

IAEA Safety Standards

for protecting people and the environment

Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report

Specific Safety Guide

No. SSG-20 (Rev. 1)



IAEA

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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

<https://www.iaea.org/resources/safety-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 ASSESSMENT FOR
RESEARCH REACTORS AND
PREPARATION OF THE
SAFETY ANALYSIS REPORT

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

IAEA SAFETY STANDARDS SERIES No. SSG-20 (Rev. 1)

SAFETY ASSESSMENT FOR RESEARCH REACTORS AND PREPARATION OF THE SAFETY ANALYSIS REPORT

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2022

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FOREWORD

by Rafael Mariano Grossi
Director General

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

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

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

Safety Fundamentals

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

Safety Requirements

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

Safety Guides

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

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

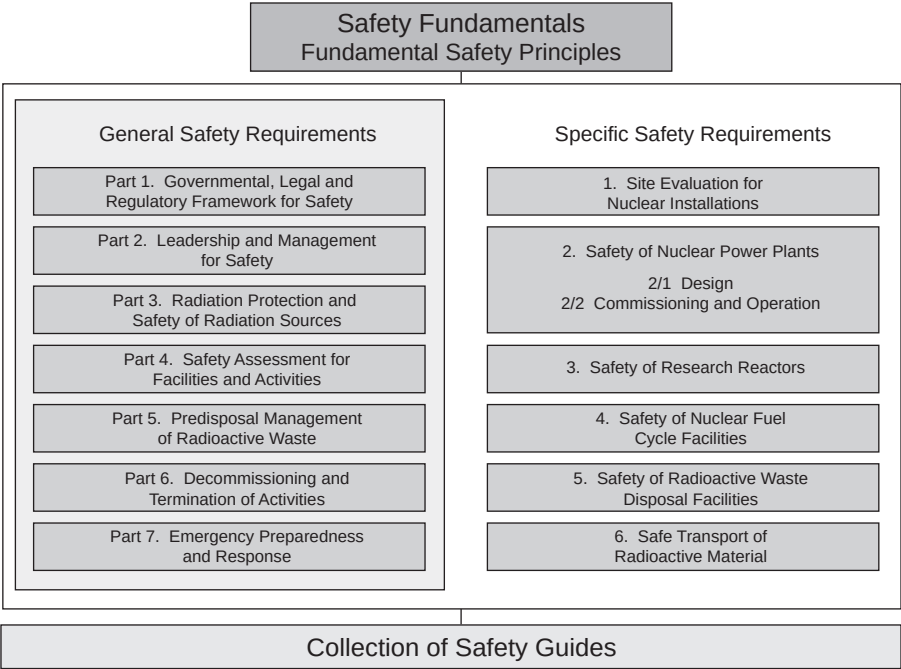


FIG. 1. The long term structure of the IAEA Safety Standards Series.

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

APPLICATION OF THE IAEA SAFETY STANDARDS

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

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

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

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

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

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

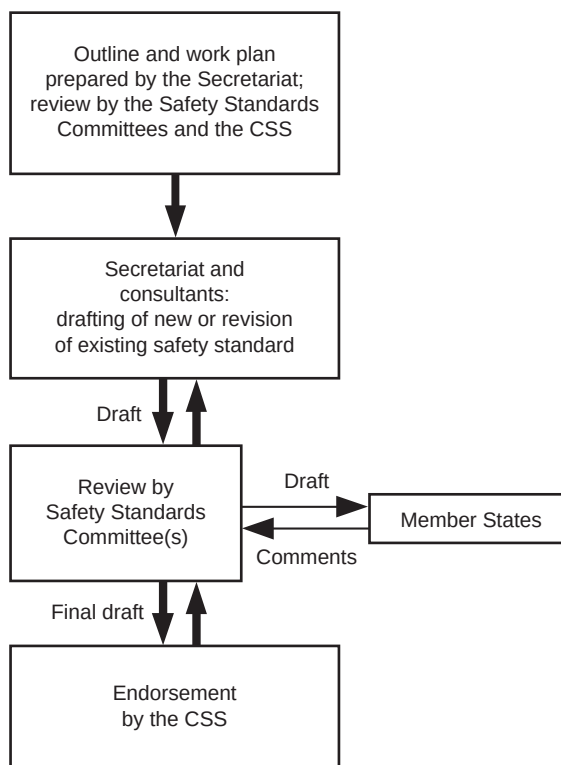


FIG. 2. The process for developing a new safety standard or revising an existing standard.

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

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

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

INTERPRETATION OF THE TEXT

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

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

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

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

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1. INTRODUCTION

BACKGROUND

1.1. Requirements for the safety of research reactors, with particular emphasis on their design and operation, are established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [1]. This Safety Guide provides recommendations on safety assessment and preparation of the safety analysis report for research reactors. This Safety Guide was developed in parallel with several other Safety Guides on the safety of research reactors, as follows:

- IAEA Safety Standards Series No. SSG-24 (Rev. 1), Safety in the Utilization and Modification of Research Reactors [2];
- IAEA Safety Standards Series No. SSG-80, Commissioning of Research Reactors [3];
- IAEA Safety Standards Series No. SSG-81, Maintenance, Periodic Testing and Inspection of Research Reactors [4];
- IAEA Safety Standards Series No. SSG-82, Core Management and Fuel Handling for Research Reactors [5];
- IAEA Safety Standards Series No. SSG-83, Operational Limits and Conditions and Operating Procedures for Research Reactors [6];
- IAEA Safety Standards Series No. SSG-84, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [7];
- IAEA Safety Standards Series No. SSG-85, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors [8];
- IAEA Safety Standards Series No. SSG-10 (Rev. 1), Ageing Management for Research Reactors [9];
- IAEA Safety Standards Series No. SSG-37 (Rev. 1), Instrumentation and Control Systems and Software Important to Safety for Research Reactors [10];
- IAEA Safety Standards Series No. SSG-22 (Rev. 1), Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [11].

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

1.3. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report¹.

OBJECTIVE

1.4. The objective of this Safety Guide is to provide recommendations on safety assessment for research reactors in the authorization process and on performance of the safety analysis and preparation of the safety analysis report, to meet the relevant requirements of SSR-3 [1]. It also provides recommendations on meeting the requirements for conducting the safety assessment as established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [13].

1.5. The recommendations provided in this Safety Guide are intended for operating organizations of research reactors; the Safety Guide can also be used by designers performing a safety assessment for a research reactor. Furthermore, this Safety Guide provides useful guidance for regulatory bodies performing a review and assessment of submitted safety analysis reports as an important document within an authorization process.

SCOPE

1.6. This Safety Guide is primarily intended for use for heterogeneous, thermal spectrum research reactors having a power rating of up to several tens of megawatts. For research reactors of higher power, specialized reactors (e.g. homogeneous reactors, fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops), additional guidance may be needed. For such research reactors, the recommendations provided in IAEA Safety Standards Series Nos SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [14], and SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants [15], might be more suitable.

1.7. Research reactors with low hazard potential having a power rating of up to several tens of kilowatts, and critical assemblies and subcritical assemblies

¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report, IAEA Safety Standards Series No. SSG-20, IAEA, Vienna (2012).

might need a less comprehensive safety assessment and safety analysis report than that outlined here. While all recommendations in this Safety Guide are to be considered, some might not be applicable to those research reactors with low hazard potential (see paras 2.15–2.17 and Requirement 12 of SSR-3 [1], and SSG-22 (Rev. 1) [11]).

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

1.9. This publication covers aspects of research reactor operation that are normally included in the safety analysis report, such as operational limits and conditions², commissioning, operating procedures, and utilization and modification; more detailed recommendations on these aspects of research reactor operation are provided in other Safety Guides [2–11].

1.10. This Safety Guide provides recommendations on carrying out a safety assessment during the initial design process and for design modifications, as well as for independent verification of the safety assessment of a new research reactor of a new or existing design. However, the recommendations are also applicable for a revised and updated safety assessment of an existing research reactor, for example in the context of a relicensing process. Recommendations on safety assessment for decommissioning of facilities are provided in IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [16].

1.11. This Safety Guide provides recommendations relating to utilization (i.e. for experiments and experimental facilities³) only with regard to safety analyses for the safety analysis report for the reactor. Detailed recommendations on safety analyses for experiments at research reactors and experimental facilities are provided in SSG-24 (Rev. 1) [2].

² The terms ‘safety specifications’, ‘technical specifications (tech. specs) for safe operation’ and ‘general operating rules’ are used by operating organizations and by regulatory bodies for nuclear reactors in some States instead of the term ‘operational limits and conditions’. These expressions usually cover safety limits, safety system settings, limiting conditions for safe operation, surveillance requirements and administrative requirements.

³ Experimental facilities include any device installed in or around a reactor to utilize the neutron flux and ionizing radiation from the reactor for research, development, isotope production or any other purpose.

1.12. Detailed recommendations on nuclear security are not provided in this Safety Guide. However, recommendations are provided on the interfaces between nuclear safety and nuclear security and on the handling of confidential information. In general, documentation and electronic records relating to safety analysis processes and outputs provide limited information regarding equipment location and facility layout. However, such information needs to be reviewed to identify any sensitive information that could be used to support malicious acts, and such information needs to be protected appropriately. Guidance on sensitive information and information security is provided in IAEA Nuclear Security Series Nos 23-G, Security of Nuclear Information [17], and 17-T (Rev. 1), Computer Security Techniques for Nuclear Facilities [18].

STRUCTURE

1.13. Section 2 describes the authorization process by which the safety of the research reactor and the issuing of licences are controlled and determined. Section 3 presents general recommendations on the preparation of the safety analysis report, in particular on the preparation of the safety analysis by the operating organization. Section 4 provides general recommendations on the information to be provided to the regulatory body to facilitate the process of review and assessment of the safety of the research reactor by the regulatory body. The Appendix provides recommendations on the standard content of the safety analysis report.

1.14. Annex I outlines the application of a basic approach to performing the safety analysis for a research reactor to analyse accidents, including their radiological consequences, while Annex II provides examples of input parameters and initial conditions for use in the safety analysis. Annex III deals with specific aspects of the reactor to be described in the safety analysis report. Finally, Annex IV provides a list of typical sources of radiation in a research reactor to be considered and described in the safety analysis report.

2. SAFETY ASSESSMENT IN THE AUTHORIZATION PROCESS

2.1. Safety assessment activities during the authorization process for a research reactor can be extensive. During the siting and design stages, safety assessment

can be iterative, confirming that the design meets acceptance criteria. Prior to the construction phase, the main safety assessment activities support the preparation of the safety analysis report and supporting documents and their submission for review by the regulatory body. Safety assessment activities are broad and necessitate specialist skills, and as such might involve several organizations to support the operating organization and the regulatory body (see GSR Part 4 (Rev. 1) [13]). Safety assessment continues throughout all stages of the research reactor's lifetime, conducted in accordance with the potential magnitude and nature of the hazard associated with the particular research reactor or activity (see Requirement 5 of SSR-3 [1]).

RESPONSIBILITIES

2.2. The government is required to establish an appropriate governmental, legal and regulatory framework for safety, which will provide the legal and regulatory basis for assessing the safety implications of a research reactor (see Requirements 1 and 2 of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [19]). The establishment of an independent regulatory body is an important requirement for an adequate legal and regulatory framework. Requirements on the establishment of a regulatory body are established in GSR Part 1 (Rev. 1) [19]. SSR-3 [1] also establishes requirements for the framework of the system for ensuring safety.

2.3. Compliance with the requirements imposed by the regulatory body does not relieve the operating organization of its prime responsibility for safety throughout the lifetime of the research reactor. The operating organization retains the responsibility for demonstrating to the satisfaction of the regulatory body that this prime responsibility has been, and will continue to be, adequately discharged. The prime responsibility for safety cannot be delegated. One of the ways the operating organization demonstrates that it has achieved adequate safety is through the information provided in the safety analysis report. This information also constitutes the prime basis for the regulatory body's decision on the authorization of the research reactor. A close liaison should be maintained between the regulatory body and the operating organization throughout the entire process of regulatory control over the research reactor.

2.4. The content of the application for a licence will depend on the legal and regulatory framework of the State. Relevant requirements for the authorization process are established in GSR Part 1 (Rev. 1) [19]. The information provided in support of a licence application should be commensurate with the magnitude of

the potential hazard associated with the research reactor and its utilization, and should be consistent with the particular stage of the authorization process.

2.5. Authorization is an ongoing process, starting at the stages of siting and site evaluation and continuing up to and including decommissioning and the release of the research reactor from regulatory control. The authorization process should be understandable to interested parties and should be predictable (i.e. well defined, clear, transparent and traceable). The different stages of the authorization process should be established in a coherent yet flexible way in order to achieve the most efficiency. These stages should be discrete and should follow a logical order. Detailed recommendations on the authorization process are provided in IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [20].

2.6. In all cases, the major stages of the authorization process for research reactors encompass the regulation of the following:

- (a) Siting and site evaluation;
- (b) Design;
- (c) Construction;
- (d) Commissioning;
- (e) Operation, including utilization and modification;
- (f) Decommissioning and release from regulatory control.

2.7. The operating organization is required to submit a demonstration of nuclear safety, including an adequate safety analysis, at each stage of the authorization process [1]; this demonstration of safety should be reviewed and assessed by the regulatory body before the next stage is authorized. Before each stage of the authorization process, the operating organization should revise the safety analysis report on the basis of feedback from the previous stages. Some States have adopted a ‘pre-licensing’ process. Such a pre-licensing process contributes to fostering a shared understanding of licensees, vendors and the regulatory body on the design concept, safety concepts, safety expectations and requirements to be fulfilled. Such an approach may help to minimize the duplication of effort at different stages of the authorization process and may allow for some stages to be conducted in parallel. It also provides for a clear division of responsibilities, at different stages, between the regulatory body, the vendor and the operating organization; gives the public opportunities for early participation; and ensures that the most important safety issues are dealt with properly at the pre-licensing stage (see para. 2.6 of SSG-12 [20]).

2.8. At all stages, the operating organization should be able to demonstrate that it has control over the research reactor and that it has an adequate organizational structure, a management system and adequate resources to discharge its obligations and, as appropriate, its liabilities. Further requirements and recommendations on the management system are provided in IAEA Safety Standards Series Nos GSR Part 2, Leadership and Management for Safety [21], GS-G-3.1, Application of the Management System for Facilities and Activities [22], and GS-G-3.5, The Management System for Nuclear Installations [23].

2.9. To meet the requirements for the review and assessment of information relevant to safety by the regulatory body (see GSR Part 1 (Rev. 1) [19]), the operating organization is required to submit to the regulatory body, in a timely manner, any information that the regulatory body has requested. The operating organization is required to make arrangements with vendors to ensure the availability of the information that has been requested by the regulatory body. The operating organization is also required to provide relevant new information to the regulatory body and to inform the regulatory body of any changes to information submitted previously (see Requirement 83 of SSR-3 [1]).

2.10. The format and content of documents submitted by the operating organization in support of an application for an authorization are required to be based on the requirements established in paras 3.6–3.9 of SSR-3 [1] and the recommendations presented in this Safety Guide. However, the regulatory body may require or may use additional information in the authorization process.

2.11. The review and assessment of information by the regulatory body is a continuous process. The safety analysis report or other documents with information appropriate for each stage of the authorization process (see para. 2.22) is required to be submitted to the regulatory body. A schedule for the submission of documents for review and assessment by the regulatory body is required to be established early in the research reactor project and made available to the operating organization.

2.12. The operating organization should revise all documentation associated with any modification or activity that might affect the safety of a research reactor (and all documentation having an indirect but significant influence on the safety of the research reactor), as appropriate. The revised documentation is required to be submitted to the regulatory body for its review and assessment (see para. 4.45 of GSR Part 1 (Rev. 1) [19]), with the potential magnitude and nature of the associated hazards being taken into account (see Requirement 26 of GSR Part 1 (Rev. 1) [19]).

2.13. The operating organization should submit information to the regulatory body, on the basis of which the regulatory body can determine whether the proposed research reactor can be sited, designed, constructed, commissioned, operated, utilized, modified, placed in extended shutdown or decommissioned without undue radiation risks to workers, the public or the environment. On the basis of the documentation submitted, the regulatory body should be able to do the following:

- (a) Understand the reactor design, the safety concepts on which it is based, and the management system and the approach to operational safety proposed by the operating organization.
- (b) Perform a review and assessment of the operating organization's technical submissions. This review and assessment should proceed from an overall survey of the reactor to an in-depth review and assessment of the design of individual structures, systems and components, and their performance in operational states and accident conditions.

Modifications to the reactor design or its utilization, or to the management system should also be submitted to the regulatory body for review and approval.

2.14. The primary basis for the review and assessment of the safety aspects of the proposed research reactor is the information contained in the safety analysis report submitted by the operating organization to the regulatory body. The safety analysis report should provide sufficient information for the regulatory body to determine the following:

- Whether the operating organization has provided the information that is both necessary and adequate for the purpose and scope of the review and assessment (see para. 4.2).
- Whether this information is in compliance with all applicable regulatory requirements.
- Whether this information is accurate. This might be determined by means of independent checks of the design, including calculations, and by inspections of the operating organization's facilities and management system.

ACCEPTANCE CRITERIA

2.15. In addition to the acceptance criteria established within the regulatory framework, the operating organization should develop additional acceptance criteria to demonstrate adequate application of the principles and objectives of

safe design and operation established in the IAEA safety standards, including the application of requirements for radiation protection (see IAEA Safety Standards Series Nos GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [24], and GSG-7, Occupational Radiation Protection [25]).

2.16. Acceptance criteria should be applied to judge the acceptability of the results of the safety analysis for operational states and accident conditions of the research reactor considered in its design. Acceptance criteria may be one of the following:

- A set of numerical limits on the values of predicted parameters;
- A set of conditions for facility states during and after an accident;
- A set of performance requirements on systems, including safety features for design extension conditions (see Requirement 22 of SSR-3 [1]);
- A set of requirements on the need for, and the ability to credit, actions by the operating organization; for design extension conditions these might include protective measures that are limited in terms of times and areas of application.

2.17. Acceptance criteria may be specified as ‘basic acceptance criteria’ or ‘specific acceptance criteria’. Basic acceptance criteria are aimed at achieving an adequate level of defence in depth. Examples include maximum allowable doses to the public and the prevention of fuel failure [26]. Specific acceptance criteria should include additional margins beyond the basic acceptance criteria as established within the regulatory framework, to allow for uncertainties. Specific acceptance criteria may be proposed by the operating organization and should be satisfactory to the regulatory body.

2.18. Acceptance criteria for design extension conditions without significant core degradation should be defined to ensure, with an adequate level of confidence, that core melting can be prevented, that there are adequate margins to avoid cliff edge effects⁴ and that there is no, or only minor, off-site radiological impact. In accordance with para. 6.68 of SSR-3 [1], conditions that could lead to an early

⁴ A cliff edge effect in a nuclear installation is an instance of severely abnormal system behaviour caused by an abrupt transition from one system status to another following a small deviation in a system parameter, and thus a sudden large variation in system conditions in response to a small variation in an input [12].

radioactive release or a large radioactive release⁵ are required to be practically eliminated, and acceptance criteria for design extension conditions with core melting should be defined in a way that ensures the mitigation of radiological consequences, as far as reasonably practicable. The analysis of design extension conditions might lead to the implementation of additional safety features, or to the extension of the capability of safety systems to fulfil the main safety functions and 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 the degradation of the reactor core.

2.19. In the development of the specific acceptance criteria, consideration should be given to the criteria listed below, as appropriate for the type of the research reactor:

- (a) Radiological criteria such as the following:
 - Dose limits and design target doses (see SSG-85 [8]) for public exposure;
 - Dose limits and design target doses (see SSG-85 [8]) for occupational exposure;
 - Radiation levels for accident conditions and for lifesaving actions in an emergency, consistent with Requirements 11 and 14 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [27];
 - Authorized limits on radioactive discharges to the environment during normal operation and acceptable limits on radioactive releases to the environment in accident conditions;
 - Risk criteria (where applicable).
- (b) Nuclear fuel performance criteria, including the following:
 - Maximum cladding temperature below blistering temperature;
 - Maximum heat flux not exceeding the critical heat flux during a transient;
 - Maximum heat flux not exceeding the onset of significant voiding during a transient;
 - Flow conditions not exceeding the onset of flow instability;
 - Maximum fuel and fuel cladding temperature below failure;
 - Other limits to prevent significant damage to fuel and fuel cladding failures.

⁵ An early radioactive release is a release for which off-site protective actions are necessary but are unlikely to be fully effective in due time. A large radioactive release is a release for which off-site protective actions that are limited in terms of times and areas of application are insufficient to protect people and the environment [12].

- (c) Performance criteria, including the following:
- Limits on parameters to prevent damage of the primary coolant system boundary;
 - Limits on parameters to prevent damage to systems important to safety caused by in-core or out-of-core experimental facilities;
 - Limits on parameters to prevent damage to containment systems;
 - Maintenance of core cooling;
 - Limits on parameters to prevent reactivity accidents;
 - Frequency limits for anticipated operational occurrences and for accident conditions.

2.20. Where the specific acceptance criteria mentioned above are determined not to be applicable to a research reactor with low hazard potential, a critical assembly or a subcritical assembly, the decision should be justified and documented. For subcritical assemblies, additional acceptance criteria may be specified for limits on insertion of reactivity that prevent criticality.

2.21. The following acceptance criteria should be included:

- An event should not generate a more serious condition of the research reactor without the occurrence of a further independent failure. Thus, an anticipated operational occurrence by itself should not generate a design basis accident, and a design basis accident by itself should not generate design extension conditions.
- There should be no consequential loss of function of the safety systems necessary to mitigate the consequences of an accident.
- Systems used for the mitigation of the consequences of accidents should be designed and constructed in accordance with their importance to safety and to withstand the maximum loads and stresses and the most extreme environmental conditions for the accident analysed.

INFORMATION REQUIREMENTS AT THE VARIOUS STAGES OF THE AUTHORIZATION PROCESS

2.22. The operating organization should provide the regulatory body with all relevant information on the safety of the research reactor. This information is normally presented in a safety analysis report, which is described comprehensively in the Appendix to this Safety Guide. Recommendations on the preparation and presentation of the safety analysis report are provided in Section 3, and recommendations on its review and assessment are provided in Section 4.

Paragraphs 2.26–2.50 provide recommendations on the information that is normally required for each stage of the authorization process. Sequential requests for information might lead to successive updating, with each version of the safety analysis report corresponding to a particular stage of the authorization process (see para. 3.4 of SSR-3 [1]).

2.23. The preparation of the safety analysis report should start as early as possible in the project, to allow the designers to derive the maximum benefit from the safety analysis and to allow the regulatory body to become familiar with the design and the safety features of the reactor. The amount of information provided in the safety analysis report, at each stage, should be sufficient to allow both the operating organization and the regulatory body to make a decision on the acceptability of the reactor for that stage.

2.24. At various stages in the course of the design process (e.g. before the start of construction, before the start of operation), the status of the design should be described in the safety analysis report, and the description should include the design and safety assessment that has been carried out up to that point (see GSR Part 4 (Rev. 1) [13]).

2.25. For research reactors with low hazard potential and particularly for critical assemblies and subcritical assemblies, the amount of information and analysis to be provided according to paras 2.26–2.50 could be reduced in accordance with a graded approach.

SUBMISSION OF INFORMATION TO THE REGULATORY BODY

Schedule for the submission of information

2.26. A schedule should be developed by the operating organization and agreed between the operating organization and the regulatory body that indicates the scope and timescale for the preparation of the safety analysis report. Since the approval of one authorization stage is normally required before commencement of the next stage, the timescale should include reasonable periods of time for each assessment phase such that they can be completed before commencement of the next phase (see paras 4.3 and 4.4).

Siting and site evaluation

2.27. The operating organization should provide sufficient information commensurate with the type, complexity and hazards associated with the research reactor to demonstrate to the regulatory body that the proposed site is suitable for the type and design of the proposed research reactor. Difficulties that will need to be resolved during the subsequent stages of the authorization process should be identified. Information on the site itself, and preliminary information on the research reactor and its interaction with the site and the surrounding environment, should be provided. In addition, a preliminary statement on the potential radiological impacts on site personnel, on the population in the surrounding area and on the environment should be provided. A radiological environmental impact assessment should be performed as a part of the authorization process (see IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [28]).

2.28. The characteristics of the site that might affect the safety of the research reactor should be investigated and assessed by the operating organization (see Requirement 8 of GSR Part 4 (Rev. 1) [13]). The objective of the assessment should be to assess how the site characteristics would influence the design and operation of the research reactor and to demonstrate the adequacy of the characteristics of the site from the point of view of safety. The requirements for the initial site evaluation and site selection, the general criteria for site evaluation and the external events that should be considered for site evaluation are provided in section 5 of SSR-3 [1]. Additional recommendations on siting and site evaluation are provided in the Appendix to this Safety Guide (see Chapter 3 of the safety analysis report on site characteristics) and in SSG-12 [20], and requirements on site evaluation are established in IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [29]. Nuclear security issues and their interface with safety should also be considered at the siting and site evaluation stage (see IAEA Nuclear Security Series No. 35-G, Security During the Lifetime of a Nuclear Facility [30]). The details on siting that should be addressed in the safety analysis report are presented in the Appendix (see Chapter 3 of the safety analysis report on site characteristics).

Design and construction

2.29. Before authorization of the construction of the research reactor, features such as the physical layout and the type of construction of the research reactor, as well as the key elements of the construction process, should be carefully considered, and their effects on the safety of the research reactor throughout its

lifetime should be assessed. At this stage, due consideration should be given to the selection of materials, to ageing mechanisms for materials and structures, systems and components, and to the effects of these ageing mechanisms on safety; additional recommendations on ageing management are provided in SSG-10 (Rev. 1) [9]. Consideration should also be given to nuclear security, including physical protection [31, 32], information security [17] and the interface of nuclear security with safety. The operating organization should describe the arrangements for the control of construction, manufacture and installation activities. The initial decommissioning plan prepared at the design stage should cover issues such as strategies to be applied, radiation doses to be expected and amounts of waste expected to be generated (see SSG-47 [16]). Information on the matters addressed in this paragraph should be submitted to the regulatory body for review and assessment.

2.30. To obtain a construction licence or an approval for the start of construction, the operating organization should submit to the regulatory body information that demonstrates that the design will result in a safe research reactor and that construction will achieve the design intent. The information should contain a description of the design of the research reactor and the associated structures, systems and components and their safety classification. It should also present the results of the safety analysis to demonstrate the adequacy of the design of structures, systems and components. This information should be included in the safety analysis report and should be updated as the design and construction proceed.

2.31. Those aspects of the design that should be submitted to the regulatory body for review and assessment before the design is finalized should be identified in agreement with the regulatory body. The information should be updated and resubmitted to the regulatory body as the detailed design of the reactor proceeds. In some cases, revised versions of documents will be sufficient; in other cases, technical supplements may be appropriate. Additional recommendations on the authorization process for this stage are provided in SSG-12 [20].

2.32. The safety analysis report is the main document provided at this stage for review and assessment by the regulatory body for the authorization of the detailed design and for construction.

Commissioning

2.33. When construction is at a sufficiently advanced stage, the information contained in the safety analysis report should be reviewed and, where necessary, updated. The updated safety analysis report should be resubmitted to the regulatory

body for review and assessment in order to obtain the required authorization for commissioning.

2.34. Paragraph 7.52 of SSR-3 [1] states that “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.” Every test sequence should be completed successfully, and the results should be submitted to the regulatory body for review and assessment. The test results should be made subject to approval by the operating organization at the appropriate level of management and, as necessary depending on the direct effect on safety, by the regulatory body before the subsequent test sequence is started. Paragraph 7.52 of SSR-3 [1] further states (footnote omitted):

“The commissioning programme shall...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...;
- (c) Stage C: Power ascension tests and power tests.”

Commissioning should be carried out in accordance with the commissioning programme that has been reviewed and assessed by the regulatory body. Detailed recommendations on the commissioning of research reactors are provided in SSG-12 [20] and SSG-80 [3].

2.35. For subcritical assemblies, the initial criticality tests and low power tests of Stage B and the tests of Stage C are not applicable. However, additional tests such as verification that the configuration is adequately subcritical and measurements of neutron flux should be performed. Such tests and measurements should be used to verify the results from computational models and tools that are used for design and safety analysis of the subcritical assembly.

2.36. The updated safety analysis report should include and demonstrate the adequacy of the commissioning programme. The ‘as-built’ reactor, the analyses of postulated accidents and the capability of safety systems to limit the consequences of postulated accidents should also be fully documented in the updated safety analysis report.

2.37. The commissioning procedures for a particular commissioning stage should be reviewed before the start of that commissioning stage and should be updated,

as necessary. The ‘as-built’ design of the research reactor and the results of the previous commissioning stages should be taken into account. The updated commissioning procedures should be submitted to the regulatory body for review and assessment to obtain the required authorizations for commissioning.

2.38. Stage A tests should be carried out to ensure that the reactor has been constructed, manufactured and installed correctly and in accordance with the design documentation. Any deviations from the design documentation should be recorded and assessed, and it should be shown that the safety analysis remains valid and that safety has not been compromised. The results of this stage should also confirm the operational features of the research reactor and should be used to develop operating procedures, which should be confirmed in Stages B and C.

2.39. Stage B is a significant step in the authorization process. The results of stage A of the commissioning programme should be available and the organizational structure of the operating organization, the qualifications of operating personnel, the radiation protection programme, the arrangements for emergency preparedness and response, the operational limits and conditions for commissioning, the preliminary operating procedures, and the arrangements for managing the interface between safety and nuclear security should all be in place at this commissioning stage, before the commencement of fuel loading. Whenever there are deviations from the design parameters, these should be analysed by the operating organization and reported to the regulatory body for review and assessment.

2.40. As Stage C moves closer to completion, this commissioning stage should focus on how the research reactor will be operated, utilized and maintained, and on procedures for controlling and monitoring operation and for responding to deviations and other occurrences. Before authorization for operation is requested, the test results, any corrections of non-conformances, modifications to the design or modifications to the operating procedures, and any proposed changes to the operational limits and conditions should be submitted to the regulatory body for review and assessment.

2.41. The information referred to in paras 2.33–2.40 should be updated after each commissioning stage, and submission of this information to the regulatory body should form the basis of the start of the next commissioning stage as a part of the authorization process.

Operation

2.42. In its application for an operating licence, the operating organization should submit to the regulatory body all the information listed in the preceding sections. Additional information to prove the capability for safe operation should also be submitted to the regulatory body. Some of this information will be required at the different stages of the authorization process, and some should be submitted after the formal licence has been obtained. Additional recommendations on the authorization stages are provided in SSG-12 [20].

2.43. The safety analysis report should be updated for the application for the authorization for operation. The results from the commissioning programme should be included in the application and assessed by the regulatory body to demonstrate that the design requirements have been met.

2.44. Systematic periodic safety reviews of the research reactor are required to be performed throughout its lifetime (see para. 4.25 of SSR-3 [1]). Such periodic reviews of the safety of the research reactor include periodic safety reviews required by the regulatory body (see paras 7.121 and 7.122 of SSR-3 [1]) and self-assessments performed by the operating organization. Such reviews should address important issues such as the cumulative effects of ageing of the research reactor. The nature of such reviews and the interval between reviews should reflect the risks that the research reactor presents. For such reviews, the existing safety analysis report should be compared with operating experience, including operating experience from accidents and information on radiation protection, modifications, experiments and other aspects of operation. If required as a result of a periodic safety review, the operating organization should submit to the regulatory body a request for an amendment of the licence, which should include a revised safety analysis report, as appropriate.

Utilization and modification

2.45. The operating organization should submit to the regulatory body for review and assessment information on experiments and modifications that might affect the safety of the research reactor. The specific requirements for submission will depend on the safety significance of the experiments and modifications. These requirements are set out in Requirement 83 of SSR-3 [1]. Specific recommendations on the development of appropriate procedures for the control of experiments and modifications, including review by the safety committee, are provided in SSG-24 (Rev. 1) [2] and SSG-83 [6].

2.46. Experiments and modifications having major safety significance should be categorized and subjected to procedures for design, construction, commissioning and safety analysis that are equivalent to those for the research reactor itself (see Requirement 83 of SSR-3 [1]). This safety analysis may need to be performed in stages according to the various phases of the modification project or utilization activity. These stages could be (a) pre-implementation (i.e. project initiation, project definition and design), (b) implementation (i.e. fabrication, installation and commissioning) and (c) post-implementation (i.e. operation and surveillance). Further recommendations on utilization and modification, including categorization of experiments and modifications, are provided in SSG-24 (Rev. 1) [2]. The safety aspects of each phase of the modification or utilization project should be analysed and presented in a separate safety analysis report for the modification or utilization activity, or the appropriate chapters of the existing safety analysis report for the research reactor should be revised and submitted to the regulatory body for review and assessment. In addition, the safety analysis report provides boundaries for operational limits and conditions that have been demonstrated to be safe, and any experiments and modifications should fall within these boundaries.

2.47. If applicable, the operating organization should revise the relevant acceptance criteria and should submit them to the regulatory body for review and assessment, and for approval for their use in the safety analysis of the proposed experiment or modification.

2.48. Commissioning of the experiment or of the modified research reactor should be conducted to demonstrate compliance with the design requirements in the safety analysis report. In addition, if changes to the safety analysis report or to some analyses are made, it should be ensured that all other safety analyses remain valid.

Decommissioning and release from regulatory control

2.49. The operating organization should submit the decommissioning plan and supporting documents to the regulatory body for review and assessment. The type of information and level of detail in the decommissioning plan, including safety assessment, will depend on the hazards associated with the decommissioning of the research reactor. Requirements on decommissioning of research reactors are established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [33].

2.50. At some point in the decommissioning process (e.g. after the removal of all fuel from the site), the safety analysis report ceases to be a major working document, and a safety assessment for decommissioning, commensurate with

the remaining hazards, should be prepared by the operating organization. Further recommendations on decommissioning are provided in SSG-47 [16], and recommendations on release from regulatory control are provided in IAEA Safety Standards Series No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices [34].

3. PREPARATION OF THE SAFETY ANALYSIS REPORT

PURPOSE AND SCOPE OF THE SAFETY ANALYSIS REPORT

3.1. The operating organization is required to prepare a safety analysis report for the research reactor (see Requirement 1 of SSR-3 [1]). The safety analysis report is to be submitted to the regulatory body for review and assessment as part of the authorization process (see para. 3.5 of SSR-3 [1]). The safety analysis report is required to provide the basis for the safe operation of the research reactor and should be the basis for the interaction between the operating organization and the regulatory body in the authorization process.

3.2. In addition, the preparation of a safety analysis report should also serve the following purposes:

- To aid the designer in confirming that individual systems are integrated correctly with one another, since the design of the research reactor and the development of the safety analysis report are complementary and interactive processes;
- To ensure that the safety analysis has properly identified the safety issues relevant to the design and that the safety analysis and the design are consistent;
- To aid in the understanding of the relevant design criteria, limitations and requirements, and in the evaluation of the hazards posed by the research reactor in operational states and accident conditions;
- To aid in the training and retraining of operating personnel and in their familiarization with the research reactor;
- To aid in the establishment of operational limits and conditions on certain parameters that have to be met at all stages of the lifetime of the reactor in order to ensure adequate margins of safety for the reactor;
- To aid in the establishment of the programme for maintenance, periodic testing and inspection and operating procedures;

- To identify ageing mechanisms and their effects on safety for the development of an ageing management programme;
- To aid in the understanding of the interface between safety and nuclear security;
- To aid the development and establishment of the decommissioning plan;
- To aid in the development and establishment of the emergency preparedness and response arrangements.

3.3. Over the lifetime of the research reactor, the safety analysis report should be continuously updated to describe the following:

- The evolution of the design, operation and utilization of the research reactor and the associated experimental facilities and devices, and any safety significant modifications to the research reactor;
- Changes made in the research reactor and its operation as a result of events that might have occurred during the lifetime of the research reactor or operating experience feedback, including from other nuclear installations, and that might influence the actions that will need to be taken in decommissioning the research reactor.

3.4. In accordance with para. 3.6 of SSR-3 [1], the safety analysis report is required to give a detailed description of the research reactor site, the research reactor itself, the experimental facilities and devices, and all other facilities with significance for safety. It is required to provide a detailed description of the general safety principles and criteria, as well as of the codes and standards applied in the design for the safety of the reactor, and for the protection of the operating personnel, the research reactor users, the public and the environment. The potential hazards associated with the operation of the research reactor are also required to be addressed in the safety analysis report. The safety analysis report is required to include the safety analyses of accident sequences and to describe the safety features incorporated into the design and the additional safety features for design extension conditions to prevent accidents or to mitigate their consequences through the design, as well as operating procedures and emergency procedures.

3.5. In accordance with para. 3.7 of SSR-3 [1], the safety analysis report is required to form the basis for the operational limits and conditions of the reactor. The operational limits and conditions should be incorporated into the authorization for operation. As a minimum, the safety analysis report should describe the content of the operational limits and conditions if they will be described in detail in a separate document. The safety analysis report is also required to provide details of the conduct of operations intended by the operating organization, including its

organizational structure and the management system procedures established for the design and operation of the research reactor. The safety analysis report is also required to provide information on the emergency arrangements for the research reactor, and it should set out the design provisions and operating procedures relating to decommissioning.

3.6. All topics treated in the Appendix to this Safety Guide should be adequately covered in the safety analysis report. Information on all these topics should be prepared in accordance with the corresponding recommendations in the Appendix. However, some of the topics may be addressed in separate documents (e.g. in the operational limits and conditions, operating procedures, physical protection plan, emergency plans and procedures, and the decommissioning plan). In this case, these topics should be treated briefly in the safety analysis report and reference made to the appropriate separate document.

SPECIFIC RECOMMENDATIONS ON THE SAFETY ANALYSIS REPORT

3.7. In accordance with Requirement 5 of SSR-3 [1], the operating organization is required to ensure that an independent verification of the safety assessment is performed by individuals or groups separate from those carrying out the design. The independent verification should be carried out under the responsibility of the operating organization by a team of experts who are independent of the designers and of those performing the safety assessment and who have not participated in any part of the design or the safety assessment. This verification should be conducted either by the operating organization or by another qualified organization on its behalf (see paras 4.66 and 4.67 of GSR Part 4 (Rev. 1) [13]). Irrespective of the process followed for the development and verification of the safety analysis, the operating organization remains responsible for the content, comprehensiveness and quality of the safety analysis (see Requirement 2 of SSR-3 [1]). This independent verification should be performed before the safety analysis report is submitted to the regulatory body for review and assessment as a part of the authorization process.

3.8. Whereas the safety assessment is a comprehensive study carried out by the designers, under the responsibility of the operating organization, throughout the design process to address all relevant safety requirements, the independent verification should be carried out by or on behalf of the operating organization as an activity separate from the reviews carried out within the design organization (see Requirement 21 of GSR Part 4 (Rev. 1) [13]).

3.9. The proposal for a research reactor and the application for an authorization should be subject to open public participation by means of regular meetings, formal hearings and/or other appropriate means of communication. For these purposes, the operating organization may have to develop a non-technical version of the safety analysis report that can be understood by the public, and that takes into account confidentiality aspects (see para. 3.12). Recommendations on public participation are provided in SSG-12 [20].

3.10. Paragraph 3.9 of SSR-3 [1] states:

“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.”

Such references need not be submitted together with the safety analysis report, but they should be retained by the operating organization or the designers so that they can be provided to the regulatory body upon request.

3.11. Some regulatory bodies request the assistance of a technical support organization or an independent peer review group in reviewing the safety analysis report. In this case, the results of the review may be reported directly to the regulatory body.

3.12. Certain information provided by the operating organization or its contractors should be considered confidential, because of its proprietary nature, for security reasons or because of the right of individuals to privacy, in accordance with national laws and regulations. Such confidential information should be made available, as necessary, without restriction to the regulatory body; that is, it should be made available to the regulatory body’s staff, to technical support organizations, consultants and advisory committees, and to any governmental bodies involved in the review and assessment process. The regulatory body should formally inform the operating organization which consultants and advisers will be involved on behalf of the regulatory body. Those persons to whom such information is to be entrusted should be advised of its confidential nature and should be obliged, consistent with national laws and regulations, to protect its confidentiality. If consultants, technical support organizations and external advisory committees need to have confidential documents at their disposal, a process to ensure confidentiality should be put in place.

3.13. Owing to the amount of documentation required to support a safety analysis report, a document control system should be established to manage the indexing and to control the issuing of the separate documents that make up the safety analysis report. The document control system should be used to control the updating, revision, issuing and removal of documents in accordance with the management system procedures, so that information is always kept up to date.

3.14. The type of research reactor, its site and its characteristics (e.g. design, power, utilization) might influence the extent of the information to be presented in the safety analysis report. Accident scenarios for research reactors with higher power levels or with a significant inventory of radioactive material should require more details to be provided about the site and about the safety features to protect against any significant release of radioactive material to the environment and to mitigate the consequences of such releases if they occur.

3.15. For low risk facilities — such as some critical assemblies, subcritical assemblies and research reactors with low hazard potential — these requirements are much less stringent. However, as the safety analysis report is often the only comprehensive document produced, every topic addressed in the Appendix should be considered. Although the extent of information on each topic would be limited, the scope of some topics (e.g. the protection of operating personnel against overexposure in a critical assembly facility) might still be large for facilities with low hazard potential.

DEVELOPMENT OF THE SAFETY ANALYSIS

3.16. The safety analysis, as part of the safety assessment used in the authorization of a research reactor, should proceed in parallel with the design process, with iteration between the two activities. The scope and level of detail of the safety analysis should increase as the design process progresses, so that the final safety analysis reflects the final design of the research reactor as constructed.

General considerations

3.17. The general requirements for the development of the safety analysis are established in SSR-3 [1]. To ensure that the safety analysis meets the intended objective, the detailed recommendations on the preparation of the safety analysis as presented in the Appendix to this Safety Guide (see Chapter 16 of the safety analysis report on safety analysis) should be taken into account.

3.18. The safety analysis should identify design basis accidents, design extension conditions without significant fuel degradation and design extension conditions with melting of the reactor core. In addition, accidents that have more severe consequences should be analysed for the purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident.

3.19. The safety analysis should be used mainly to enable the operating personnel to understand the basis for the safe operation of the reactor, and to demonstrate to the regulatory body the way in which the design of the research reactor and the related operating and emergency procedures will contribute to the prevention of accidents or mitigation of the consequences of accidents. The safety analysis should include analyses of the response of the reactor to the postulated initiating events listed in para. 3.23. The safety analysis should also serve as a basis for the determination of the operational limits and conditions, the safety classification of the structures, systems and components, and the development of the accident management procedures and the emergency plan.

3.20. The consideration of possible incidents during operational states and accident conditions should determine the design of the research reactor and the design limits for the safety systems and for most structures, systems and components necessary for the operation of the research reactor. The incidents during operational states and accident conditions should also be considered in the operating instructions and procedures for operating personnel. In addition, the potential radiological consequences of incidents during operational states and accident conditions for workers, the public and the environment is typically more severe than the radiological consequences of normal operation. For this reason, an important part of the effort in the peer review and in the verification by the operating organization should focus on the safety analysis of incidents during operational states and accident conditions. The scope and extent of this analysis should be in accordance with the magnitude and nature of the hazards associated with the particular research reactor. Safety analysis may be considered to consist of the following major steps:

- Identification and selection of the postulated initiating events;
- Categorization of the postulated initiating events;
- Determination of enveloping postulated initiating events;
- Evaluation of the sequence of events, development of the postulated initiating events in relation to system responses and their consequences;
- Comparison against acceptance criteria.

3.21. The following aspects should be verified in the safety analysis:

- That sufficient defence in depth has been provided, that the levels of defence are independent and preserved to the extent possible and that the potential accident sequences are arrested as early as possible.
- That the research reactor can withstand the physical and environmental conditions that it would experience. These would include extreme environmental conditions and other extreme conditions.
- That human factors and human performance issues have been adequately addressed (see IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [35]).
- That long term ageing mechanisms that could detract from the reliability of structures, systems and components over their design life are identified, monitored and managed (i.e. by upgrading, refurbishment or replacement), so that safety is not affected and risks do not unacceptably increase.

Identification, categorization and grouping of postulated initiating events

3.22. The identification and selection of the postulated initiating events should be the first step of the safety analysis. The selection method used should be systematic and should be auditable. Moreover, as complete as possible a listing of postulated initiating events should be established. An important feature of the review and assessment process should be to consider whether the method of identification of postulated initiating events meets these recommendations and whether the list of postulated initiating events is acceptable as the basis for the safety analysis. The use of systematic techniques such as hazard and operability studies or failure modes and effects analysis, among others, could facilitate the selection process.

3.23. Typical examples of postulated initiating events leading to event sequences categorized as anticipated operational occurrences, design basis accidents and design extension conditions include those given below, sorted by types of sequence. This list is broadly indicative. The actual list will depend on the type of reactor, actual design and potential hazards associated with the research reactor⁶. The list of selected postulated initiating events is based on appendix I to SSR-3 [1]:

⁶ Depending on national regulations and interface arrangements with nuclear security, the list may be complemented by relevant nuclear security events.

- (a) Loss of electrical power supplies:
 - Loss of normal electrical power.⁷
- (b) Insertion of excess reactivity:
 - Criticality during fuel handling and loading (e.g. owing 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 (e.g. a moderator, a reflector);
 - Unbalanced rod positions;
 - Failure or collapse of structural components;
 - Insertion of cold or hot water;
 - Changes in the moderator material (e.g. voids, leakage of D₂O into H₂O systems, leakage of H₂O into D₂O systems);
 - Effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material, 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.
- (c) Loss of flow:
 - Primary pump failure;
 - Reduction in flow of primary coolant (e.g. owing 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. owing to foreign material);
 - Improper power distribution owing to, for example, unbalanced rod positions, in in-core experiments or in fuel loading (e.g. power flow mismatch);
 - Reduction in coolant flow owing to bypassing of the core;
 - Deviation of system pressure from specified limits;
 - Loss of heat sink (e.g. owing to the failure of a valve or a pump, or a system rupture).
- (d) Loss of coolant:
 - Rupture of the primary coolant boundary;

⁷ Although it is not considered an initiating event, consideration needs to be given to the loss of normal power followed by the loss of emergency power, to ensure that the consequences would be acceptable under emergency conditions.

- 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, dropping of a transfer flask onto the fuel);
 - Failure of the engineered safety features (e.g. 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;
 - Malfunction in reactor experiments;
 - Improper access by persons to restricted areas;
 - Fluid jets and pipe whip;
 - Exothermic chemical reactions;
 - Drop of heavy loads.
- (g) External events⁸:
- Earthquakes (including seismically induced faulting and landslides);
 - Flooding (including failure of an upstream or downstream dam and blockage of a river and damage owing to a tsunami or high waves);
 - Tornadoes and tornado missiles;
 - Sandstorms;
 - Hurricanes, storms and lightning;
 - Tropical cyclones;
 - Explosions;
 - Aircraft crashes;
 - Fires;
 - Toxic spills;

⁸ The possibility of extreme weather conditions associated with climate change needs to be taken into account for the determination of the external events.

- Accidents on transport routes (including collisions into the research reactor building);
 - Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities, 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.

3.24. The list of postulated initiating events specified in para. 3.23 should be reviewed for applicability to subcritical assemblies. The resulting list of postulated initiating events should be justified and documented for the specific configuration of the research reactor. For example, the following postulated initiating events might not be applicable to some subcritical assemblies, depending on their specific design features:

- (1) Loss of electrical power supplies.
- (2) Insertion of excess reactivity:
 - Control rod failure or control rod follower failure;
 - Control drive failure or control drive system failure;
 - Failure of other reactivity control devices (e.g. a moderator, a reflector);
 - Unbalanced rod positions;
 - Insertion of hot or cold water;
 - Maintenance errors with reactivity devices;
 - Spurious control system signals;
 - Removal of poisons from the coolant or moderator.
- (3) Loss of flow.
- (4) Loss of coolant:
 - Pump down of the pool.
- (5) Erroneous handling or failure of equipment or components:
 - Failure of the emergency core cooling system;
 - Malfunction of the reactor power control;
 - Loss of coolant to fuel in transfer or storage;
 - Exceeding of fuel ratings.
- (6) Special internal events:
 - Fluid jets and pipe whip.

3.25. The categorization of postulated initiating events should be developed on the basis of initiating frequency, likelihood of system recovery and potential consequences of a postulated initiating event, to determine the following:

- (a) Postulated initiating events that are likely to occur during the lifetime of a research reactor but that do not lead to accident conditions (i.e. they could lead to anticipated operational occurrences) should be analysed to show that the research reactor has a sufficient safety margin to comply with the acceptance criteria for such events. Such a safety margin may be present owing to the provision of specific safety systems and engineered safety features in the design and the establishment of operating procedures to (i) restore the safe state and (ii) prevent or minimize damage.
- (b) Postulated initiating events of low likelihood that reflect the specific characteristics of the design and could lead to an accident (design basis accidents) should be analysed to show that the research reactor has an adequate safety margin to comply with the acceptance criteria for such events. Such a safety margin may be present owing to the provision of safety systems or engineered safety features or because of an inherent behaviour of the reactor tending to (i) restore the safe state, (ii) prevent the release of radioactive material and (iii) limit any release to an acceptably low level.

3.26. Postulated initiating events should be grouped in accordance with the expected frequencies of the initiating events and should be clearly assigned to different plant states. The purpose of this categorization is as follows:

- To justify the basis for the range of events under consideration;
- To reduce the number of initiating events necessitating detailed analysis to a set that includes the enveloping cases in each of the various event groups credited in the safety analysis but that does not contain different events that are associated with identical system performance (such as events that are identical in terms of timing, response of plant systems and radiological release fractions);
- To allow for different acceptance criteria for the safety analysis to be applied to different event groups.

3.27. Both internal and external postulated initiating events of all types, for all operational states, including shutdown and fuel loading, should be considered in this process of event grouping. The process of event grouping should lead to a list of enveloping postulated initiating events to be analysed. Failures in other systems such as experimental facilities, failures in the availability of off-site power or the