulatory body. The operating organization shall also be responsible for informing the regulatory body of any additional new information on the research reactor and of any changes to information submitted previously. All information provided by the operating organization to the regulatory body shall be complete and accurate. The format and content of documents submitted to the regulatory body by the operating organization in support of the authorization shall be based on the requirements presented in paras 3.6–3.9. The functions and responsibilities of the operating organization for ensuring safety at each stage of the lifetime of the research reactor are presented in Section 3 (see Requirement 1) and here in Section 4 as well as in the relevant paragraphs of Sections 5 to 9.

**Requirement 3: Safety policy**

**The operating organization for a research reactor facility shall establish and implement safety policies that give safety the highest priority.**

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[11] Independent assessments such as audits or surveillance are carried out to determine the extent to which the requirements for the management system are fulfilled, to evaluate the effectiveness of the management system and to identify opportunities for improvement. They can be conducted by, or on behalf of, the organization itself for internal purposes, by interested parties such as customers and regulators (or by other persons on their behalf), or by independent external organizations.

4.4. The safety policy established and implemented by the operating organization shall give safety the highest priority, overriding all other demands, including the demands of production and of reactor users. The safety policy shall promote a strong safety culture, including a questioning attitude and a commitment to excellent performance in all activities important to safety.

4.5. The safety policy shall stipulate clearly the leadership role of the highest level of management in safety matters. Senior management[12] shall be responsible for communicating and implementing the provisions of the safety policy throughout the organization. All personnel in the organization shall be made aware of the safety policy and of their responsibilities for ensuring safety. The expectations of management for safety performance shall be clearly communicated to all personnel, and it shall be ensured that such expectations are understood and followed within the organization.

4.6. The safety policy of the operating organization shall include a commitment to achieving enhancements in operational safety. The strategy of the operating organization for enhancing safety and for finding more effective ways of applying and, where feasible, improving existing standards shall be continuously monitored, periodically revised and supported by means of a clearly specified programme with clear objectives and targets.

## MANAGEMENT SYSTEM

### Requirement 4: Integrated management system

**The operating organization for a research reactor facility shall establish, implement, assess and continuously improve an integrated management system.**

[12] 'Senior management' means the person who, or group of people that, is accountable for meeting the terms established in the licence and directs, controls and assesses an organization at the highest level. Many different terms are used, including, for example: board of directors, chief executive officer, director general, executive team, plant manager, top manager, chief regulator, site vice-president, managing director and laboratory director.

4.7. The requirements for an integrated management system[13] for facilities and activities are established in GSR Part 2 [4]. These requirements and the associated objectives and principles shall be taken into account in the establishment and implementation of the management system for the research reactor by means of a graded approach on the basis of the importance to safety of each item, service or process. The level of detail of the management system that is required for a particular research reactor or experiment shall be governed by the potential hazard of the reactor and the experiment (see paras 2.15–2.17 on the graded approach and SSG-22 [2]).

4.8. The operating organization shall ensure through the establishment and use of an integrated management system that the research reactor is sited, designed, constructed, commissioned, operated and utilized (including the associated activities such as those mentioned in Appendix II), and decommissioned, in a safe manner and within the limits and conditions that are specified in the operational limits and conditions and established in the authorization.

4.9. The management system shall be developed and established at a time consistent with the schedule for accomplishing activities at all stages in the lifetime of the research reactor. In particular, activities for site investigation, which are usually initiated a long time before the establishment of a project, shall be covered by the management system.

4.10. The management system shall include all the elements of management so that processes and activities important to safety are established and conducted in accordance with relevant requirements, including those relating to leadership, protection of health, human performance, emergency preparedness and response, protection of the environment, security and quality.

4.11. The management system shall identify and include the following requirements:

(a) The statutory and regulatory requirements of the State;
(b) The relevant IAEA safety standards;

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[13] An integrated management system is a single coherent management system in which all constituents of an organization are integrated to enable the organization's objectives to be achieved. Such constituents include the organizational structure, resources and organizational processes. This system integrates all elements of management, including safety, health, environmental, security, quality, human-and-organizational-factor, societal and economic elements, so that safety is not compromised.

(c) Any requirements formally agreed with interested parties.

4.12. The documentation of the management system shall be reviewed and made subject to approval at appropriate levels of management in the operating organization and shall be submitted to the regulatory body for review and assessment as requested.

4.13. The provisions of the management system shall be based on four functional categories:

(a) Management responsibility;
(b) Management of resources;
(c) Management of processes and activities;
(d) Measurement, assessment and improvement of the management system.

**Management responsibility**

4.14. Management shall be responsible for providing the means and support needed to achieve the organization's objectives. In this regard, the management system shall include provisions for effective communication and clear assignment of responsibilities to ensure that processes and activities important to safety are controlled and performed in a manner that ensures that safety objectives are achieved.

**Resource management**

4.15. Resource management shall ensure that the resources[14] essential to the implementation of the organizational strategy and the achievement of the organization's objectives are identified and made available. The management system shall ensure that:

(a) Suppliers, manufacturers and designers of structures, systems and components important to safety have an effective integrated management system in place, with audits to confirm its effectiveness;
(b) External personnel (including suppliers and experimenters) are adequately trained and qualified and perform their activities under the same controls and to the same standards as the reactor personnel;

[14] Resources includes individuals, infrastructure, the working environment, information and knowledge, and suppliers, as well as material and financial resources.

(c) The equipment, tools, materials, hardware and software necessary to conduct the work in a safe manner are identified, provided, checked, and verified and maintained.

**Process implementation**

4.16. The management system shall include provisions for the implementation of processes to ensure that the design, including subsequent changes, modifications or safety improvements, construction, commissioning, operation and utilization, and decommissioning of the reactor are 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.

4.17. In the manufacturing and construction of structures, systems and components of the research reactor, including its associated experimental facilities and devices and modification projects, processes shall be established to ensure that the relevant regulations and safety requirements are met and that the construction work is properly implemented. Such processes shall enable the operating organization to ensure that the manufacturing and construction of items important to safety are performed in accordance with the design requirements and the regulatory requirements.

4.18. As part of the management system, processes for utilization and modification shall be established and shall be graded on the basis of their safety significance. Such processes shall include the design, review, assessment and approval, fabrication, testing and implementation of a utilization and modification project. Relevant procedures describing the processes shall be put into effect by the operating organization early in the reactor operation stage.

4.19. The management system shall ensure 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 and improvements**

4.20. The effectiveness of the management system shall be regularly measured and assessed through independent assessments and self-assessments. Weaknesses in processes shall be identified and corrected. The operating organization shall evaluate the results of such assessments and shall determine and take the necessary actions for continuous improvements.

## VERIFICATION OF SAFETY

**Requirement 5: Safety assessment**

**The adequacy of the design of the research reactor shall be verified in accordance with the management system by means of comprehensive deterministic safety analysis and complementary probabilistic analysis as appropriate and shall be validated by independent verification by individuals or groups independent from those who originally performed the design work. The safety assessment shall be continued throughout all the stages of the reactor's lifetime (in periodic safety reviews) and shall be conducted in accordance with the potential magnitude and nature of the hazards associated with the particular facility or activity.**

4.21. Verification, validation and approval of the reactor design shall be completed as soon as practicable in the design and construction processes, and in any case before commissioning of the facility is commenced.

4.22. Safety assessment[15] 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.

4.23. Safety assessment shall commence at an early stage in the design process. Deterministic safety analysis shall be the primary tool for safety assessment of research reactors. Probabilistic safety analysis may be used as a complementary tool for detecting potential weaknesses and improving the safety assessment.

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[15] Requirements for safety assessment for facilities and activities are established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [12].

4.24. The safety assessments (and periodic safety reviews) shall be documented to facilitate their evaluation.

4.25. Systematic periodic safety reviews of the research reactor in accordance with the regulatory requirements shall be performed throughout its operating lifetime, with account taken of operating experience, the cumulative effects of ageing, applicable safety standards and safety information from all relevant sources. The operating organization shall verify by analysis, surveillance, testing and inspection that the physical state of the reactor facility, including experimental devices and facilities, is as described in the safety analysis report and other safety documents, and that the facility is commissioned and operated in accordance with safety requirements and the safety analysis and operational limits and conditions.

4.26. Activities for systematic periodic safety reassessments include, among others, periodic safety reviews such as self-assessments and peer reviews[16] to confirm that the safety analysis report and other selected documents (such as documentation for operational limits and conditions, maintenance, training and qualification) for the facility remain valid in view of current regulatory requirements, or, if necessary, to update or make improvements to the extent practicable. In such reviews, changes in the site characteristics, changes in the utilization programme, cumulative effects of ageing and modifications, changes to procedures, the use of feedback from operating experience and technical developments shall be considered. It shall be verified that selected structures, systems and components and software comply with the design requirements. Specific design requirements are established in Section 6 and functional requirements are established in Section 7.

**Requirement 6: Safety committee**

**A safety committee (or an advisory group) that is independent from the reactor manager shall be established to advise the operating organization on all the safety aspects of the research reactor.**

[16] 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.

4.27. The safety committee (or advisory group) shall advise the operating organization on: (i) the safety assessment of design, commissioning and operational issues; and (ii) relevant aspects of the safety of the reactor and the safety of its utilization.[17] Members of the safety committee shall be experts in different fields associated with the design and operation of research reactors. The safety committee shall be fully functioning before the design of the research reactor begins. The list of items that the safety committee is required to consider, provide advice on, or recommend approval of shall also be established. Such a list shall include, among other things, the following:

(a) The design of structures, systems and components and in particular the design and qualification of nuclear fuel elements[18] and reactivity control elements;
(b) Safety documents and their modifications;
(c) Proposed new tests, experiments, equipment, systems or procedures that have significance for safety;
(d) Proposed modifications to items important to safety and changes in experiments that have implications for safety;
(e) Violations of the operational limits and conditions, of the licence and of procedures that are significant to safety;
(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 the safety performance of the research reactor facility;
(h) Reports on routine radioactive discharges to the environment;
(i) Reports on radiation doses to the personnel at the facility and to the public;
(j) Reports to be provided to the regulatory body;
(k) Reports on regulatory inspections.

[17] In some States, an additional safety committee (or advisory group) is established to advise the reactor manager on the safety aspects of the day to day operation and utilization of the reactor (see para. 7.26).

[18] 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. Adequate design and safety margins are established to take into account unknown behaviour of experimental fuel that is not yet qualified.

# 5. SITE EVALUATION FOR RESEARCH REACTOR FACILITIES

5.1. The main safety objective in evaluating the site for a research reactor is the protection of the public and the protection of the environment against the radiological consequences of normal and accidental releases of radioactive material (see NS-R-3 (Rev. 1) [5]). 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 research reactors with low potential hazard and critical and subcritical assemblies, the amount of detail to be provided can be substantially reduced below that required for a medium power research reactor or high power research reactor (see also paras 1.6–1.9). The results of the site evaluation shall be documented and presented in sufficient detail to permit an independent assessment by the regulatory body.

5.2. 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 and associated emergency arrangements, 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 the effects on safety and on emergency preparedness and response.

5.3. The site evaluation shall establish the boundaries of the site area, including exclusion and monitoring areas satisfying the main safety objective (see para. 5.1) and the exact location of the reactor and associated facilities (the operations area), which is under the control of the reactor management[19] and its legal rights within the site area. Any activities that are unrelated to the operation of the research reactor within these boundaries shall be evaluated and justified.

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

(a) The effects of natural and human induced external events (e.g. seismic events, fire or flooding) that may occur in the region of the site;

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[19] 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.

(b) The characteristics of the site and its environment that could influence the transfer of released radioactive material to humans;
(c) The population density and population distribution and other characteristics in the vicinity of the site having relevance to emergency arrangements, and the need to evaluate the risks to individuals and the population;
(d) Other collocated site facilities such as other research reactors, radioisotope plants, fuel cycle related facilities, post-irradiation examination or non-nuclear facilities (e.g. chemical facilities);
(e) The capability for an ultimate heat sink at the site, as appropriate;
(f) The 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 material to the environment.

5.5. If the evaluation of the site and the operations area for these six aspects, including their foreseeable evolution, indicates that deficiencies of the site or the operations area 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.)

5.6. Hazards arising from external events (or from a combination of events) shall be considered in the design of the reactor. Consideration shall be given to those cases in which anticipated operational occurrences or accidents are caused by a combination of the external and consequential internal events and where there is a need to consider long lasting external events (such as flooding) and long post-event recovery times.

5.7. Information and records relating to the occurrence and severity of important natural phenomena and postulated worst combinations of low probability but high consequence events that may exceed those conditions assumed for design basis accidents shall be collected for the region in which the potential facility site is located and shall be carefully analysed for reliability, accuracy and completeness (see paras 2.14–2.21 of NS-R-3 (Rev. 1) [5]).

5.8. During the site evaluation and before the start of construction of the research reactor, it shall be confirmed that off-site emergency arrangements, where appropriate, will be available prior to the start of reactor operation (see GSR Part 7 [6] and paras 2.26–2.29 of NS-R-3 (Rev. 1) [5]).

5.9. The external events to be considered for the site evaluation include the following (see section 3 of NS-R-3 (Rev. 1) [5]):

(a) Earthquakes, volcanoes and surface faulting;
(b) Meteorological events, including extreme values of meteorological phenomena and rare events such as lightning, tornadoes and tropical cyclones;
(c) Flooding, including water waves induced by earthquakes or other geological phenomena or floods and waves caused by the failure of water control structures;
(d) Geotechnical hazards, including slope instability, collapse, subsidence or uplift of the site surface, and soil liquefaction;
(e) Human induced external events (present and future), including security related incidents, transportation events, such as aircraft crashes, and accidents at surrounding activities, such as chemical explosions.

5.10. 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 and release from regulatory control.

5.11. Changes in site characteristics such as climate, population or use of nearby facilities that may affect the safety of the research reactor facility shall be investigated and periodically reassessed.

5.12. When a new research reactor project is planned for an existing site such as a research centre or university campus in an urban or suburban environment, the capacity of the site to accommodate a research reactor facility shall be carefully analysed to ensure regulations relating to radiation risk to site personnel and the public will be met.

## 6. DESIGN OF RESEARCH REACTOR FACILITIES

### GENERAL

6.1. The research reactor shall be designed in such a way that the fundamental safety objective (see paras 2.2 and 2.3) is 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 structures, systems and components for particular reactor types.

6.2. Application of the design requirements is an interactive process and the requirements shall be implemented throughout the design, with full consideration of the results of the safety analysis (see paras 6.119–6.125).

6.3. The achievement of a safe design requires that 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 safety analysis report.

6.4. The design of the reactor facility shall consider not only the reactor itself but also any associated facilities, such as experimental devices, that may affect safety. In addition, the reactor design shall consider the effects of the reactor on the associated facilities at all the stages of the reactor's lifetime (e.g. in terms of service conditions, electromagnetic fields and other sources of interference).

6.5. The design of the research reactor facility shall consider the different modes of operation (e.g. operation on demand rather than continuous operation, operation at different power levels, pulsed operation, operation with different core configurations, changes in the overall configuration of the reactor or assembly and operation with different nuclear fuels). In the design of the safety systems, due consideration shall be given to the stability of the reactor at different modes of operation.

## PRINCIPAL TECHNICAL REQUIREMENTS

### Requirement 7: Main safety functions

**The design for a research reactor facility shall ensure the fulfilment of the following main safety functions for the research reactor for all states of the facility: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel storage; and (iii) confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.**

6.6. A systematic approach shall be taken to identifying those items important to safety that are necessary to fulfil the main safety functions and to defining the conditions and inherent features that contribute to, or affect the fulfilment of, the main safety functions for all states of the facility.

6.7. Means of monitoring the status of the reactor facility shall be provided for ensuring that the main safety functions are fulfilled for all states of the facility.

**Requirement 8: Radiation protection**

**The design of a research reactor facility shall be such as to ensure that radiation doses to workers and other personnel at the research reactor facility and to members of the public do not exceed the established dose limits, and that they are kept as low as reasonably achievable for operational states for the entire lifetime of the research reactor facility, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions.**

6.8. The design shall ensure that facility states that could lead to high radiation doses or large radioactive releases are practically eliminated[20] and that there are no, or only minor, potential radiological consequences for facility states with a significant likelihood of occurrence.[21]

**Requirement 9: Design**

**The design of a research reactor facility shall ensure that the reactor facility and items important to safety have the appropriate characteristics to ensure that the safety functions can be performed with the necessary reliability, that the research reactor can be operated safely within the operational limits and conditions for its entire lifetime and can be safely decommissioned, and that impacts on the environment are minimized.**

6.9. The design of a research reactor shall be such as to ensure that the requirements of the operating organization, the requirements of the regulatory

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[20] The possibility of certain conditions occurring is considered to have been practically eliminated (i.e. eliminated from further consideration) if it is physically impossible for the conditions to occur or if the conditions can be considered with a high level of confidence to be extremely unlikely to arise.

[21] Requirements on radiation protection and the safety of radiation sources for facilities and activities are established in GSR Part 3 [7].

body and the requirements of relevant legislation, as well as applicable national and international codes and standards, are all met. The design shall consider human capabilities and limitations and factors that could influence human performance. Adequate information on the design shall be provided for ensuring the safe operation, utilization, maintenance and decommissioning of the reactor facility, and to allow subsequent modifications to be made and new experiments to be implemented.

6.10. The design shall take due account of relevant available experience that has been gained in the design, construction and operation of other research reactors, and of the results of relevant research and development programmes.

6.11. The design shall take due account of the results of deterministic safety analyses and, as appropriate, complementary probabilistic safety analyses to ensure that due consideration has been given to the prevention of accidents and to mitigation of the consequences of any accidents that do occur.

6.12. The design shall be such as to ensure that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume and that wastes and discharges are classified.

**Requirement 10: Application of the concept of defence in depth**

**The design of a research reactor shall apply the concept of defence in depth. The levels of defence in depth shall be independent as far as is practicable.**

6.13. The defence in depth concept (see paras 2.10–2.14) shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment, and at ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails.

6.14. The design:

(a) Shall provide for successive verifiable physical barriers to the release of radioactive material from the reactor;
(b) Shall use conservative margins, and the manufacturing and construction shall be of high quality so as to provide assurance that failures and deviations from normal operation are minimized and that accidents are prevented as far as is practicable;

(c) Shall provide for the control of reactor behaviour by means of inherent and engineered features, such that failures and deviations from normal operation requiring actuation of safety systems are minimized or excluded to the extent possible;
(d) Shall provide for automatic actuation of safety systems, such that failures and deviations from normal operation that exceed the capability of control systems can be controlled with a high level of confidence, and the need for operator actions in the early phase of such failures or deviations from normal operation is minimized;
(e) Shall provide for structures, systems and components and procedures to control the course of and, as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems;
(f) Shall provide effective means for ensuring that each of the main safety functions is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation.

6.15. To ensure that the concept of defence in depth is maintained, the design shall prevent, as far as is practicable:

(a) Challenges to the integrity of physical barriers;
(b) The failure of one or more barriers;
(c) The failure of a barrier as a consequence of the failure of another barrier;
(d) The possibility of harmful consequences of errors in operation and maintenance.

6.16. The design shall ensure, as far as is practicable, that the first, or at most the second, level of defence in depth is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the research reactor.

6.17. The levels of defence in depth shall be independent as far as practicable to avoid a failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.

**Requirement 11: Interfaces of safety with security and the State system of accounting for, and control of, nuclear material**

**Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a research reactor shall be designed and implemented in an integrated manner so that they do not compromise one another.**

**Requirement 12: Use of the graded approach**

**The use of the graded approach in application of the safety requirements for a research reactor shall be commensurate with the potential hazard of the facility and shall be based on safety analysis and regulatory requirements.**

6.18. The use of a graded approach in the application of the safety requirements shall not be considered as a means of waiving safety requirements and shall not compromise safety. Grading of the application of requirements shall be justified and supported by safety analysis or engineering judgement.

**Requirement 13: Proven engineering practices**

**Items important to safety for a research reactor shall be designed in accordance with the relevant national and international codes and standards.**

6.19. Items important to safety shall preferably be of a design that has previously been proven in equivalent applications, and if not, shall be items of high quality and of a technology that has been qualified and tested.

6.20. National and international codes and standards that are used as design rules for items important to safety shall be identified and evaluated to determine their applicability, adequacy and sufficiency, and shall be supplemented or modified as necessary to ensure that the quality of the design is commensurate with the associated safety function.

6.21. Codes and standards applicable to structures, systems and components shall be identified and their use shall be in accordance with the classification of the structures, systems and components (see paras 6.29 and 6.32). In particular, if different codes and standards are used for different types of item (e.g. for piping or for electrical systems), consistency between the codes and standards shall be demonstrated.

6.22. In the case of structures, systems and components for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment having similar environmental and operational requirements 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. The use of such a results based approach shall be justified.

6.23. Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, a process shall be established under the management system to ensure that safety is demonstrated by means of appropriate supporting research programmes, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications. The new design or feature or new practice shall be adequately tested to the extent practicable before being brought into service, and shall be monitored in service to verify that the behaviour of the reactor facility is as expected.

6.24. Acceptance criteria shall be established for operational states and for accident conditions. In particular, the design basis accidents considered in the design of the research reactor and selected design extension conditions shall be identified for the purposes of establishing acceptance criteria. For the design of structures, systems and components, acceptance criteria may be applied in the form of engineering design rules. These rules may include requirements in relevant codes and standards established in the State or internationally. The acceptance criteria shall be reviewed by the regulatory body.

**Requirement 14: Provision for construction**

**Items important to safety for a research reactor facility shall be designed so that they can be manufactured, constructed, assembled, installed and erected in accordance with established processes that ensure the achievement of the design specifications and the required level of safety.**

6.25. In the provision for construction, due account shall be taken of relevant experience that has been gained in the construction of similar facilities and their associated structures, systems and components. Where good practices from other relevant industries are adopted, such practices shall be shown to be appropriate to the specific nuclear application.

6.26. The construction shall start only after the operating organization has verified that the main safety issues in the design have been resolved and after the regulatory body has granted an authorization (e.g. a construction licence or an authorization for modification). The responsibility for ensuring that the construction is in accordance with the design lies with the operating organization.

**Requirement 15: Features to facilitate radioactive waste management and decommissioning**

**Special consideration shall be given at the design stage of a research reactor facility to the incorporation of features to facilitate radioactive waste management and the future decommissioning of the facility.**

6.27. In particular, the design shall take due account of:

(a) The choice of materials so that amounts of radioactive waste will be minimized to the extent practicable and decontamination will be facilitated;
(b) The access capabilities and the means of handling that might be necessary;
(c) The facilities necessary for the processing (i.e. pretreatment, treatment and conditioning) and storage of radioactive waste generated in operation and provision for managing the radioactive waste that will be generated in the decommissioning of the research reactor facility.

6.28. This requirement shall also be considered in the design of any modifications, new utilizations and experiments.

## GENERAL REQUIREMENTS FOR DESIGN

**Requirement 16: Safety classification of structures, systems and components**

**All items important to safety for a research reactor facility shall be identified and shall be classified on the basis of their safety function and their safety significance.**

6.29. The method for classifying the safety significance of items important to safety[22] shall be based primarily on deterministic methods complemented, where

[22] Safety classification reflects the significance for nuclear safety of the structures, systems and components. Its purpose is to establish a grading in the application of the requirements for design. There are other possible classifications or categorizations of structures, systems and components in accordance with other aspects (e.g. seismic or environmental qualification, or quality categorization of structures, systems and components).

appropriate, by probabilistic methods (if available), with due account taken of factors such as:

(a) The safety function(s) to be performed by the item;
(b) The consequences of failure to perform a safety function;
(c) The frequency with which the item will be called upon to perform a safety function;
(d) The time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function.

6.30. The design shall be such as to ensure that any interference between items important to safety will be prevented, and in particular that any failure of items important to safety in a system in a lower safety class will not propagate to a system in a higher safety class.

6.31. Equipment that performs multiple functions shall be classified in a safety class that is consistent with those functions having the highest safety significance.

6.32. Structures, systems and components, including software, that are important to safety shall be first identified and then classified in accordance with their function and significance for safety. The basis for the safety classification of the structures, systems and components shall be stated and the design requirements shall be applied in accordance with their safety classification.

**Requirement 17: Design basis for items important to safety**

**The design basis for items important to safety for a research reactor facility shall specify the necessary capability, reliability and functionality for the relevant operational states, for accident conditions and for conditions arising from internal and external hazards, to meet the specific acceptance criteria over the lifetime of the research reactor.**

6.33. The design basis for each item important to safety shall be systematically justified and documented. The documentation shall provide the necessary information for the operating organization to operate the reactor safely.

6.34. The challenges that the reactor may be expected to face during its operating lifetime shall be taken into consideration in the design process. These challenges include all the foreseeable conditions and events relating to stages in the operating lifetime of the reactor and to operational states and accident conditions, site characteristics and modes of operation.

**Requirement 18: Postulated initiating events**

**The design for the research reactor shall apply a systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design.**

6.35. Postulated initiating events shall be selected appropriately for the purpose of analysis (see Appendix I). 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.

6.36. The postulated initiating events shall be identified on the basis of engineering judgement, operating experience feedback and deterministic assessment, complemented, where appropriate and available, by probabilistic methods.

6.37. The postulated initiating events shall include all foreseeable failures of structures, systems and components of the reactor facilities and experiments as well as operating errors and possible failures arising from internal and external hazards for all operational and shutdown states.

6.38. An analysis of the postulated initiating events shall be made to establish the preventive and protective measures that are necessary to ensure that the required safety functions will be performed.

6.39. The expected behaviour of the reactor in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority:

(1) A postulated initiating event would produce no safety significant effects and would produce only a change towards a safer and more stable condition by means of inherent safety characteristics of the reactor.
(2) Following a postulated initiating event, the reactor would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event.

(3) Following a postulated initiating event, the reactor would be rendered safe by the actuation of active items important to safety that need to be brought into operation in response to the postulated initiating event.
(4) Following a postulated initiating event, the reactor would be rendered safe by following specified procedures.

6.40. The postulated initiating events used for developing the performance requirements for the items important to safety in the overall safety assessment and the detailed analysis of the reactor facility shall be grouped into representative event sequences that identify bounding cases and that provide the basis for the design and the operational limits for the items important to safety.

6.41. A technically supported justification shall be provided for the exclusion from the design of any initiating event that is identified in accordance with the comprehensive set of postulated initiating events.

6.42. Where prompt and reliable action would be necessary in response to a postulated initiating event, provision shall be made in the design for automatic safety actions for the actuation of safety systems to prevent progression to more severe reactor conditions.

6.43. Where prompt action in response to a postulated initiating event would not be necessary, it is permissible for reliance to be placed on the manual initiation of systems or on other operator actions. For such cases, the time interval between detection of the postulated initiating event or accident and the required action shall be sufficiently long, and adequate procedures (such as administrative, operational and emergency procedures) shall be specified to ensure the performance of such actions. An assessment shall be made of the potential for an operator to worsen an event sequence through erroneous operation of equipment or incorrect diagnosis of the necessary recovery process.

6.44. The operator actions necessary to diagnose the state of the reactor following a postulated initiating event and to put it into a stable long term shutdown condition in a timely manner shall be facilitated by the provision in the design of adequate instrumentation to monitor the status of the reactor, and adequate means for the manual operation of equipment.

**Requirement 19: Internal and external hazards**

**All foreseeable internal hazards and external hazards for a research reactor, including the potential for human induced events directly or indirectly to**

**affect the safety of the research reactor, shall be identified and their effects, both individually and in credible combinations, shall be evaluated. Hazards shall be considered in designing the layout of the facility and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the reactor facility.**

6.45. Items important to safety shall be designed and located with due consideration of other implications for safety, to withstand the effects of hazards or to be protected, in accordance with their importance to safety, against hazards and against common cause failure mechanisms generated by hazards. This also applies to non-permanent equipment.

**Internal hazards**

6.46. 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.

6.47. The potential for internal hazards such as fires and explosions, 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 combinations of external and internal events shall also be considered in the design where appropriate.

**Fire and explosion**

6.48. Structures, systems and components 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:

(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 safety of the facility. Internal fires and explosions shall not challenge redundant trains of safety systems.

6.49. Firefighting systems shall be automatically initiated where necessary. Firefighting systems shall be designed and located so as to ensure that their use or rupture or spurious or inadvertent operation would not increase the risk of criticality[23], would not harm personnel, would not significantly impair the capability of structures, systems and components 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 6.76–6.79).

6.50. Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the research reactor facility (including for tests and experiments), 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 the minimum necessary amounts and shall be stored in adequate facilities to keep reactive substances segregated.

6.51. Fires and explosions shall not prevent achievement of the main safety functions as well as monitoring the status of the facility. These shall be maintained by means of the appropriate incorporation of redundant structures, systems and components, diverse systems, physical separation and design for fail-safe operation.

**External events**

6.52. 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).

6.53. Natural external events shall be addressed, including meteorological, hydrological, geological and seismic events, and all credible combinations thereof (see para. 6.69). Human induced external events arising from nearby

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[23] This aspect is important in particular for critical assemblies and subcritical assemblies and dry fuel storage facilities, which shall be designed to be safely subcritical following activation of the fire protection system and during firefighting activities.

industries and transport routes shall be addressed. In the short term, the safety of the facility shall not be dependent on the availability of off-site services such as the electricity supply and firefighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.

6.54. A research reactor facility located in a seismically active region shall be equipped with a seismic detection system that actuates the automatic reactor shutdown systems if a specified threshold value is exceeded.

6.55. Features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and instrumentation and control cabling) and any other structure as a result of external events considered in the design.

6.56. The design shall be such as to ensure that all items important to safety are capable of withstanding the effects of external events considered in the design, and if not, other features such as passive barriers shall be provided to protect the reactor facility and to ensure that the main safety functions will be achieved.

6.57. The design shall provide for an adequate margin to protect items important to safety against levels of external hazards more severe than those selected for the design basis, derived from the site hazard evaluation.

**Requirement 20: Design basis accidents**

**A set of accident conditions that are to be considered in the design for a research reactor shall be derived from postulated initiating events for the purpose of establishing the boundary conditions for the research reactor to withstand, without acceptable limits for radiation protection purposes being exceeded.**

6.58. Design basis accidents shall be used to define the design bases, including performance criteria, for safety systems and for other items important to safety that are necessary to control design basis accident conditions, with the objective of returning the reactor to a safe state and mitigating the consequences of any accident.

6.59. The design shall be such that for design basis accident conditions, key reactor parameters do not exceed the specified design limits. A primary objective shall be to manage all design basis accidents so that they have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site emergency response actions.

6.60. 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. The design shall reduce demands on the operator as far as reasonably practicable, in particular during and following a design basis accident.

6.61. The design basis accidents shall be analysed in a conservative manner. This approach involves the application of the single failure criterion (see Requirement 25) to safety systems, specifying design criteria and using conservative assumptions, models and input parameters in the analysis.

6.62. The design of subcritical assemblies shall include technical provisions to prevent criticality (see para. 6.66).

**Requirement 21: Design limits**

**A set of design limits for a research reactor consistent with the key physical parameters for each item important to safety for the research reactor shall be specified for all operational states and for accident conditions.**

6.63. The design limits shall be specified for each operational state of the reactor and its experimental devices and shall be consistent with relevant national and international standards and codes, as well as with relevant regulatory requirements.

**Requirement 22: Design extension conditions**

**A set of design extension conditions for a research reactor shall be derived for the purpose of enhancing the safety of the research reactor by enhancing its capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. The set of design extension conditions shall be derived on the basis of engineering judgement and by using a graded approach, deterministic assessments and complementary probabilistic assessments, as appropriate. The design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences if they do occur.**

6.64. An analysis of design extension conditions shall be performed[24] to determine whether the potential radiological consequences would exceed those deemed acceptable by the relevant authority. The main technical objective of considering the design extension conditions is to provide assurance that the design of the facility is such as to prevent accident conditions beyond those considered design basis accident conditions, or to mitigate their consequences, as far as is reasonably practicable. This might require additional safety features for design extension conditions, or extension of the capability of safety systems to maintain the main safety functions, especially the confinement function[25]. These additional safety features for design extension conditions, or this extension of the capability of safety systems, shall be such as to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material confined in the facility (including radioactive material resulting from degradation of the reactor core).

6.65. The design extension conditions shall be used to define the design specifications for safety features and for the design of all other items important to safety that are necessary for preventing such conditions from arising, or, if they do arise, for controlling them and mitigating their consequences. For existing research reactors, complementary safety reassessment shall be performed to determine whether there is a need for mitigatory measures or modifications of the facility to be implemented.

6.66. For subcritical assemblies, the likelihood of criticality shall be sufficiently remote to be considered a design extension condition. To ensure subcriticality, the design shall include safety provisions such as the use of only natural uranium

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[24] The analysis of design extension conditions could be performed by means of a best estimate approach (more stringent approaches may be used according to States' requirements).

[25] Confinement is the prevention or control of releases of radioactive material to the environment in operation or in accidents [8]. 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 design extension conditions. The function of confinement is usually fulfilled by means of several barriers surrounding the main parts of a nuclear reactor that contain radioactive material. For a research reactor, the reactor building may be 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.

or limited amounts of fissile materials, or a fixed fuel/moderator ratio. If no such provisions can be provided, measures for mitigating the consequences shall be determined and implemented on the basis of safety analysis.

6.67. The analysis undertaken shall include identification of the safety features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features:

(a) Shall be independent, to the extent practicable, of those used in more frequent accidents;
(b) Shall be capable of performing, to the extent practicable, in the environmental conditions pertaining to design extension conditions, as appropriate;
(c) Shall be reliable commensurate with the function that they are required to fulfil.

6.68. The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release[26] is practically eliminated. The design shall be such that for design extension conditions, protective measures that are limited in terms of times and areas of application shall be sufficient for protection of the public, and sufficient time shall be available to take such measures.

**Combinations of events and failures**

6.69. Where the results of engineering judgement and deterministic safety assessments, complemented, as appropriate, by probabilistic safety assessments, indicate that combinations of postulated initiating events could lead to accident conditions, such combinations of events shall be considered to be design basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence. Certain events might be consequences of other events, such as a flood following an earthquake. Such consequential effects shall be considered to be part of the original postulated initiating event.

[26] An early radioactive release is a release for which off-site protective measures are necessary but are unlikely to be fully effective in due time. A large radioactive release is a release for which off-site protective measures limited in terms of times and areas of application are insufficient to protect people and the environment.

**Requirement 23: Engineered safety features**

**Engineered safety features shall be provided for a research reactor to prevent anticipated operational occurrences and design basis accidents and to mitigate their consequences, if they occur.**

6.70. Examples of engineered safety features for a research reactor are an emergency core cooling system and means of confinement (in particular, an emergency ventilation system). Specific requirements on these systems and their supplementary features are established in paras 6.128–6.137 and 6.164–6.166. Other engineered safety features, such as a second shutdown system or a containment structure, shall also be designed in accordance with these requirements.

6.71. The necessity and capabilities for engineered safety features shall be determined from the safety analysis. The accidents where these systems are required to 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.

6.72. 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 (including reliability of supporting and auxiliary systems necessary for operating the engineered safety features, see Requirement 60), independence, redundancy, fail-safe characteristics, diversity and physical separation of redundant systems, preference of passive systems over active systems, and functional separation of redundant safety systems;
(b) The use of materials to withstand the postulated accident conditions (e.g. in relation to radiation levels or radiolytic decomposition);
(c) Provisions for maintenance, periodic testing and inspection (including under simulated design basis accident conditions where possible) to verify that the engineered safety features continue to function or are in a state of readiness to perform their functions reliably and effectively upon demand.

**Requirement 24: Reliability of items important to safety**

**The reliability of items important to safety for a research reactor facility shall be commensurate with their safety significance.**

6.73. The design of items important to safety shall be such as to ensure that the equipment can be qualified, procured, installed, commissioned, operated and maintained to be capable of withstanding, with sufficient reliability and effectiveness, all conditions specified in the design basis for the items.

6.74. In the selection of equipment, consideration shall be given to both spurious operation and unsafe failure modes. Preference shall be given in the selection process to equipment that exhibits a predictable and revealed mode of failure and for which the design facilitates repair or replacement.

6.75. Maximum authorized unavailability limits for operation of the research reactor shall be established for items important to safety to ensure the reliable performance of safety functions. The unavailability limits shall be documented in the operational limits and conditions.

**Requirement 25: Single failure criterion**

**The single failure criterion shall be applied to each safety group incorporated in the design of the research reactor.**

6.76. Spurious action shall be considered to be one mode of failure when applying the single failure criterion to a safety group or safety system.

6.77. The design shall take due account of the failure of a passive component, unless it has been justified in the single failure analysis with a high level of confidence that a failure of that component is very unlikely and that its function would remain unaffected by the postulated initiating event.

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

6.79. 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[27].

**Requirement 26: Common cause failures**

**The design of equipment for a research reactor facility shall take due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence have to be applied to achieve the necessary reliability.**

6.80. The principle of diversity shall be adopted wherever practicable, after consideration of its possible disadvantages from complications in operating, maintaining and testing the diverse equipment.

**Requirement 27: Physical separation and independence of safety systems**

**Interference between safety systems or between redundant elements of a system for a research reactor facility shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication (data transfer), as appropriate.**

**Requirement 28: Fail-safe design**

**The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety for a research reactor.**

6.81. Systems and components important to safety shall be designed for fail-safe behaviour, as appropriate, so that their failure or the failure of a support feature does not prevent the performance of the intended safety function.

**Requirement 29: Qualification of items important to safety**

**A qualification programme shall be implemented for a research reactor facility to verify that items important to safety are capable of performing their intended functions when necessary, and in the prevailing environmental**

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[27] 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.

**conditions, throughout their design life, with due account taken of reactor conditions during maintenance and testing.**

6.82. Any environmental and service conditions that could reasonably be anticipated and that could arise in specific operational states shall be included in the qualification programme.

6.83. The environmental conditions considered in the qualification programme for items important to safety at a research reactor shall include the variations in ambient environmental conditions that are anticipated in the anticipated operational occurrences and the design basis accidents for the facility.

6.84. The qualification programme for items important to safety shall include the consideration of ageing effects caused by environmental factors (such as conditions of vibration, irradiation, humidity or temperature) over the expected service life of the items important to safety. When the items important to safety are subject to natural external events and are required to perform a safety function during or following such an event, the qualification programme shall replicate as far as is practicable the conditions imposed on the items important to safety by the natural event, either by test or by analysis or by a combination of both.

**Requirement 30: Design for commissioning**

**The design for a research reactor facility shall include features as necessary to facilitate the commissioning process for the reactor facility, including experimental facilities. These design features may include provisions to operate with transition cores of different characteristics.**

6.85. The provision for the installation and removal of additional equipment needed only for commissioning, such as filters, filling and draining provisions, and instrumentation, shall be considered in the design.

**Requirement 31: Calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety**

**Items important to safety for a research reactor facility shall be designed to be calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions specified in their design basis.**

6.86. Items important to safety shall be designed to allow for appropriate functional testing to ensure that these items will perform their safety functions with the required reliability and shall be arranged so that they can be adequately tested and maintained as appropriate, before commissioning and at regular intervals thereafter, in accordance with their importance to safety.

6.87. 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 tests. Where possible and appropriate, self-testing circuits shall be installed in electrical and electronic systems.

6.88. Items important to safety shall be designed and arranged so they can be adequately inspected, tested, maintained and replaced as appropriate. The layout of the reactor shall be such that activities for calibration, testing, maintenance, repair or replacement, inspection and monitoring are facilitated and can be performed in accordance with relevant national and international codes and standards 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.

6.89. Provision shall be made in the design of the reactor to facilitate maintainability and the replacement of items important to safety as well as to facilitate routine in-service inspection.

**Requirement 32: Design for emergency preparedness and response**

**For emergency preparedness and response purposes, the design for a research reactor facility shall provide:**

- **(a) A sufficient number of escape routes, clearly and durably marked, with reliable emergency lighting, ventilation and other services essential to the safe use of these escape routes;**
- **(b) Effective means of communication throughout the facility for use following all postulated initiating events and in accident conditions.**

6.90. 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 national requirements for radiation zoning, fire protection, industrial safety and nuclear security (see also Section 9), and shall take into account the relevant international requirements, as applicable.

6.91. Suitable alarm systems and means of communication shall be provided so that all persons present at the reactor facility and on the site can be given warnings and instructions in an emergency. The availability of reliable and diverse means of communication necessary for safety within the reactor facility[28] shall be ensured at all times, with due account taken of postulated initiating events that may compromise their availability.

**Requirement 33: Design for decommissioning**

**Decommissioning of a research reactor facility shall be considered in the design for the research reactor and its experimental facilities.**

6.92. In the design of the research reactor and its experimental facilities and in any modifications of them, consideration shall be given to facilitation of decommissioning [11]. In accomplishing this, the following shall be considered:

(a) The selection of materials so as to minimize activation of the materials with regard to decommissioning and radioactive waste management and to provide for easy decontamination;
(b) Optimizing of the facility's layout and access routes to facilitate the removal of large components and the detachment and handling (remotely where necessary) of activated components;
(c) The predisposal management of radioactive waste (i.e. pretreatment, treatment, conditioning and storage of waste arising from operation and decommissioning of the reactor).

6.93. Full details shall be retained of the design requirements and of information relating to the site and its final design, construction and modification, such as the 'baseline' radiological characterization, as built drawings relating to the facility's layout, piping and cable penetrations, as information necessary for decommissioning.

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[28] Including means of communication within the supplementary control room, if one exists.

**Requirement 34: Design for radiation protection**

**Provision shall be made at a research reactor facility for ensuring that doses to operating personnel, reactor users (experimenters) and the public will be maintained below the prescribed dose limits and will be kept as low as reasonably achievable, and that the relevant dose constraints will be taken into consideration.**

6.94. In accordance with the fundamental safety objective of protecting people and the environment (see para. 2.1. of SF-1 [1]) for all operational states and accident conditions, adequate provision shall be made in the design, on the basis of the radiation protection programme, 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.

6.95. 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 design basis accidents.

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

6.97. The design shall include any necessary provisions to segregate materials in accordance with their radiological, physical and chemical characteristics, to facilitate their handling and to protect workers and other personnel at the facility and the public by means of access control. This shall be accomplished by establishing zones within the facility (in supervised areas and controlled areas, see Requirement 24 of GSR Part 3 [7]) that are classified in accordance with their hazard potential. Such zones shall be clearly delineated and designated. Surfaces shall be appropriately designed to facilitate their decontamination.

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

6.99. Provision shall be made for controlling the release and preventing the dispersion of radioactive substances and contamination at the facility. Ventilation systems with appropriate filtration shall be provided for use in operational states and accident conditions.

6.100. 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 structures, systems and components to limit exposure during maintenance, testing and inspection by providing shielding from direct and scattered radiation, and the provision of means of monitoring and controlling access to the reactor and its experimental devices and facilities.

6.101. Provision shall be made in the design for safe handling of the radioactive waste generated at the research reactor facility. Provision shall be made for appropriate decontamination facilities for both personnel and equipment and for handling the radioactive waste arising from decontamination activities.

6.102. Equipment subject to frequent maintenance or manual operation shall be located in areas of low dose rate to reduce the exposure of workers and other personnel at the facility.

**Requirement 35: Design for optimal operator performance**

**Systematic consideration of human factors, including the human–machine interface, shall be applied at an early stage in the design process for a research reactor facility, including its experimental facilities, and shall be continued throughout the entire design process.**

6.103. Consideration shall be given in design to ensuring that, if reliance on administrative controls and procedures is necessary, such controls are feasible and the associated procedures are applicable.

6.104. Consideration shall be given to human factors and the application of ergonomic principles in the design of the control room and reactor systems.

6.105. The human–machine interface shall be designed to provide the operators with comprehensive but easily manageable information, in accordance with the necessary decision times and action times. The information necessary for the operator to make a decision to act shall be simply and unambiguously presented and shall enable:

(a) Assessment of the general state of the facility in any condition;
(b) Operation of the facility within the specified limits on parameters associated with facility systems and equipment (operational limits and conditions);
(c) Confirmation that safety actions for the actuation of safety systems are automatically initiated when needed and that the relevant systems perform as intended;
(d) Determination of both the need for and the time for manual initiation of the specified safety actions.

6.106. 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.

6.107. The design shall support operating personnel in the performance of their tasks and shall limit the effects of operating errors on safety. Due consideration shall be given in the design process to the layout of the facility and equipment, and to procedures, including procedures for maintenance and inspection, for facilitating intervention of the operating personnel on the reactor structures, systems and components in all states of the research reactor.

**Requirement 36: Provision for safe utilization and modification**

**The design for a research reactor facility shall include provisions for the safe utilization and modification of the research reactor.**

6.108. Research reactors are operationally flexible in nature and they may be in various different states. Precautions shall be taken in the design regarding the utilization and modification of the research reactor to ensure that the configuration of the reactor is known at all times. In particular, 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.

6.109. Every proposed modification to the reactor or to an experiment that may have a major significance for safety shall be designed in accordance with the same principles as apply for the reactor itself (see paras 7.100–7.101). In particular, all experimental devices 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.

6.110. Modifications of research reactors and experimental devices shall be designed such that the means of confinement and shielding of the reactor is preserved. Protection systems for experimental devices shall be designed to protect both the device and the reactor. A formal commissioning programme shall be established for experiments and modifications with major safety significance.

6.111. 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.

**Requirement 37: Design for ageing management**

**The design life of items important to safety at a research reactor facility shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, such as neutron embrittlement and wear-out, and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions in operational states and accident conditions in case of demand throughout their design life. The life cycles of the technology utilized and the possible obsolescence of the technology shall be considered.**

6.112. The design for a research reactor shall take due account of physical ageing, the effects of wear and tear and obsolescence in all operational states for which a component is credited, including testing, maintenance, and operational states during and following a postulated initiating event.

6.113. An ageing management programme that includes 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.

6.114. The design shall include provisions for the necessary monitoring, testing, sampling and inspection for the detection, assessment, prevention and mitigation of ageing effects. The ageing management of the research reactor facility shall include the management of obsolete structures, systems and components and the management of spare parts.

**Requirement 38: Provision for long shutdown periods**

**In the design of the research reactor facility, consideration shall be given to ensuring the safety of the facility in long shutdown periods.**

6.115. Provision shall be made in the design to meet the needs arising in long shutdown periods, such as the need for maintaining the conditions of the nuclear fuel, the coolant or the moderator and cover gas, the need for appropriate preservation of structures, systems and components, and the need for the maintenance, periodic testing and inspection of the relevant structures, systems and components. Consideration shall be given to long lived neutron poisoning of the reflector material, which may affect the restarting of the reactor.

**Requirement 39: Prevention of unauthorized access to, or interference with, items important to safety**

**Unauthorized access to, or interference with, items important to safety at a research reactor facility, including computer hardware and software, shall be prevented.**

6.116. Provision shall be made in the design for the control of access to the reactor facility and/or to equipment by operating personnel and reactor users, including emergency workers and vehicles, with particular consideration given to the prevention of any unauthorized entry of persons and goods to the site or to buildings on the site, for the main purposes of preventing theft or the unauthorized removal of nuclear material and preventing sabotage (see also Section 9).

**Requirement 40: Prevention of disruptive or adverse interactions between systems important to safety**

**The potential for disruptive or adverse interactions between systems important to safety at a research reactor facility that might be required to**

**operate simultaneously shall be evaluated, and any disruptive or adverse interactions shall be prevented.**

6.117. In the analysis of the potential for disruptive or adverse interactions of systems important to safety, due account shall be taken of physical interconnections and of the possible effects of one system's operation, spurious operation or malfunction on local environmental conditions of other systems, to ensure that changes in environmental conditions do not affect the reliability of systems or components in functioning as intended.

6.118. If two systems important to safety and containing fluid are interconnected and are operating at different pressures, either the systems shall both be designed to withstand the higher pressure, or provision shall be made to prevent the design pressure of the system operating at the lower pressure from being exceeded.

**Requirement 41: Safety analysis of the design**

**A safety analysis of the design for a research reactor facility shall be conducted in which methods of deterministic analysis and complementary probabilistic analysis as appropriate shall be applied to enable the challenges to safety in all facility states to be evaluated and assessed.**

6.119. A safety analysis shall be conducted of the design of the research reactor. The safety analysis shall include the response of the facility to a range of postulated initiating events (such as malfunctions or failures of equipment and experimental devices, operator errors or external and internal events) that could lead either to anticipated operational occurrences or to accident conditions (see also GSR Part 4 (Rev. 1) [12]). These analyses shall be used:

(a) As the design basis for items important to safety;
(b) For the selection of the operational limits and conditions for the reactor;
(c) For the development of operating procedures, inspection and periodic testing programmes, record keeping practices, maintenance schedules, proposals for modifications and emergency planning.

6.120. The safety analysis shall provide assurance that defence in depth has been implemented and uncertainties have been given adequate consideration in the design.

6.121. 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 design basis accidents is possible by means of an automatic response of safety systems in combination with prescribed operator actions;
(e) Specification of the design extension conditions and of how they are addressed;
(f) Determination of the operational limits and conditions for normal operation;
(g) The analysis of safety systems and the engineered safety features and the safety features for design extension conditions;
(h) The analysis of the means of confinement.

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

(a) The input parameters, initial conditions, boundary conditions, assumptions, models, uncertainties 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 derivation of source terms;
(h) The evaluation of radiological consequences.

6.123. For each accident sequence considered, the extent to which the safety systems and any operable process systems are required to function under accident 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 shall provide input to the design of the safety systems and the definition of their functions.

6.124. Where applicable, the safety analysis shall include consideration of the experimental devices with regard to both their own safety aspects and their effects on the research reactor.

6.125. The applicability of the methods of analysis, the analytical assumptions and the degree of conservatism used in the design of the research reactor shall be updated and verified for the as built facility.

## SPECIFIC REQUIREMENTS FOR DESIGN

### Buildings and structures

**Requirement 42: Buildings and structures**

**The buildings and structures important to safety for a research reactor facility shall be designed to keep radiation levels and radioactive releases on and off the site as low as reasonably achievable and below authorized limits for all operational states, for design basis accidents and, as far as practicable, for design extension conditions.**

6.126. The buildings and structures important to safety shall be designed for all operational states, for design basis accidents and, as far as practicable, for design extension conditions.

6.127. The required 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.

**Requirement 43: Means of confinement**

**Means of confinement shall be provided for a research reactor to ensure, or to contribute to, the fulfilment of the following safety functions: (i) confinement of radioactive substances in operational states and in accident conditions; (ii) protection of the reactor against natural external events and human induced events; and (iii) radiation shielding in operational states and in accident conditions.**

6.128. Means of confinement (see footnote 25) shall be designed to ensure that a release of radioactive material (fission products and activation products) following an accident involving disruption or damage of the nuclear fuel, core components or experimental devices 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 an unplanned release of radioactive material in operational states or to mitigate its consequences if one does occur, in design basis accidents and, to the extent practicable, in design extension conditions. 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.

6.129. 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 through the barrier. In setting this pressure, variations in atmospheric conditions (e.g. wind speed and atmospheric pressure) shall be taken into account.

6.130. In the design of the means of confinement, the effects of extreme conditions (e.g. pressure waves or explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events listed in Appendix I, as relevant, (e.g. fire conditions and the associated increases in local pressures) shall be taken into account.

6.131. Barriers shall be designed to withstand, with suitable margins, the highest calculated pressure and temperature loads expected in design basis accident conditions.

6.132. The release rate under accident conditions and associated consequences shall be determined, with account taken of the source term and other parameters such as extent of filtration, the point of release, environmental conditions, and the pressure and temperature under design basis accident conditions.

6.133. In the event of an accident (including an accident that may produce an increase in pressure), the leakage from the barrier shall be controlled by means of appropriate engineering features to prevent the release of radioactive material to the environment in excess of acceptable limits.

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

6.135. Where confinement is dependent on the efficiency of filters, the design shall include provisions as appropriate for in situ periodic testing of the efficiency of the filters.

6.136. For structures and components performing the function of confinement, coverings and coatings shall be such as to ensure their safety functions and to minimize interference with other safety functions in the event of their deterioration.

6.137. For research reactors that have greater potential hazards associated with them, a containment structure shall ensure that, in design basis accident conditions, any release of radioactive material would be kept below authorized limits and that, in design extension conditions, any release of radioactive material would be kept below acceptable limits.

**Reactor core and associated features**

**Requirement 44: Reactor core and fuel design**

**Research reactor core components and fuel elements and assemblies for a research reactor shall be designed to maintain their structural integrity and to withstand satisfactorily the conditions in the reactor core in all operational states and in design basis accident conditions.**

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

6.139. 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 shall be given in the design of the fuel elements to the requirements relating to the long term management of irradiated elements, which may include reprocessing or conditioning for disposal.

6.140. All foreseeable reactor core configurations, including the initial core configuration through to the equilibrium core configuration, as appropriate, shall be considered in the core design. The effect of the inserted experimental devices or materials under irradiation shall also be considered. For subcritical assemblies, this includes assurance that all of these configurations are subcritical with justified margins.

6.141. The reactor core (i.e. the fuel elements, reflectors, geometry of cooling channels, irradiation devices and structural parts) shall be designed to maintain the relevant parameters below predetermined limits in all operational states. Provisions shall be considered in the design for monitoring the physical conditions and integrity of the fuel. The design shall ensure that inadvertent movement of fuel elements or core components is not possible (e.g. by upward thrust due to flow).

6.142. The reactor core, including fuel elements, reactivity control mechanisms[29] and experimental devices shall be designed and constructed so that the maximum permissible design limits that are determined for all operational states are not exceeded. A suitable margin, including margins for uncertainties and engineering tolerances, shall be incorporated in setting these limits.

6.143. The reactor core shall be designed so that the reactor can be shut down, cooled[30] and maintained subcritical with an adequate margin for all operational states and accident conditions. The end state of the reactor core shall be assessed for selected design extension conditions.

6.144. Wherever possible, the design of the reactor core shall make use of inherent safety characteristics to minimize the consequences of accident conditions due to transients and instabilities.

6.145. The design and construction of the core of a subcritical assembly shall ensure that criticality cannot be reached for any core configuration (fuel, reflector and neutron source, if any), temperatures, moderation and reflection circumstances.

**Requirement 45: Provision of reactivity control**

**The design of a research reactor shall provide adequate means to control the reactivity.**

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[29] 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 or the reflection.

[30] The cooling requirement might not apply to some types of critical assembly and subcritical assembly.

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

6.147. Sufficient negative reactivity shall be available in the reactivity control devices(s) so that the reactor can be brought to a subcritical condition and maintained subcritical in all operational states and in accident 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 and tear and the effects of irradiation, such as burnup, poison buildup and decay, changes in physical properties and the production of gas. This requirement might not apply to some subcritical assemblies; however, subcriticality shall be justified for any configuration (see para. 6.145).

6.148. The maximum rate of addition of positive reactivity allowed by the reactivity control system or an experiment shall be specified and shall be limited to values justified in the safety analysis report and documented in the operational limits and conditions.

6.149. If a subcritical assembly will remain subcritical in any condition (even in the most reactive case), reactivity control devices might not be required.

**Requirement 46: Reactor shutdown systems**

**Means shall be provided for a research reactor to ensure that there is a capability to shut down the reactor in operational states and in accident conditions, and that the shutdown condition can be maintained for a long period of time, with margins, even for the most reactive conditions of the reactor core.**

6.150. At least one automatic shutdown system shall be incorporated into the design.[31] 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 in the design.

[31] A subcritical assembly can be 'shut down' by the withdrawal of the neutron source.

6.151. The effectiveness, speed of action and shutdown margin[32] of the reactor shutdown system shall be such that the conditions and the design limits for the fuel specified in the safety analysis report are met.

6.152. No single failure in the shutdown system shall be capable of preventing the system from fulfilling its safety function when required.

6.153. A capability to initiate manual reactor emergency shutdown shall be provided. This manual reactor trip signal shall also be provided as an input to the reactor protection system. The manual reactor trip shall be able to shut down the reactor directly. Consideration shall be given to the provision of the capability to initiate manual emergency shutdown of the reactor from locations other than the main control room (e.g. from the reactor operational area(s) or from the supplementary control room).

6.154. Instrumentation shall be provided and tests shall be performed to ensure that the means of shutdown are in the state stipulated for the given condition of the reactor.

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

**Requirement 47: Design of reactor coolant systems and related systems**

**The coolant systems for a research reactor shall be designed and constructed to provide adequate cooling to the reactor core.**

6.156. Systems containing reactor coolant shall be designed to allow pre-service and in-service tests and inspections to detect the possible occurrence of leaks, cracks and brittle fractures.[33] Consideration shall be given in the design to ensuring material characteristics that ensure the slow propagation of failures.

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[32] 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.

[33] Some subcritical assemblies and critical assemblies do not require cooling systems.

6.157. In the design of water cooled reactors, particular attention shall be paid to preventing the uncovering of the core.

6.158. Where the primary cooling system is not designed for cooling the core after shutdown, a reliable separate system shall be provided for the removal of residual heat.

6.159. For reactor systems that use flappers[34] or equivalent systems for the transition from forced to natural circulation cooling, or for operation with natural circulation cooling, and for which this mode is part of the safety system (or is considered an engineered safety feature), the single failure criterion shall be applied. Instrumentation to verify their functioning and to provide signals to the reactor protection system shall be provided.

6.160. If two coolant systems that are operating at different pressures are interconnected, the requirement of para. 6.118 applies.

6.161. Provision shall be made in the design for controlling the volume, temperature and pressure of the reactor coolant in any operational state of the facility, with due account taken of volumetric changes and leakage.

6.162. Provisions shall be made in the design to monitor and control the properties of the reactor coolant (e.g. the pH and conductivity of the water) and/or the moderator, and to remove radioactive substances, including activated corrosion products and fission products, from the coolant. Despite the fact that subcritical assemblies might not require cooling systems for heat removal, such provisions shall be applied to the fluids contained within such assemblies, to preserve fuel elements and structures, systems and components and to avoid radioactive releases.

6.163. Design features (such as leak detection systems, appropriate interconnections and capabilities for isolation) and suitable redundancy and diversity shall be provided to fulfil the requirements of paras 6.73–6.81 with adequate reliability for each postulated initiating event. Such measures also apply to subcritical assemblies.

[34] A flapper is a passive valve that opens when the flow (pressure) is below a set value to allow for the creation of natural circulation in the event of a loss of forced flow.

**Requirement 48: Emergency cooling of the reactor core**

**An emergency core cooling system shall be provided for a research reactor, as required, to prevent damage to the fuel in the event of a loss of coolant accident.**

6.164. The emergency core cooling system shall be capable of preventing significant failure of fuel for the range of accidents specified in the design basis (i.e. under design basis accidents, damage to the fuel and releases of radioactive material shall be kept within authorized limits).[35] Special procedures for cooling the core shall be considered in the case of selected design extension conditions.

6.165. For design basis accidents, the emergency core cooling shall be designed to perform its intended function in the event of any single failure in the system.

6.166. 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.

**Instrumentation and control systems**

**Requirement 49: Provision of instrumentation and control systems**

**Instrumentation shall be provided for a research reactor facility for monitoring the values of all the main variables that can affect the performance of the main safety functions and the main process variables that are necessary for its safe and reliable operation, for determining the status of the facility under accident conditions and for making decisions for accident management. Appropriate and reliable control systems shall be provided at the facility to maintain and limit the relevant process variables within the specified operating ranges.**

6.167. The reactor shall be provided with sufficient instrumentation and recording means to monitor important reactor parameters and the status of essential equipment of the reactor (including the position of neutron source) and associated experimental devices in all facility states. The expected response of

[35] Critical assemblies and subcritical assemblies might not require emergency core cooling systems.

such instrumentation and control systems in an emergency shall be assessed and taken into account in the emergency arrangements (see GSR Part 7 [6]).

6.168. The reactor shall be provided with appropriate controls, both manual and automatic as appropriate, to maintain parameters within specified operating ranges.

6.169. In the design of the instrumentation and control systems, provision shall be made as appropriate for startup neutron sources and dedicated startup instrumentation for conditions in which they are needed. This requirement shall be fulfilled for commissioning and startup after long shutdown periods.

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

6.171. Interconnections between reactor instrumentation and control systems and systems for controlling experimental devices shall in general be prohibited. Exceptions shall be permitted only if interconnections for controlling specific parameters of experimental devices are mandatory for the safe operation of the reactor.

**Requirement 50: Reactor protection system**

**A protection system shall be provided for a research reactor to initiate automatic actions to actuate the safety systems necessary for achieving and maintaining a safe state.**

6.172. The reactor protection system shall be independent of other systems and shall be capable of overriding unsafe actions of the control system.

6.173. The reactor protection system shall be capable of automatically initiating the required safety actions, for the full range of postulated initiating events, to actuate the safety systems necessary for achieving a safe state.

6.174. The reactor protection system shall be designed in such a way that, once the sequence of protective actions has been initiated automatically by the reactor protection system, it will proceed to completion and no manual actions will be necessary within a short period of time following activation of the reactor protection system. Such automatic actions by the reactor protection system shall not be self-resetting, and deliberate operator action shall be required to return to normal operation.

6.175. The possibility of bypassing interlocks and trips of the reactor protection system that might result in the bypassing of a safety function shall be carefully evaluated and justified. Appropriate means of preventing interlocks and trips that are important to safety from being inadvertently bypassed shall be incorporated into the reactor protection system.

6.176. The design of the reactor protection system shall be such that no single failure could result in the loss of automatic protective actions.

6.177. The reactor protection function shall be designed to bring the reactor to a safe condition and to maintain it in a safe condition even if the reactor protection systems are subjected to a credible common cause failure.

6.178. The reactor protection system shall be designed to permit periodic testing of its functionality.

6.179. 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 to be considered in establishing this margin are:

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

6.180. Where a computer based system is intended to be used in a reactor protection system, the following requirements apply in addition to that of para. 6.176:

(a) Hardware and software of high quality and proven design shall be used.
(b) The whole development process, including control, testing and commissioning of the design, shall be systematically documented and reviewable.
(c) In order to confirm the reliability of the computer based systems, a systematic, fully documented and reviewed assessment shall be undertaken by expert personnel who are independent of the designers and the suppliers.

(d) Protection shall be provided against accidental disruption of, or deliberate interference with, system operations.

6.181. Where the necessary high reliability 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 shall be provided.

**Requirement 51: Reliability and testability of instrumentation and control systems**

**Instrumentation and control systems for items important to safety at a research reactor shall be designed for high functional reliability and periodic testability commensurate with the safety function(s) to be performed.**

6.182. 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.

6.183. Design techniques such as testability, including a self-checking capability where necessary, fail-safe characteristics, functional diversity and diversity in component design and in concepts of operation shall be used to the extent practicable to prevent loss of a safety function.

**Requirement 52: Use of computer based equipment in systems important to safety**

**If a system important to safety at a research reactor is dependent upon computer based equipment, appropriate standards and practices for the development and testing of computer hardware and software shall be established and implemented throughout the lifetime of the system, and in particular throughout the software development cycle. The entire development shall be subject to an integrated management system.**

6.184. For computer based equipment in safety systems and systems important to safety:

(a) A high quality of, and best practices for, hardware and software shall be used, in accordance with the importance of the system to safety.
(b) The entire development process, including the control, testing and commissioning of design changes, shall take into account all phases of the life cycle of the computer based system, shall be systematically documented and shall be reviewable.
(c) An assessment of the equipment shall be undertaken by experts who are independent of the design team and the supplier team to provide assurance of its high reliability.
(d) When the necessary high reliability of the equipment cannot be demonstrated with a high level of confidence, diverse means of ensuring fulfilment of the safety functions shall be provided (see also para. 6.181).
(e) Common cause failures deriving from software shall be taken into consideration.
(f) Protection shall be provided against accidental disruption of, or deliberate interference with, system operation (computer based systems and communication and network systems important to safety, including the reactor protection system, are to be adequately protected against cyber-attacks, up to and including the design basis threat [13]).
(g) Appropriate verification and validation and testing of the software systems shall be performed.

**Requirement 53: Control room**

**A control room shall be provided at a research reactor facility from which the facility can be safely operated in all operational states, either automatically or manually, and from which measures can be taken to maintain the research reactor in a safe state or to bring it back into a safe state after anticipated operational occurrences and accident conditions.**

6.185. Appropriate measures shall be taken and adequate information shall be provided for the protection of occupants of the control room, for an extended period of time, against hazards such as high radiation levels resulting from accident conditions, releases of radioactive material, fire, or explosive or toxic gases. See also para. 6.91 for requirements on the means of communication between the control room and the supplementary control room and the emergency centre.

6.186. Special attention shall be paid to identifying those events, both internal and external to the control room, that could challenge its continued operation, and the design shall provide for practicable measures to minimize the consequences of such events. The design shall provide for escape routes for the occupants in case of events necessitating the evacuation of the control room.

6.187. The design of the control room shall provide an adequate margin against natural hazards more severe than those selected for the design basis.

**Requirement 54: Supplementary control room**

**Provision of a supplementary control room for a research reactor facility, separate and functionally independent from the main control room, shall be considered in the design.**

6.188. The means provided in the supplementary control room (sometimes known as a remote shutdown panel) shall be sufficient for fulfilment of the main safety functions (shutdown, cooling, confinement and monitoring of the facility status) 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 items important to safety. A supplementary control room might not be necessary for critical assemblies and subcritical assemblies. In this case, the decision shall be justified on the basis of a comprehensive analysis.

**Requirement 55: Emergency response facilities on the site**

**A research reactor facility shall include the necessary emergency response facilities on the site. Their design shall be such that personnel will be able to perform expected tasks for managing an emergency under conditions generated by accidents as well as initiating events.**

6.189. Information about important reactor parameters and radiological conditions at the reactor facility and the site, and information from monitoring systems and laboratory facilities that is to be used to determine the need to initiate emergency measures, as well as information to be used for continuing assessment, shall be provided to the relevant emergency response facilities[36]. Each emergency response facility shall be provided with means of communication with the control room, the supplementary control room and other important locations at the facility, and with on-site and off-site emergency response organizations.

## POWER SUPPLIES

**Requirement 56: Electrical power supply systems**

**The design for a research reactor facility shall include reliable normal electrical power supply systems and shall consider reliable emergency electrical power supply systems.**

6.190. Reliable electrical power supplies for essential safety functions shall be available in normal operational states and in accident conditions.

6.191. The design shall consider the provision of uninterruptible power supplies for those safety systems that require a continuous energy supply, such as the reactor protection system and the radiation monitoring system.

6.192. In the design basis for the emergency power supply, due account shall be taken of the postulated initiating events and the associated safety functions to be performed to determine the requirements for capability, availability, duration of the required power supply, capacity and continuity.

**Requirement 57: Radiation protection systems**

**Equipment shall be provided at a research reactor facility to ensure that there is adequate radiation monitoring in operational states and accident conditions.**

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[36] Emergency response facilities and locations are addressed in GSR Part 7 [6]. For research reactors, emergency response facilities (which are separate from the control room and the supplementary control room) include the emergency centre, and the technical support centre and the operational support centre, as appropriate.

6.193. The design of radiation protection systems shall include:

(a) Stationary dose rate meters for monitoring the local radiation dose rate at places routinely accessible by operating personnel and at other places where the changes in radiation levels in operational states could be such that access is allowed only for certain specified periods of time (e.g. beam tube areas, and areas where neutron sources are located in the subcritical facilities).
(b) Stationary dose rate meters to indicate the general radiation levels at suitable locations of the facility in anticipated operation occurrences and accident conditions. The stationary dose rate meters shall provide sufficient information in the control room or in the appropriate control position that operating personnel can initiate protective actions and corrective actions if necessary.
(c) Monitors for measuring the activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel, including experimental areas, 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, in a timely manner, 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 and accident conditions.
(e) Stationary equipment for monitoring and controlling effluents prior to or during their discharge to the environment.
(f) Devices for measuring radioactive surface contamination.
(g) Installations and equipment for measuring doses to and contamination of personnel.
(h) Radiation monitoring at gates and other entrances of the facility to detect radioactive material being moved without permission or unintentional contamination.

6.194. In addition to monitoring within the facility, arrangements shall also be made to assess exposures and other radiological impacts in the vicinity of the facility, where necessary.

**Requirement 58: Handling and storage systems for fuel and core components**

**The design for a research reactor facility shall include provisions for the safe handling and storage of fresh and irradiated fuel and core components.**

6.195. The design shall include provisions for safely storing a sufficient number of spent fuel elements and irradiated core components. These provisions shall be in accordance with the programmes for core management and for removing or replacing fuel elements and core components.

6.196. The design shall include provisions to unload all fuel from the core safely at any time.

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

6.198. The handling and storage systems shall be designed:

(a) To prevent criticality by an adequate margin, 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 appropriate 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 prevent unacceptable levels of stress in the fuel elements;
(i) To identify and track individual fuel elements and assemblies.

6.199. Handling and storage systems for irradiated fuel shall be designed to permit adequate heat removal and shielding in operational states and accident conditions.

6.200. Critical assemblies and subcritical assemblies are unlikely to include spent fuel or significantly irradiated fuel and therefore the requirements relating to handling and storage of spent fuel or significantly irradiated fuel might not apply. The other requirements established in paras 6.195–6.198 apply.

**Requirement 59: Radioactive waste systems**

**The design of a research reactor facility and its associated experimental facilities shall include provisions to enhance safety in waste management and to minimize the generation of radioactive waste. Systems shall be provided for treating solid, liquid and gaseous radioactive waste to keep the amounts**

**and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges.**

6.201. 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 and provided as necessary.

6.202. Means shall be provided in the design for the handling, 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. Where gaseous radioactive material is to be handled, provision shall be made for the detection of leakage and the prevention and control of releases to below authorized limits for a radioactive release.

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

### Supporting systems and auxiliary systems

**Requirement 60: Performance of supporting systems and auxiliary systems**

**The design of supporting systems and auxiliary systems for a research reactor shall be such as to ensure that the performance of these systems is consistent with the safety significance of the system or component that they serve at the research reactor.**

6.204. 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.

**Requirement 61: Fire protection systems**

**Fire protection systems for a research reactor facility, including fire detection systems and fire extinguishing systems, fire containment barriers and smoke control systems, shall be provided throughout the research reactor facility, with due account taken of the results of the fire hazard analysis.**

6.205. The fire protection systems installed at the research reactor shall be capable of dealing safely with postulated fire events. The design of the fire protection system shall consider the potential for accidental criticality in a critical assembly or subcritical assembly. Fire hazards due to experiments shall be considered.

6.206. Fire extinguishing systems shall be capable of automatic actuation where appropriate. Fire extinguishing systems shall be designed and located to ensure that their rupture or spurious or inadvertent operation would not impair the capability of items important to safety.

6.207. Fire detection systems shall be designed to provide alarms and prompt information on the location and spread of fires that start in the reactor facility at any time.

6.208. Fire detection systems and fire extinguishing systems that are necessary to protect against a possible fire following a postulated initiating event shall be appropriately qualified to resist the effects of the postulated initiating event.

6.209. Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the facility, in particular in locations such as the means of confinement and the control rooms.

**Requirement 62: Lighting systems**

**Adequate lighting shall be provided in all operational areas of a research reactor facility for operational states and in accident conditions.**

**Requirement 63: Lifting equipment**

**Equipment shall be provided for lifting and lowering items important to safety at a research reactor facility, and for lifting and lowering other items in the proximity of items important to safety.**

6.210. The lifting equipment shall be designed so that:

(a) Measures are taken to prevent the lifting of excessive loads, including those for experimental programmes;
(b) Conservative design measures are applied to prevent any unintentional dropping of loads that could affect items important to safety or could cause a radiological hazard (e.g. a spent fuel cask);

(c) The facility layout permits safe movement of the lifting equipment and of items being transported, in accordance with analysed safe load pathways;
(d) Such equipment for use in areas where items important to safety are located is seismically qualified;
(e) Such equipment can be inspected on a periodic basis.

**Requirement 64: Air conditioning systems and ventilation systems**

**Systems for air conditioning, air heating, air cooling and ventilation for a research reactor facility shall be provided as appropriate in areas at the facility to maintain the required environmental conditions.**

6.211. Systems shall be provided for the ventilation of buildings at the reactor facility with appropriate capability for the conditioning and cleaning of air:

(a) To prevent unacceptable dispersion of airborne radioactive substances within the facility;
(b) To reduce the concentration of airborne radioactive substances to levels compatible with the need for access by personnel to the area;
(c) To keep the levels of airborne radioactive substances in the reactor facility below authorized limits and as low as reasonably achievable;
(d) To ventilate rooms containing inert gases or noxious gases without impairing the capability to control radioactive effluents;
(e) To maintain the required efficiency of the filtration system and to control releases of gaseous radioactive material to the environment and maintain them below the authorized limits on discharges and to keep them as low as reasonably achievable.

**Requirement 65: Compressed air systems**

**The design basis for any compressed air system that serves an item important to safety at a research reactor facility shall specify the quality, flow rate and cleanness of the air to be provided.**

**Requirement 66: Experimental devices**

**Experimental devices for a research reactor shall be designed so that they will not adversely affect the safety of the reactor in any operational states or accident conditions. In particular, experimental devices shall be designed so that neither the operation nor the failure of an experimental device will result in an unacceptable change in reactivity for the reactor, affect operation**

**of the reactor protection system, reduce the cooling capacity, compromise confinement or lead to unacceptable radiological consequences.**

6.212. A design basis shall be established for each experimental device associated directly or indirectly with the reactor. Experimental devices shall be classified on the basis of their importance to safety. 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. The safety analysis shall also cover the interaction between the experimental devices and the reactor (see also para. 6.124).

6.213. 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.

6.214. The design of experiments and experimental devices shall facilitate their dismantling operations, interim storage and final disposition.

# 7. OPERATION OF RESEARCH REACTOR FACILITIES

## ORGANIZATIONAL PROVISIONS

**Requirement 67: Responsibilities of the operating organization**

**The operating organization for a research reactor facility shall have the prime responsibility for safety in the operation of the facility.**

7.1. The prime responsibility for safety shall be assigned to the operating organization for the research reactor facility. This prime responsibility shall cover all the activities relating to the operation directly and indirectly, including activities for experiments.[37] It includes the responsibility for supervising the activities of all other related groups, such as designers, suppliers, manufacturers and constructors, employers, contractors and experimenters, as well as the responsibility for operation of the reactor facility by the operating organization itself. The operating organization shall discharge this responsibility in accordance with its management system [4].

7.2. The operating organization shall establish an appropriate management structure for the research reactor and shall provide for all necessary infrastructure for the conduct of reactor operations. The organization for reactor operation (the reactor management, see footnote 21) shall include the reactor manager[38] 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 maintenance, periodic testing and inspection, radiation protection, quality assurance and relevant support services.

7.3. The responsibility of the operating organization for the safety of the research reactor shall not be delegated. The reactor manager shall have the direct responsibility and the necessary authority for the safe operation of the research reactor.

7.4. The operating organization shall establish, in accordance with the management system, the functions and responsibilities for the key positions in the organization for reactor operation. In particular, the operating organization shall establish clear lines of authority and communication with the reactor manager, the safety committee(s), the radiation protection group, maintenance groups, the management system personnel and the experimenters.

7.5. The staff positions that require a licence or certificate shall be determined in accordance with the legal framework of the State. These positions shall receive

[37] Operation includes all activities performed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified. Besides operating the reactor, this includes: 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; the use of the research reactor systems for the purposes of research and development and education and training; and other associated activities.

[38] 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 fulfilment of this responsibility.

adequate training as required by the regulatory body (see also paras 7.13–7.22). In particular, in accordance with regulatory requirements, the reactor manager[39], the shift supervisors and the reactor operators shall hold an authorization (a licence or certificate) issued by the regulatory body, operating organization or other competent authority.

7.6. In collaboration with the supplier or the designer, the operating organization shall have overall responsibility for the preparation and satisfactory completion of the commissioning programme (see para. 7.51).

7.7. The operating organization shall prepare and issue specifications and procedures in accordance with the classification of structures, systems and components and the management system, in particular for the procurement, manufacturing, loading, utilization, unloading, storage, movement and testing of items important to safety, including fuel and core components and other fresh or irradiated fissile material.

7.8. 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.

7.9. It shall be the responsibility of the operating organization to ensure the following:

(a) Safety policies are issued and clearly understood by everyone.
(b) The establishment of its advisory safety committee.
(c) The design enables the reactor to be operated safely, and the reactor is constructed in accordance with the approved design.
(d) An adequate safety analysis report is prepared and kept up to date, in accordance with Requirement 1.
(e) The commissioning process demonstrates that the design requirements have been met and that the reactor can be operated in accordance with the design assumptions.
(f) A system for reporting and reviewing abnormal events is established and operated.

[39] The reactor manager does not necessarily need to hold a licence to operate the reactor, but needs to have completed a training programme (see para. 7.30).

(g) On-site emergency arrangements, including the emergency plan and procedures, are established and maintained in accordance with GSR Part 7 [6].
(h) The research reactor is operated and maintained in accordance with the safety requirements by suitably qualified and experienced personnel certified by the relevant authorities.
(i) 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 7.28–7.31).
(j) Adequate resources, facilities and services are made available during operation.
(k) Information on events with safety significance that are required to be reported to the regulatory body, including any assessments of such events and the corrective actions intended, is submitted to the regulatory body.
(l) 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 conduct of activities during operation of the facility (see paras 4.1 and 4.4).
(m) An integrated management system (see footnote 15) is established and implemented, in accordance with a graded approach (see paras 4.7–4.13).
(n) The reactor management is provided with sufficient authority and resources to enable it to fulfil its duties effectively.
(o) The research reactor is operated and maintained in accordance with the operational limits and conditions and operating procedures (see paras 7.32–7.34 and 7.57–7.62).
(p) The fissile material and radioactive material that are utilized or generated are controlled.
(q) Operating 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.
(r) An exclusion programme for foreign objects is implemented and monitored, in accordance with regulatory requirements.

**Requirement 68: Structure and functions of the operating organization**

**The structure of the operating organization for a research reactor facility and the functions, roles and responsibilities of its personnel shall be established and documented.**

7.10. Functional responsibilities, lines of authority, and lines of internal and external communication for the safe operation of the research reactor in all operational states and in accident conditions shall be clearly specified in writing.

7.11. The organizational structure and the arrangements for discharging responsibilities shall be documented in the safety analysis report and made available to the staff and, if required, to the regulatory body. The structure of the operating organization shall be specified so that all roles that are critical for safe operation are specified and described. Proposed organizational changes to the structure and associated arrangements, which might be of importance to safety, shall be analysed in advance by the operating organization and submitted to the regulatory body for approval.

7.12. The operating organization shall be responsible for ensuring that the necessary knowledge, skills, attitudes and safety expertise are sustained at the research reactor, and that long term objectives for human resources are met and knowledge preservation policies are developed.

**Requirement 69: Operating personnel**

**The operating organization for a research reactor facility shall assign direct responsibility and authority for the safe operation of the reactor to the reactor manager. The reactor manager shall have overall responsibility for all aspects of operation, training, maintenance, periodic testing, inspection, utilization and modification of the reactor. Discharge of this responsibility shall be the primary duty of the reactor manager.**

**Reactor manager**

7.13. 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. The duties, responsibilities and lines of communication of other personnel involved in the operation or use of the reactor (e.g. technical support personnel and experimenters) shall also be clearly documented.

7.14. 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 in accordance with the operational limits and conditions. These requirements include both the number of personnel and the duties for which they are required to be authorized. The person with qualification and responsibility for the direct supervision of the operation of the reactor shall be clearly identified at all times. The availability of the staff that would be required to deal with accident conditions shall also be specified (see also Requirement 21 of GSR Part 7 [6]).

7.15. 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. Adequate training in the procedures to be followed in both operational states and accident conditions shall be conducted (see paras 7.57–7.62 of this publication and Requirement 25 of GSR Part 7 [6]).

7.16. Notwithstanding the presence of independent radiation protection personnel (see para. 7.23), the operating personnel, including technical support personnel and experimenters, shall be given suitable training in radiation protection before the start of their duties. Periodic refresher training in operational radiation protection shall be carried out.

7.17. 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.

7.18. 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.

7.19. The reactor manager shall periodically review the operation of the research reactor, including experiments, and shall take appropriate corrective actions in respect of any problems identified. The reactor manager shall seek the advice of the safety committee(s) or shall call upon advisors to review important safety issues arising in the commissioning, operation, maintenance, periodic testing and inspection, and modification of the reactor and experiments (see para. 7.26).

**Operating personnel**

7.20. The operating personnel shall operate the facility in accordance with the approved operational limits and conditions and operating procedures (see paras 7.32–7.34 and 7.57–7.62). 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.

7.21. Every licensed or authorized member of the operating personnel shall have the authority to shut down the reactor in the interest of safety.

7.22. A maintenance group shall be established by the operating organization to implement the programmes for maintenance, periodic testing and inspection, as set out in paras 7.38–7.39.

**Radiation protection personnel**

7.23. 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 described in Requirement 84, paras 7.107–7.114.

**Additional support personnel**

7.24. The operating organization shall make provision as needed for additional technical personnel[40] such as training officers, safety officers and reactor chemists.

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

**Reactor safety committee**

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

7.27. Notwithstanding any advice of the operating organization's safety committee (see Requirement 6), the reactor manager (see para. 7.3) 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 have the authority to refer such a proposal to higher authority for additional review.

[40] Facilities of low potential hazard might not need to have these positions. However, the functions need to be covered within such facilities.

**Requirement 70: Training, retraining and qualification of personnel**

**The operating organization for a research reactor facility shall ensure that safety related functions are performed by suitably qualified, competent and fit-for-duty personnel.**

7.28. The operating organization shall clearly define the requirements for qualification and competence to ensure that personnel performing safety related functions are capable of safely performing their duties. Certain operating positions may require formal authorization or a licence.

7.29. Suitably qualified personnel shall be selected and shall be given the necessary training and instruction to enable them to perform their duties correctly for different operational states and in accident conditions, in accordance with the appropriate procedures. Safety related functions shall be performed by suitably qualified, competent and fit-for-duty personnel.

7.30. Suitable training and retraining programmes shall be established and maintained for the operating personnel, including the reactor manager, the shift supervisors, the reactor operators, the radiation protection staff, the maintenance personnel and others working at the research reactor facility. The training programme shall include provision for periodic confirmation of the competence of personnel, which shall be documented, and for refresher training on a regular basis. The refresher training shall also include retraining provision for personnel who have had extended absences from their authorized duties. The training shall emphasize the importance of safety in all aspects of reactor operation and shall promote safety culture.

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

**Requirement 71: Operational limits and conditions**

**The operating organization for a research reactor facility shall ensure that the research reactor is operated in accordance with the operational limits and conditions.**

7.32. The operational limits and conditions shall form an important part of the basis for the authorization of the operating organization to operate the research reactor facility. The facility shall be operated within the operational limits and conditions to prevent situations arising that could lead to anticipated operational occurrences or accident conditions, and to mitigate the consequences of such events if they do occur. The operational limits and conditions shall be developed to ensure that the reactor is being operated in accordance with the design assumptions and intent, as well as in accordance with its licence conditions.

7.33. The operational limits and conditions shall reflect the provisions made in the final design as described in the safety analysis report. The set of operational limits and conditions important to reactor safety, including safety limits, safety system settings, limiting conditions for safe operation, requirements for surveillance, testing and maintenance, and administrative requirements, shall be established and submitted to the regulatory body for review and assessment and approval before the commencement of operation. All operational limits and conditions shall be substantiated by a written statement or by analysis of the reason for their adoption.

7.34. The operational limits and conditions shall be adequately defined, clearly established and appropriately substantiated (e.g. by clearly stating for each operational limit or condition its objective, its applicability and its specification; i.e. its specified limit and its basis). The selection of, and the values for, the operational limits and conditions shall be based on the safety analysis, on the reactor design or on aspects relating to the conduct of operations, and shall be demonstrably consistent with the updated safety analysis report, shall reflect the present status of the reactor and shall correspond to the licence conditions imposed by the regulatory body.

**Safety limits**

7.35. Safety limits shall be set to protect the integrity of the physical barriers that protect against the uncontrolled release of radioactive material or exposure over regulatory limits.

**Safety system settings**

7.36. Safety system settings shall be defined so that the safety limits are not exceeded.

**Limiting conditions for safe operation**

7.37. Limiting conditions for safe operation shall be established to ensure that there are acceptable margins between normal operating values and the safety system settings. Limiting conditions for safe operation shall include limits on operating parameters, requirements relating to the minimum availability of operable equipment and minimum staffing levels, and prescribed actions to be taken by operating personnel to preserve the settings of the safety system.

**Requirements for maintenance, periodic testing and inspection**

7.38. Requirements shall be established for the frequency and scope of inspections, periodic testing and maintenance, operability checks and calibrations of all items important to safety to ensure compliance with the safety analysis report.

7.39. The requirements for maintenance, surveillance, periodic testing and inspection shall include a specification that clearly defines the objectives and the applicability, prescribes the frequency for the performance of activities and establishes criteria for acceptable deviations. In order to provide operational flexibility, the specification shall prescribe the frequency of activities in terms of average intervals with a maximum interval that is not to be exceeded. Deferrals that exceed the maximum interval shall be justified and made subject to approval, and safety measures shall be put in place where necessary.

**Administrative requirements**

7.40. The operational limits and conditions shall include administrative requirements or controls concerning organizational structure and the responsibilities for key positions for 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 the operational limits and conditions.

**Violations of operational limits and conditions**

7.41. In the event that the operation of the reactor deviates from one or more operational limits and conditions, corrective actions shall be taken.

7.42. 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.

7.43. If a safety limit is exceeded, the reactor shall be shut down and maintained in a safe state and inspections on challenged items important to safety shall be performed. 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.

**Requirement 72: Performance of safety related activities**

**The operating organization for a research reactor facility shall ensure that safety related activities are adequately analysed and controlled to ensure that the risks associated with harmful effects of ionizing radiation are kept as low as reasonably achievable.**

7.44. All routine and non-routine operational activities shall be assessed for potential risks associated with harmful effects of ionizing radiation. The level of assessment and control shall depend on the safety significance of the task.

7.45. All activities important to safety shall be carried out in accordance with approved written procedures to ensure that the research reactor is operated within the established operational limits and conditions. Acceptable margins shall be ensured between normal operating values and the established safety system settings to avoid undesirably frequent actuation of safety systems (see para. 7.37).

7.46. No experiments shall be conducted without adequate review and justification. If there is a need to conduct a non-routine operation or test that is not covered by existing operating procedures, a specific safety review shall be performed and a special procedure shall be developed and made subject to approval in accordance with national or other relevant regulations.

## COMMISSIONING

**Requirement 73: Commissioning programme**

**The operating organization for a research reactor facility shall ensure that a commissioning programme for the research reactor is established and implemented.**

7.47. 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 cover the full range of facility conditions required in the design. The commissioning programme shall establish the organization and responsibilities for commissioning, the commissioning stages, the suitable testing of structures, systems and components 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.

7.48. During construction and commissioning, a comparison shall be carried out between the as built reactor facility and its design parameters. A comprehensive process shall be established under the management system of the operating organization to address non-conformances in design, manufacturing, construction and operation. Resolutions to correct differences from the initial design and non-conformances shall be documented and reviewed before starting the commissioning.

7.49. The detailed 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.

7.50. Experimental devices and their potential impact on reactor operations shall be given adequate consideration during the commissioning of the reactor. Experimental devices shall be subject to an adequate commissioning programme prior to being placed in service.

**Organization and responsibilities for commissioning**

7.51. The operating organization, designers and manufacturers shall be involved in the preparation and execution of the commissioning programme. The commissioning process shall involve cooperation 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**

7.52. 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 in the following sequence:

(a) Stage A: Tests prior to fuel loading;
(b) Stage B: Fuel loading tests, initial criticality tests and low power tests[41];
(c) Stage C: Power ascension tests and power tests.

**Commissioning procedures and reports**

7.53. Procedures shall be prepared, reviewed and made subject to approval for each commissioning test prior to the commencement of the tests. 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, external agencies, manufacturers and the regulatory body.

7.54. The commissioning programme shall include provisions and procedures for audits, reviews and verifications intended to ensure that the programme has 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.

7.55. Commissioning procedures 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 commissioning reports shall be kept for the entire lifetime of the facility including the decommissioning stage. The reports shall cover the following:

(a) The purpose of the tests and the expected results;
(b) The safety provisions required to be in force during the tests;

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[41] Initial criticality tests and low power tests and Stage C of the commissioning programme might not apply to subcritical assemblies, providing adequate subcriticality has been verified (e.g. through $1/M$ calculations, where $M$ is the subcritical neutron multiplication factor).

(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.

7.56. 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.

**Requirement 74: Operating procedures**

**Operating procedures for the research reactor shall be developed that apply comprehensively (for the reactor and its associated facilities) for normal operation, anticipated operational occurrences and accident conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body.**

7.57. Procedures shall be developed for normal operation to ensure that the reactor is operated within the operational limits and conditions.

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

(a) Commissioning;
(b) Operation in normal operational states[42];
(c) The maintenance of major components or systems that could affect reactor safety;
(d) Periodic inspections, calibrations and tests of structures, systems and components 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 irradiation and experiments that could affect reactor safety or the reactivity of the core;
(g) The reactor operator's response to anticipated operational occurrences and design basis accidents, and, to the extent feasible, to design extension conditions;

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[42] Normal operation is operation within specified operational limits and conditions. For a research reactor, this includes startup, low and nominal power operation, shutting down, shutdown, maintenance, testing and refuelling.

(h) Emergencies[43];
(i) Handling of radioactive waste and monitoring and control of radioactive releases;
(j) Utilization;
(k) Modifications;
(l) The management system.

7.59. Operating procedures shall be developed by the reactor operating personnel, in cooperation 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 contribute to the observance of the operational limits and conditions.

7.60. The operating procedures shall be reviewed and updated periodically on the basis of lessons learned from operating experience, or in accordance with predetermined internal procedures. They shall be made available as relevant for the particular mode of operation of the reactor.

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

7.62. 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 activity is started. Additional training of relevant staff in these procedures shall be provided.

**Requirement 75: Main control room, supplementary control room and control equipment**

**The operating organization for a research reactor facility shall ensure that the operation control rooms and control equipment are maintained in a suitable condition.**

7.63. The habitability and good condition of control rooms shall be maintained. Where the design of the research reactor foresees additional or local control rooms that are dedicated to the control of experiments that could affect the

[43] Emergency procedures are developed as an element of separate emergency arrangements (see paras 7.89–7.93) and in accordance with GSR Part 7 [6].

reactor conditions, clear communication lines shall be developed for ensuring an adequate transfer of information to the operators in the main control room.

7.64. The supplementary control room or a shutdown panel and all other safety related local control rooms or operational panels outside the control room shall be kept operable and free from obstructions, as well as from non-essential material that would prevent their operation. The operating organization shall periodically confirm that the supplementary control room or shutdown panel and all other safety related operational panels are in the proper state of operational readiness, including proper documentation, communications and alarm systems as well as sufficient power supply.

7.65. A hierarchy of precedence shall be established between the supplementary and the main control rooms to prevent conflicting inputs (e.g. by interlocks) being given from different control rooms or panels.

**Requirement 76: Material conditions and housekeeping**

**The operating organization for a research reactor facility shall develop and implement programmes to maintain a high standard of material conditions, housekeeping and cleanliness in all working areas.**

7.66. Administrative controls shall be established to ensure that operational premises and equipment are maintained, well lit and accessible, and that temporary storage is controlled and limited. Equipment that is degraded (e.g. owing to leaks, corrosion, loose parts or damaged thermal insulation) shall be identified, reported and corrected in a timely manner.

7.67. The operating organization shall be responsible for ensuring that the identification and labelling of safety equipment and safety related equipment, rooms, piping and instruments are accurate, legible and well maintained, and that they do not introduce any degradation.

**Requirement 77: Maintenance, periodic testing and inspection**

**The operating organization for a research reactor facility shall ensure that effective programmes for maintenance, periodic testing and inspection are established and implemented.**

7.68. Maintenance (both preventive maintenance and corrective maintenance), periodic testing and inspection shall be conducted to ensure that structures, systems and components are able to function in accordance with the design intent, in compliance with the operational limits and conditions.

7.69. The maintenance, periodic testing and inspection programmes shall be reviewed at regular intervals to incorporate lessons learned from experience. All maintenance, periodic testing and inspection 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. In accordance with the requirements of the management system, a system of work permits shall be used for maintenance, periodic testing and inspection, including appropriate procedures and checklists 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.

7.70. Non-routine inspections or corrective maintenance of systems or items important to safety shall be performed 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.

7.71. 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 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 operational limits and conditions.

7.72. The frequency of maintenance, periodic testing and inspection of individual structures, systems and components shall be adjusted on the basis of experience and shall be such as to ensure adequate reliability, in accordance with the requirements established in paras 6.73–6.75.

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

7.74. Maintenance shall not be performed in such a way as to result, either deliberately or unintentionally, in changes to the design of the system under maintenance. If a maintenance activity requires a design change, procedures for the implementation of a modification shall be followed.

7.75. Properly qualified personnel, who shall verify that the activities have been accomplished as specified in the appropriate procedure and shall verify compliance with the operational limits and conditions, shall assess the results of maintenance, periodic testing and inspection.

7.76. The safety committee and the regulatory body shall be informed of any non-conformance that is significant to safety. An assessment shall be made of the impact of the non-conformance on the maintenance programme.

**Requirement 78: Core management and fuel handling**

**Core management and fuel handling procedures for a research reactor facility shall be established to ensure compliance with operational limits and conditions and consistency with the utilization programme.**

7.77. Core management and fuel handling comprise the movement, storage, transfer, packaging and transport of fresh and irradiated fuel and other core components. Applicable safety requirements shall be documented in the operational limits and conditions and the relevant procedures shall be applied.

7.78. Core components and fuel loaded into the core shall comply with the quality requirements established in the management system.

7.79. To ensure safe operational cores, in addition to the demonstration of conformance with the safety analysis report and operational limits and conditions, the operating organization:

(a) Shall determine, using validated methods and codes, the locations for fuel and reflectors, the appropriate positions of experimental devices and moderators in the core and the effectiveness of the safety devices (such as neutron absorbing rods, valves for dumping the moderator and burnable poisons), as well as the relevant thermohydraulic and neutronic parameters.
(b) Shall analyse the possible interactions (both chemical and physical) between core components and with experimental devices.

(c) Shall keep and update information on the parameters for the fuel and core configurations. This includes maintaining up to date data at all times in support of accounting for and control of the nuclear material inventory in the facility.
(d) Shall load the fuel in accordance with the procedures for fuel handling and core management.
(e) Shall utilize (burnup) 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 operational limits and conditions for the reactor, and by detecting, identifying and unloading failed fuel.
(f) Shall unload the irradiated fuel when appropriate, and as applicable[44], in accordance with the burnup values prescribed in the operational limits and conditions.

7.80. 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 of any core component or material proposed for irradiation;
(b) The conduct of investigations into the causes of fuel failures and experiment failures and means of avoiding such failures;
(c) The assessment of the effects of irradiation on core components and core support structure materials.

7.81. Procedures shall be prepared for the handling of fuel assemblies and core components to ensure their quality and safety and to avoid damage or degradation. In addition, operational limits and conditions shall be established and procedures shall be prepared for dealing with failures of fuel elements, control rods, reflectors or moderators, experimental devices or any other core components so as to minimize the amounts of radioactive material released.

[44] Low power research reactors and subcritical assemblies usually have a lifetime core, which could be specified in the operational limits and conditions in terms of factors other than burnup (e.g. completion of the experimental programme). Nevertheless, the value of the maximum burnup is one of the parameters that is considered in the determination of the core lifetime.

7.82. The integrity of the reactor core and the fuel shall be continuously monitored by a system for the detection of failures of the cladding integrity (e.g. by monitoring fission product activity in the coolant). Failed fuel shall be stored in a manner that prevents the release of radioactive material while still maintaining the requisite degree of residual heat removal and shielding and subcriticality conditions.

7.83. The packaging and transport of fresh and irradiated fuel assemblies shall be carried out in accordance with national and international requirements and, as appropriate, in accordance with IAEA Safety Standards Series No. SSR-6, Regulations for the Safe Transport of Radioactive Material (2012 Edition) [14].

7.84. A comprehensive records system shall be maintained in compliance with the management system to cover core management and the handling and storage of fuel, and core components.

**Requirement 79: Fire safety**

**The operating organization for a research reactor facility shall make arrangements for ensuring fire safety.**

7.85. The arrangements for ensuring fire safety made by the operating organization shall cover the following: adequate management for fire safety; preventing fires from starting; detecting and extinguishing quickly any fires that do start; preventing the spread of those fires that have not been extinguished; and providing protection from fire for structures, systems and components that are necessary to shut down the reactor safely. Such arrangements shall include, but are not limited to:

(a) The application of the principle of defence in depth;
(b) The control of combustible materials and ignition sources;
(c) The maintenance, testing and inspection of fire protection measures;
(d) The establishment of a manual firefighting capability at the reactor facility;
(e) The assignment of responsibilities and training and exercising of personnel;
(f) The assessment of the impact of modifications on fire safety measures.

7.86. In the arrangements for firefighting, special attention shall be given to cases for which there is a risk of release of radioactive material in a fire. Appropriate measures shall be established for the radiation protection of firefighting personnel and the management of releases of radioactive material to the environment.

7.87. A comprehensive fire hazard analysis shall be developed for the research reactor and associated facilities and shall be periodically reviewed and, if necessary, updated.

**Requirement 80: Non-radiation-related safety**

**The operating organization for a research reactor facility shall establish and implement a programme to ensure that safety related risks associated with non-radiation-related hazards to personnel involved in activities at the reactor facility are kept as low as reasonably achievable.**

7.88. The non-radiation-related safety[45] programme shall include arrangements for the planning, implementation, monitoring and review of the relevant preventive and protective measures, and it shall be integrated with the nuclear and radiation safety programme. All personnel, suppliers, contractors and visitors shall be appropriately trained to provide them with the necessary knowledge and awareness of the non-radiation-related safety programme and its interface with the nuclear and radiation safety programme, and shall comply with its safety rules and practices. The operating organization shall provide support, guidance and assistance for the personnel in the area of non-radiation-related hazards.

**Requirement 81: Emergency preparedness**

**The operating organization for a research reactor facility shall prepare emergency arrangements for preparedness for, and response to, a nuclear or radiological emergency.**

7.89. The emergency arrangements shall be commensurate with the hazards assessed and the potential consequences of an emergency should it occur. Emergency arrangements shall cover the capability of maintaining protection and safety in the event of an emergency; mitigating the consequences of accidents if they do occur; protection of site personnel and the public; protection of the environment; and communicating with the public in a timely manner. Emergency arrangements shall include arrangements for the prompt declaration and notification of an emergency; timely initiation of coordinated and preplanned response; assessment of the progress of the emergency, its consequences and any actions that need to be taken on the site; and the necessary provision of

[45] 'Non-radiation-related safety' concerns hazards other than radiation related hazards; this is sometimes referred to as industrial safety or conventional safety.

information to the off-site authorities. Appropriate emergency arrangements shall be established from the time that nuclear fuel is first brought to the site, and all emergency arrangements shall be completed before the commencement of fuel loading.

7.90. The operating organization shall develop emergency arrangements that include emergency plans and procedures for on-site preparedness and response to an emergency in relation to the research reactor under its responsibility and shall demonstrate to, and provide, the regulatory body with an assurance that the emergency arrangements provide for an effective response on the site. The on-site emergency arrangements shall be coordinated with those of off-site response organizations with responsibilities in emergency preparedness and response, as relevant (see GSR Part 7 [6]). Emergency plans and procedures shall be based on the accidents analysed in the safety analysis report as well as those additionally postulated for the purposes of emergency preparedness and response on the basis of the hazard assessment. Emergency plans and procedures shall be subject to approval by the regulatory body, as appropriate.

7.91. All personnel involved in responding to an emergency in relation to the research reactor shall be qualified, trained and retrained periodically in accordance with their assigned duties and shall be fit for their intended duty (see GSR Part 7 [6]). The emergency response shall include persons with up to date knowledge of the operations of the research reactor, for example the reactor manager or a qualified delegate. All persons on the site shall receive instructions on the steps to take in an emergency. Instructions shall be prominently displayed.

7.92. Exercises to test emergency arrangements shall be conducted at suitable intervals and shall involve, to the extent practicable, all persons with duties in responding to the emergency. The results of the exercises shall be reviewed and, as necessary, the lessons learned shall be incorporated into revisions of the emergency arrangements. The emergency plans and procedures shall be periodically reviewed and shall be revised as necessary to ensure that feedback from experience and other changes (e.g. the contact details of emergency personnel) are incorporated.

7.93. Facilities, instruments, tools, equipment, documentation and communication systems to be used in an emergency, including those necessary for communication with off-site authorities, shall be kept available for use in a range of postulated emergencies. They shall be maintained in good operational condition such that it is unlikely that they would be affected by or made unavailable as a result of the accident or an initiating event. The operating organization shall ensure that the relevant information on the research reactor safety parameters and facility conditions is available in the emergency centre and that communication is effective between the control rooms and the emergency centre in the event of an accident. These capabilities shall be tested periodically.

**Requirement 82: Records and reports**

**The operating organization for a research reactor facility shall establish and maintain a system for the control of records and reports.**

7.94. 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.

7.95. Administrative procedures consistent with the management system 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.

7.96. 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, in accordance with regulatory requirements.

7.97. The arrangements made for storing and maintaining records and reports shall be in accordance with the management system. The document management system shall be designed to ensure that obsolete documents are archived and that personnel use only the most recent approved version of each document.

**Requirement 83: Utilization and modification of a research reactor**

**The operating organization for a research reactor facility shall establish and implement a programme to manage utilization and modifications of the reactor.**

7.98. The operating organization shall have the overall responsibility 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.

7.99. The operating organization shall be responsible for ensuring the following:

(a) Safety analyses of the proposed utilization or modification are conducted to ascertain whether all applicable safety requirements and provisions have been satisfied.
(b) The relevant safety documentation for the experiment or modification is prepared and presented (submitted) to the appropriate authority for approval.
(c) The disposition path of any materials irradiated in the experiment is defined and made subject to approval.
(d) All personnel who will be involved in making a proposed modification or in conducting the proposed utilization are suitably trained, qualified and experienced.
(e) All documents affected by the experiment or modification that relate to the safety characteristics of the reactor, such as the safety analysis reports, the operational limits and conditions, and the relevant procedures for operation, maintenance and emergencies, are updated as necessary, prior to the new utilization or to the commissioning of the modification.
(f) Safety precautions and controls are applied with regard to all personnel involved in the performance of the experiment or modification.

7.100. 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 either in accordance with 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 operational limits and conditions. Limiting conditions for safe operation (see para. 7.37) shall be prepared for the device and incorporated into the operational limits and conditions of the research reactor.

7.101. Utilization and modification projects (including temporary modifications, see para. 7.104) having major safety significance (see paras 3.13–3.20 of SSG-24 [15]) shall be subject to safety analyses and to procedures for design, construction and commissioning that are equivalent to those described in paras 6.119 and 6.121 for the reactor itself.

7.102. In implementing utilization and modification projects for a research reactor, the radiation exposure of the workers and other personnel at the facility shall be kept below authorized limits and as low as reasonably achievable.

7.103. The reactor manager shall establish a procedure, in accordance with accepted engineering practice, for the review and approval of proposals for experiments and modifications and for the control of their performance.

7.104. Temporary modifications shall be limited in time and number to minimize their cumulative safety significance. Temporary modifications shall be clearly identified at their location and at any relevant control position. The operating organization shall establish a formal system for informing relevant personnel in good time of temporary modifications and of their consequences for the operation and safety of the facility.

7.105. 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 or radiation levels, shall be taken into account in these procedures.

7.106. 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.

**Requirement 84: Radiation protection programme**

**The operating organization for a research reactor facility shall establish and implement a radiation protection programme.**

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

7.108. The radiation protection programme of the operating organization shall have sufficient independence and resources to be able to advise on and enforce radiation protection regulations, standards and procedures, and safe working practices.

7.109. The radiation protection programme shall be established by the operating organization in accordance with regulatory requirements. It shall comply with the requirements of GSR Part 3 [7] and shall be subject to the approval of the regulatory body. This programme shall include a policy statement from the operating organization that includes the fundamental safety objective of protecting people and the environment (see para. 2.1 of SF-1 [1] and Requirement 1 of GSR Part 3 [7]) and a statement of the operating organization's commitment to the principle of optimization of protection (Requirement 11 of GSR Part 3 [7]).

7.110. The radiation protection programme is subject to the requirements for occupational radiation protection (see GSR Part 3 [7] and RS-G-1.1 [16]) and shall include in particular measures for the following:

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

7.111. The operating organization shall verify, by means of surveillance, inspections and audits, that the radiation protection programme is being correctly implemented and that its objectives are being met. The radiation protection programme shall be reviewed on a regular basis and shall be updated if necessary.

7.112. To assist the reactor management in ensuring that radiation doses are kept as low as reasonably achievable, the operating organization shall establish dose constraints (see paras 1.22–1.28 and Requirement 11 of GSR Part 3 [7]).

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

7.114. All personnel who may be occupationally exposed to radiation at significant levels shall have their doses measured, assessed and recorded, as required by the regulatory body or other competent authorities, and these records shall be made available to the supervisor of the health surveillance programme, the reactor manager, the regulatory body and other competent authorities as designated in the national regulations [16].

**Requirement 85: Management of radioactive waste**

**The operating organization for a research reactor facility shall establish and implement a programme for the management of radioactive waste.**

7.115. The operating organization shall establish and implement a programme for the management of radioactive waste. The programme for the management of radioactive waste shall include the characterization, classification, processing (i.e. pretreatment, treatment and conditioning), transport, storage and disposal of radioactive waste.[46] Processing and storage of radioactive waste shall be strictly controlled in a manner consistent with the requirements for the predisposal management of radioactive waste [17]. Records shall be maintained for waste generation and waste classification.

7.116. The reactor and its experimental devices shall be operated to minimize the generation of radioactive waste of all kinds, to ensure that releases of radioactive material to the environment are kept below permissible regulatory limits and as low as reasonably achievable and to facilitate the handling and disposal of waste.

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

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

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

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[46] Part of this process for the characterization, classification, processing, transport, storage and disposal of radioactive waste could be carried out by another organization.

**Requirement 86: Ageing management**

**The operating organization for a research reactor facility shall ensure that an effective ageing management programme is implemented to manage the ageing of items important to safety so that the required safety functions of structures, systems and components are fulfilled over the entire operating lifetime of the research reactor.**

7.120. The ageing management programme shall determine the consequences of ageing and the activities necessary to maintain the operability and reliability of structures, systems and components. The ageing management programme shall be coordinated with, and be consistent with, other relevant programmes, including the programmes for in-service inspections, periodic safety review[47] and maintenance. A systematic approach shall be taken to provide for the development, implementation and continuous improvement of ageing management programmes.

**Periodic safety review**

7.121. On the basis of the results of the periodic safety review, the operating organization shall take any necessary corrective actions and shall consider making justified modifications to enhance safety (see also para. 7.120 on the interaction between ageing management and periodic safety reviews).

7.122. The operating organization shall report to the regulatory body as required, in a timely manner, the confirmed findings of the periodic safety review that have implications for safety.

**Requirement 87: Extended shutdown**

**If an extended shutdown is planned or occurs, the operating organization for a research reactor facility shall establish and implement arrangements to ensure the safe management, planning, effective performance and control of work activities during the extended shutdown.**

[47] Periodic safety review is a systematic reassessment of the safety of an existing facility (or activity) carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience, technical developments and siting aspects, and aimed at ensuring a high level of safety throughout the service life of the facility (or activity) [8].

7.123. A research reactor facility may have a period of extended shutdown[48] pending a decision on its future. 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 appropriate and safe storage conditions;
(b) Changing the operational limits and conditions in accordance with the requirements for the shutdown 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 maintenance, periodic testing and inspection.

7.124. The operating organization shall be responsible for establishing programmes and issuing procedures for managing extended shutdown and for the provision of adequate resources for ensuring the safety of activities during extended shutdown. Priority shall be given to safety related considerations in the processes for planning and performing activities in the extended shutdown state. Special attention shall be given to maintaining the reactor configuration up to date in accordance with the operational limits and conditions.

7.125. 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 need to meet licence conditions, and requirements for emergency planning and for the qualification of the operating staff. Security shall be provided for as long as nuclear fuel or other radioactive material is present at the facility.

**Requirement 88: Feedback of operating experience**

**The operating organization for a research reactor facility shall establish a programme to learn from events at the reactor facility and events in other research reactors and from the nuclear industry.**

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[48] A research reactor in extended shutdown is one that is no longer operating, with no decision on its decommissioning, and where there is no clear decision about the future of the reactor as to whether it will be brought back into operation or decommissioned. Long shutdown periods for maintenance or for implementation of refurbishment and modification projects are not considered an extended shutdown state.

7.126. The operating organization shall report, collect, screen, analyse, trend, document and communicate operating experience at the reactor facility in a systematic way. It shall obtain and evaluate available information on relevant operating experience at other nuclear installations to draw and incorporate lessons for its own operations, including its emergency arrangements. It shall also encourage the exchange of experience within national and international systems for the feedback of operating experience. These activities shall be performed in accordance with the management system.

7.127. Events with significant implications for safety shall be investigated to identify their direct and root causes, including causes relating to equipment design, operation and maintenance, or to human and organizational factors. The results of such analyses shall be included, as appropriate, in relevant training programmes and shall be used in reviewing procedures and instructions.

7.128. Information on operating experience shall be examined by competent persons for any precursors to, or trends in, adverse conditions for safety so that any necessary corrective actions can be taken before serious conditions arise.

7.129. The operating organization shall maintain liaison, as appropriate, with support organizations (manufacturers, research organizations and designers) involved in the design, in order to feed back information on operating experience and to obtain advice, if necessary, in the event of equipment failure or in other events.

# 8. PREPARATION FOR DECOMMISSIONING OF A RESEARCH REACTOR

**Requirement 89: Decommissioning plan**

**The operating organization for a research reactor facility shall prepare a decommissioning plan and shall maintain it throughout the lifetime of the research reactor, unless otherwise approved by the regulatory body, to demonstrate that decommissioning can be accomplished safely and in such a way as to meet the specified end state.**

8.1. The decommissioning plan shall be prepared at the design stage and shall be updated in accordance with changes in regulatory requirements, modifications to the structures, systems and components, advances in technology, changes in the need for decommissioning activities and changes in national policies for decommissioning and/or the management of radioactive waste [11].

8.2. The decommissioning plan shall be submitted for review by the safety committee and for approval by the regulatory body before decommissioning activities are commenced.

8.3. Documentation of the reactor shall be kept up to date and information on experience with the handling of contaminated or activated structures, systems and components in the maintenance or modification of the reactor shall be recorded to facilitate the planning of decommissioning. For some operating research reactors, where the need for their ultimate decommissioning was not taken into account in their design, a decommissioning plan shall be prepared to ensure safety throughout the decommissioning process.

8.4. 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 accepted 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 contaminated components and radioactive waste;
(b) Removal of all radioactive material and all removable activated and contaminated components from the reactor and the thorough decontamination of the remaining structures to permit the unrestricted use of the facility.

8.5. In developing the decommissioning plan, aspects of the reactor's design, including those aspects that are particularly challenging to decommissioning, shall be reviewed. In addition, all aspects of the facility's operation that are important in relation to decommissioning shall be reviewed. These include any unintentional contamination whose cleanup has been deferred until the reactor's decommissioning, and any modifications that might not have been fully documented. 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.

8.6. 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 such procedures are not in place.

8.7. The operating organization shall be responsible for the preservation of knowledge of the reactor facility and for the retention of key personnel to facilitate decommissioning.

8.8. The implications for safety of the activities in the transition period, if any, between permanent shutdown of operation and approval of the final decommissioning plan shall be assessed and shall be managed so as to avoid undue hazards and to ensure safety.

## 9. INTERFACES BETWEEN SAFETY AND SECURITY FOR A RESEARCH REACTOR

**Requirement 90: Interfaces between nuclear safety and nuclear security**

**The interfaces between safety and security for a research reactor facility shall be addressed in an integrated manner throughout the lifetime of the reactor. Safety measures and security measures shall be established and implemented in such a manner that they do not compromise one another.**

9.1. The nuclear security fundamentals are provided in Ref. [18] and recommendations on nuclear security are provided in Ref. [13]. In discharging its prime responsibility for safety, the operating organization shall design, implement and maintain technical and administrative measures to achieve the regulatory requirements relating to the interfaces between safety and security, to maintain coordination with State organizations that are involved in safety and security, and to ensure availability of adequate trained personnel with knowledge and skills relating to the interfaces between safety and security, as part of the management system (see also Section 4).

9.2. The general safety requirements on the interfaces between safety and security in the areas of regulatory supervision and the management system are established respectively in GSR Part 1 (Rev. 1) [3] and GSR Part 2 [4]. These requirements apply to research reactors with the appropriate use of a graded approach.

9.3. Adequate measures shall be established by the operating organization at all stages in the lifetime of the research reactor to ensure effective communication and coordination among individuals with different objectives and backgrounds to ensure that safety measures and security measures do not compromise one another.

9.4. The selection of a research reactor site shall be based on both safety and security related criteria. Recommendations on the interfaces between safety and security in site selection and site evaluation for nuclear installations, including research reactors, are provided in Ref. [13].

9.5. Interfaces of nuclear safety with nuclear security and safeguards in the design of a research reactor are addressed by Requirement 11 (see also Requirement 39 on the prevention of unauthorized access).

9.6. A change control process shall be established to ensure that any proposed changes of design, including new experimental facilities, of the layout of the research reactor facility or of procedures are evaluated to verify that they do not jeopardize safety or security.

9.7. During the construction stage and during major modifications of a research reactor, access to the site by a large number and diversity of workers and other personnel is typical. In this regard, measures shall be implemented to prevent the inadvertent or intentional introduction of weaknesses, devices or any threat that could lead to a security breach or radioactive releases during operation and utilization of the reactor.

9.8. Adequate measures shall be implemented during the operation stage to ensure effective management of the interfaces between safety and security. Particular emphasis shall be placed on the activities relating to fuel handling and storage and the management of radioactive waste and spent fuel, emergency preparedness and response (see GSR Part 7 [6]), access control procedures, and operating procedures for reactor utilization, maintenance, periodic testing and inspection. These procedures shall be developed with the aim of ensuring an appropriate balance between safety and security. Specific arrangements shall be established to ensure the safety and security of fuel in case of long shutdown periods and for research reactors in extended shutdown.

## Appendix I

## SELECTED POSTULATED INITIATING EVENTS FOR RESEARCH REACTORS

I.1. The following are examples of selected postulated initiating events for research reactors.[49] Certain research reactors may have additional postulated initiating events depending on specific characteristics of the design:

(a) Loss of electrical power supplies:
  - Loss of normal electrical power.[50]

(b) Insertion of excess reactivity:
  - Criticality during fuel handling and loading (due to an error in fuel insertion);
  - Startup accident;
  - Control rod failure or control rod follower failure;
  - Control drive failure or control drive 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 or hot water;
  - Changes in the moderator (e.g. voids, leakage of $D_2O$ into $H_2O$ systems or leakage of $H_2O$ into $D_2O$ systems);
  - Effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material);
  - Insufficient shutdown reactivity;
  - Inadvertent ejection of control rods;
  - Maintenance errors with reactivity devices;
  - Spurious control system signals;
  - Removal of poisons from the coolant or moderator.

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[49] Some of the postulated initiating events listed are not relevant for subcritical assemblies.

[50] Although a loss of normal electrical power is not considered an initiating event, consideration has to 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 (e.g. a drop in voltage may cause devices to fail at different times).

(c) 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);
- — Effect of the failure or mishandling of an experiment;
- — Rupture of the primary coolant boundary leading to a loss of flow;
- — Fuel channel blockage or flow reduction (e.g. due to foreign material);
- — Improper power distribution due to, for example, unbalanced rod positions in core experiments or in 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).

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

(e) 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 or dropping of a transfer flask onto fuel);
- — Failure of the emergency core cooling system;
- — Malfunction of the reactor power control;
- — Criticality in fuel in storage;
- — Failure of the means of confinement, including the ventilation system;
- — Loss of coolant to fuel in transfer or storage;
- — Loss or reduction of proper shielding;
- — Failure of experimental apparatus or material (e.g. loop rupture);
- — Exceeding of fuel ratings.

(f) Special internal events:
- — Internal fires or explosions, including internally generated missiles;
- — Internal flooding;
- — Loss of support systems;
- — Security related incidents;
- — Malfunctions in reactor experiments;
- — Improper access by persons to restricted areas;
- — Fluid jets or pipe whip;
- — Exothermic chemical reactions;
- — Drop of heavy loads.

(g) External events:
  - Earthquakes (including seismically induced faulting and landslides);
  - Flooding (including failure of an upstream or downstream dam and blockage of a river and damage due to a tsunami or high waves);
  - Tornadoes and tornado missiles;
  - Sandstorms;
  - Hurricanes, storms and lightning;
  - Tropical cyclones;
  - Explosions;
  - Aircraft crashes;
  - Fires;
  - Toxic spills;
  - Accidents on transport routes (including collisions into the research reactor building);
  - 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;
  - Electromagnetic interference (e.g. from solar events);
  - Lightning strikes;
  - Power or voltage surges on the external supply line.

(h) Human errors.

## Appendix II

## OPERATIONAL ASPECTS OF RESEARCH REACTORS WARRANTING PARTICULAR CONSIDERATION

II.1. This appendix highlights operational aspects of research reactors that warrant particular consideration.

### REACTIVITY AND CRITICALITY MANAGEMENT

II.2. The core configuration of a research reactor is frequently changed 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 shall 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 affect the nuclear and thermal characteristics of the core. Measures shall be established to ensure, for each change, that 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. Measures shall be taken to ensure that the technical, nuclear and operational characteristics of experimental devices are adequately assessed for their safety implications and that this assessment is suitably documented.

## MODIFICATION OF RESEARCH 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 attention shall be given to the need to verify that every modification has been properly assessed, documented and reported in terms of its potential effects on safety, and that the research 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. Care shall be taken to ensure that the operating personnel carrying out these manipulations 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 the research reactor may have access to controlled areas and may be actively involved in the operation or utilization of the reactor. Measures such as procedures, restrictions and controls shall be established to ensure that visitors have safe working conditions, that their activities will not affect the safety of the reactor and that safety instructions are strictly observed.

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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 structures, systems and components for ensuring the safety of the reactor. The safety functions will depend on 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 structures, systems and components. The safety functions that each structure, system or component 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 | To form a barrier to the uncontrolled release of radioactive material to the environment |
| | To provide protection against external and internal events for the enclosed safety systems |
| | To provide shielding against radiation |
| Reactor core | 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 design basis accidents |
| | To provide negative feedback of reactivity |
| | To provide a means of moderating and controlling neutron fluxes |
| Fuel matrix and cladding | To form a barrier to the release of fission products and other radioactive material from the fuel |
| | To provide a coolable fuel 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 design basis accidents |

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 design basis accidents |
| 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 | To take protective actions to shut down the reactor and to cool and contain radioactive material, and to mitigate the consequences of accidents<br>To control interlocks to protect against operational errors if the required conditions have not been met |
| Other safety related instrumentation and control systems | To keep reactor parameters within operational limits without reaching safety limits<br>To provide and present to the reactor operator sufficient 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 | To minimize radiation exposure<br>To prevent inadvertent criticality<br>To limit any rise in fuel temperature<br>To store fresh and irradiated fuel<br>To prevent mechanical or corrosive damage of fuel |
| Radiation monitoring system | To provide measurements and warnings to minimize the radiation exposure of operating personnel 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

## OVERVIEW OF THE APPLICATION OF THE SAFETY REQUIREMENTS TO SUBCRITICAL ASSEMBLIES

II–1. Subcritical assemblies are of a variety of designs, operating arrangements and utilization programmes. Owing to this, all the overarching requirements (Requirements 1 to 90) are applicable to subcritical assemblies with the use of a graded approach that is commensurate with the potential hazard of the facility. Specifically, para. 1.9 states that "Each case in which the application of requirements is graded shall be identified, with account taken of the nature and possible magnitude of the hazards presented by the given facility and the activities conducted." The factors to be considered in deciding whether the application of certain requirements may be graded are set out in para. 2.17.

II–2. As a consequence, the way in which the requirements are applied for high performance subcritical assemblies might be different from that for subcritical assemblies with low potential hazards. In particular, for subcritical assemblies with natural uranium fuel with light water reflector or moderator, application of the safety requirements, including those relating to the licensing process, can be significantly graded (i.e. in view of the negligible radiation risk of certain subcritical assemblies, certain requirements might not need to be applied). It thus follows that the national authorization process for the use of radioactive material, developed and applied in accordance with IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [II–1], may be adequate for safety demonstration.

## REFERENCE TO ANNEX II

[II–1] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

# DEFINITIONS

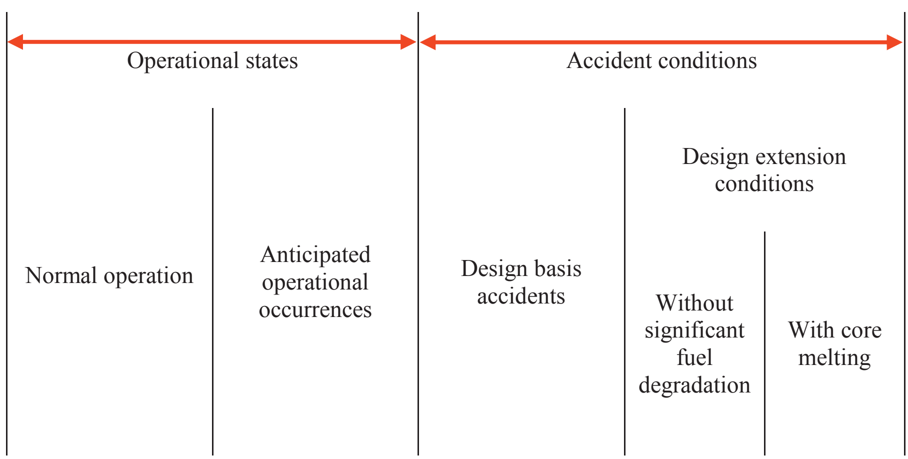
*The following definitions differ from those in the IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection (2007 Edition), IAEA, Vienna (2007): http://www-pub.iaea.org/books/IAEABooks/7648/IAEA-Safety-Glossary*

*The 2016 revision of the IAEA Safety Glossary is available at http://www-ns.iaea.org/standards/safety-glossary.asp*

*The symbol '*ⓘ*' denotes an information note.*

**controlled state.** State of the reactor facility, following an anticipated operational occurrence or accident conditions, in which fulfilment of the main safety functions can be ensured and which can be maintained for a time sufficient to implement provisions to reach a safe state.

**facility states** *(postulated states of a research reactor facility as considered for design purposes)*




**accident conditions.** Deviations from normal operation that are less frequent and more severe than anticipated operational occurrences, and which comprise design basis accidents and design extension conditions.

**design basis accident.** A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.

**design extension conditions.** Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.

ⓘ Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with melting of the reactor core.

**safe state.** State of the reactor facility, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the main safety functions can be ensured and maintained stable for a long time.

**safety feature (for design extension conditions).** Item that is designed to perform a safety function for or that has a safety function for design extension conditions.

**safety system settings.** Settings for levels at which safety systems are automatically actuated in the event of anticipated operational occurrences or design basis accidents, to prevent safety limits from being exceeded.

# CONTRIBUTORS TO DRAFTING AND REVIEW

| | |
|---|---|
| Abou Yehia, H. | Institute for Radiological Protection and Nuclear Safety, France |
| Adams, A. | Nuclear Regulatory Commission, United States of America |
| Boado Magán, H. | Consultant |
| D’Arcy, A.J. | South African Nuclear Energy Corporation, South Africa |
| Deitrich, L.W. | Consultant |
| Hargitai, T. | International Atomic Energy Agency |
| Hirshfeld, H. | Israel Atomic Energy Commission, Israel |
| Polyakov, D. | Rostechnadzor, Russian Federation |
| Sears, D.F. | International Atomic Energy Agency |
| Shokr, A.M. | International Atomic Energy Agency |

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