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

IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

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

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

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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

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

SPECIFIC SAFETY GUIDE

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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.

NOTE BY THE SECRETARIAT

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. The process of developing, reviewing and establishing the IAEA standards involves the IAEA Secretariat and all Member States, many of which are represented on the four IAEA safety standards committees and the IAEA Commission on Safety Standards.

The IAEA standards, as a key element of the global safety regime, are kept under regular review by the Secretariat, the safety standards committees and the Commission on Safety Standards. The Secretariat gathers information on experience in the application of the IAEA standards and information gained from the follow-up of events for the purpose of ensuring that the standards continue to meet users' needs. The present publication reflects feedback and experience accumulated until 2010 and it has been subject to the rigorous review process for standards.

Lessons that may be learned from studying the accident at the Fukushima Daiichi nuclear power plant in Japan following the disastrous earthquake and tsunami of 11 March 2011 will be reflected in this IAEA safety standard as revised and issued 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 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.

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

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

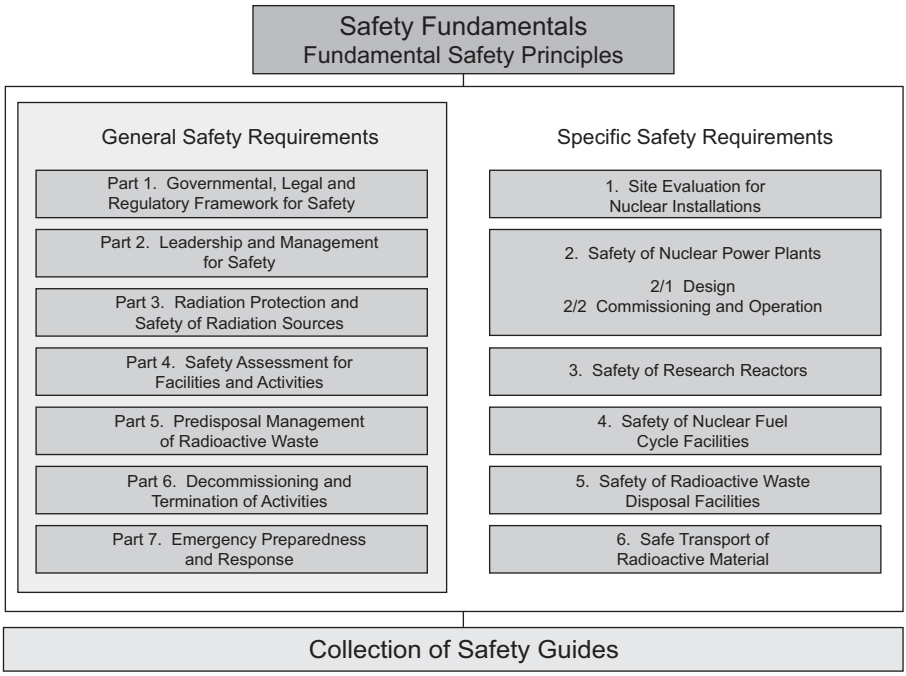

FIG. 1. The long term structure of the IAEA Safety Standards Series.

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 four safety standards committees, for 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. It articulates the mandate of the IAEA, the vision for the future application of the

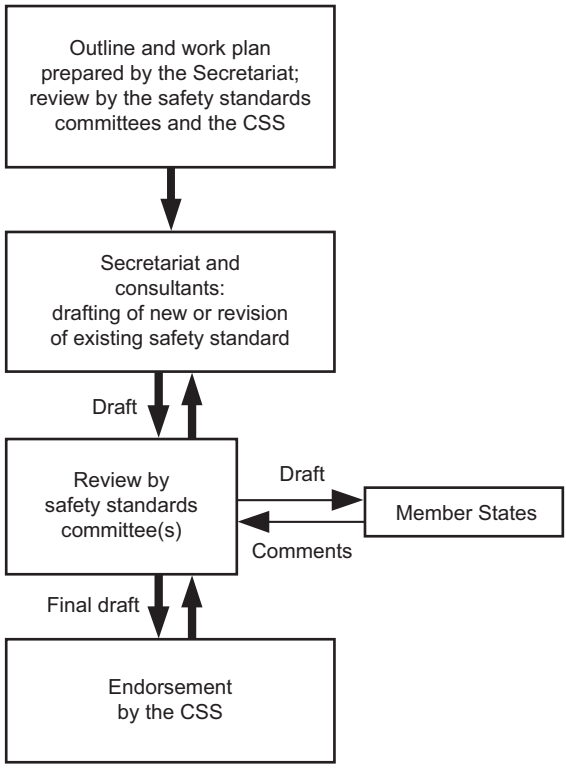

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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 Safety Guide was developed under the IAEA programme for safety standards, which covers all the important areas of research reactor safety. The Fundamental Safety Principles publication [1] establishes principles for ensuring the protection of workers, the public and the environment from harmful effects of ionizing radiation. Requirements that apply seven of these principles[1] are directly addressed in this Safety Guide. Recommendations are provided on which analyses, acceptance criteria, verifications and evaluations should be performed and used to prove that the safety objectives will be met to fulfil the safety requirements for the operating organization that are established in paras 2.15, 2.17–2.20, 3.6–3.12 and 4.14 of Ref. [2].

1.2. This publication supersedes IAEA Safety Series No. 35-G1[2]. The main changes and adaptations relate to consistency with Ref. [2], the other recently published Safety Guides for research reactors and other relevant safety standards. Where applicable, references to the safety standards have been incorporated.

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[1] These are Principles 1, 2, 3, 5, 6, 8 and 9 (see Ref. [1]):
- — "Principle 1: Responsibility for safety: The prime responsibility for safety must rest with the person or organization 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 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 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."

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, IAEA Safety Series No. 35-G1, IAEA, Vienna (1994).

1.3. Owing to the particular characteristics of research reactors, safety aspects relating to all stages in the lifetime of a research reactor have been given special emphasis and have been incorporated into Ref. [2]. These characteristics include: the large variety of designs; the wide range of maximum power levels; the different modes of operation and different purposes of utilization; the particularities of siting; and the differences between organizations operating research reactors, in particular concerning their available resources. These characteristics necessitate a graded approach (Ref. [2], paras 1.11–1.14; Ref. [3]) in the setting and the fulfilment of the requirements when dealing with certain specific topics. These circumstances have been taken into account in the present Safety Guide.

1.4. The organizations involved in ensuring the safety of research reactors and the protection of site personnel, the public and the environment have a number of responsibilities that are interrelated. Most important are the performance of the safety analysis by the operating organization and the review and assessment of the safety analysis report by the regulatory body. The preparation, submission and evaluation of other important safety related documents is also necessary during the licensing process or in other special circumstances, such as for a modification or utilization project. The present Safety Guide develops the general concepts in these areas and is to be used in conjunction with Ref. [2], in which the concepts are presented. Furthermore, this publication covers other aspects of reactor operation normally included in the safety analysis report, such as operational limits and conditions[3], commissioning, operating procedures, and utilization and modification, which are also discussed in other publications.

1.5. The use of the terms ‘safety assessment’, ‘safety analysis’ and ‘review and assessment’ in this Safety Guide requires explanation. ‘Safety assessment’ is assessment of all aspects of a practice that are relevant to protection and safety; for an authorized facility, this includes siting, design and operation of the facility. The safety assessment of the reactor may cover many and varied activities during the licensing process, with iterations between the design and confirmatory analytical activities, such as analyses and the preparation and submission of documents for review, and may involve several organizations. The safety assessment should continue throughout all the stages of the reactor’s lifetime and

[3] 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.

in accordance with the potential magnitude and nature of the hazard associated with the particular research reactor or activity. In general, in the safety standards for research reactors the term 'safety assessment' is used instead of the term 'safety analysis', which has a more specific meaning. Safety analysis is the evaluation of the potential hazards associated with the conduct of an activity. In this regard, the safety analysis is performed by the designer and/or by the operating organization. The safety analysis is one of the most important parts of the reactor licensing process, and it is required to be included in the safety analysis report and submitted to the regulatory body for review and assessment [2]. The term 'review and assessment' is used in this Safety Guide specifically in the frame of the licence application with regard to the information to be submitted to the regulatory body, which has responsibility for assessing safety documentation (see Requirements 25 and 26 of Ref. [4]).

1.6. This Safety Guide provides guidance to operating organizations on carrying out independent verification of the safety assessment for a new research reactor with a new design or an existing design, or of a revised and updated safety assessment for an existing research reactor. Although this Safety Guide provides guidance to operating organizations, it is also suitable for designers performing a safety assessment for a new or an existing research reactor.

OBJECTIVE

1.7. The objective of this Safety Guide is to provide recommendations on meeting the requirements for safety assessment for research reactors in the licensing process, such as the responsibilities and functions of the organizations involved in the licensing process (Ref. [2], paras 3.2 and 3.3, and 4.1–4.4) and the steps towards the issuing of the licence (Ref. [2], paras 3.4 and 3.5), as well as on meeting the requirements for conducting the safety assessment of facilities and activities [5]. This Safety Guide also provides recommendations on the performance of the safety analysis (Ref. [2], paras 6.72–6.78) and the preparation of the safety analysis report (Ref. [2], paras 3.6–3.10). Guidance is also given on which analyses, verifications and evaluations should be performed to demonstrate that the safety objectives will be met and to fulfil the safety requirements on the operating organization. Finally, recommendations are provided on what information has to be submitted for the review and assessment of the safety analysis report by the regulatory body.

1.8. 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 new research reactors of a new or existing design. The guidelines are also applicable for a revised and updated safety assessment of an existing reactor.

1.9. This Safety Guide provides recommendations with respect to utilization[4] (for experiments and experimental facilities) only in direct relation to safety analyses for the safety analysis report for the reactor. Recommendations on safety analyses for experiments at research reactors and experimental facilities are provided in Ref. [6].

SCOPE

1.10. The recommendations provided in this Safety Guide are applicable to any type of research reactor[5]. However, the particular characteristics of research reactors, as set out in para. 1.3, necessitate that a graded approach (see Ref. [2], paras 1.11–1.14; Ref. [3]) be adopted in implementing the recommendations in this Safety Guide. It is therefore necessary that users of this Safety Guide make a conscious and justified selection of the recommendations provided. In addition, the extent of detail in the application of the requirements will depend on the potential hazard and on discussions and agreements between the operating organization and the regulatory body, with the final decision to be made by the regulatory body.

1.11. This Safety Guide focuses mainly on research reactors of a capacity of up to a few tens of megawatts. The amount of detail required in the safety analysis report for small research reactors (i.e. those with a capacity of less than a few tens of kilowatts) and critical assemblies may be substantially less. Nevertheless, when using the graded approach, all items included in this Safety Guide should be assessed. Additional recommendations on the safety analysis, on preparation of the safety analysis report and on the licensing process for high powered or otherwise advanced or complex research reactors are provided in IAEA Safety

[4] Utilization (or reactor utilization) is the use of the research reactor, including for isotope production, or the conduct of experiments or the use of experimental facilities during operation of the reactor.

[5] In this Safety Guide, the term ‘research reactor’ includes associated experimental facilities and subcritical and critical assemblies. An experimental facility includes 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.

Guides for power reactors.[6] Use of the Safety Guides for power reactors also necessitates that a graded approach (see Ref. [2], paras 1.11–1.14; Ref. [3]) be applied in implementing the recommendations on the basis of the potential hazard associated with the research reactor.

1.12. Although this Safety Guide mainly concerns newly designed and constructed research reactors, its content is applicable to any relicensing process or reassessment of a research reactor requested by the regulatory body or decided on by the operating organization. In any case, the justification for the approach selected on the basis of this Safety Guide should be provided to the regulatory body. Licensing of decommissioning activities[7] is not discussed in this Safety Guide.

1.13. Most research reactors have a small potential for hazard to the public compared with power reactors, but they may pose a greater potential hazard to operating personnel. The scope, extent and detail of the safety assessment should be based on the potential hazard associated with the research reactor and its utilization. The requirement for a graded approach should be adopted (see Ref. [2], paras 1.11–1.14; Ref. [3]) in applying the recommendations and guidance in this Safety Guide.

1.14. The interfaces between nuclear safety and nuclear security should be considered in such a way that the impacts of safety on security and the impacts of security on safety are taken into account from the design stage and an appropriate compatibility is achieved. However, security aspects are subject to confidentiality requirements, and they are not discussed in this Safety Guide (see paras A.13.12 and A.13.13).

STRUCTURE

1.15. This Safety Guide addresses two interrelated issues: the safety assessment of the research reactor and the preparation of the safety analysis report. It also provides general recommendations on the conduct of the steps in the licensing of a research reactor. The main reason for presenting these two topics together in a

[6] Further recommendations on preparation of the safety analysis report for a research reactor with greater potential hazard are provided in Refs [7, 8].

[7] Recommendations on decommissioning activities are provided in Ref. [9].

single Safety Guide is their interrelationship and their joint importance in the licensing process.

1.16. Section 2 describes the licensing process by which the safety of the research reactor and the issuing of licences are controlled and determined.

1.17. Section 3 presents general recommendations on the preparation of the safety analysis report, in particular the preparation of the safety analysis by the operating organization.

1.18. 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.

1.19. The Appendix provides comprehensive guidance on the preparation of the safety analysis report for a research reactor having the characteristics discussed in paras 1.3 and 1.10–1.13. It provides recommendations on the standard content of the safety analysis report.

1.20. Annexes I and II outline, and provide information on, the application of a basic approach to performing the safety analysis for a research reactor using mainly deterministic methods[8] to analyse accidents, including their radiological consequences. 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.

[8] Annex I deals mainly with deterministic methods which are normally used for safety evaluations for research reactors. Probabilistic techniques could be used to supplement deterministic methods; see paras 3.28 and 3.29.

2. SAFETY ASSESSMENT IN THE LICENSING PROCESS

RESPONSIBILITIES

2.1. In accordance with the Fundamental Safety Principles (Ref. [1], para. 2.1), it should be ensured that for a research reactor to be built (or to undergo a major modification), the highest safety standards that can reasonably be achieved should be met, to protect people and the environment in the vicinity of the site where it is operated. This assurance is provided by the governmental, legal and regulatory framework, which ensures that an adequate legal and regulatory basis for assessing the safety implications of the project is available (Ref. [4], Requirements 1 and 2). The establishment of an independent regulatory body is an important requirement for an adequate legal and regulatory framework. The IAEA Safety Requirements publication on the Safety of Research Reactors [2] establishes general requirements for the framework of the system for ensuring safety, including the licensing process. Further requirements on the establishment of a regulatory body are established in Ref. [4]. These requirements also apply, as appropriate, for research reactors.

2.2. 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 normally incorporated into a safety analysis report. This information also constitutes the prime basis for the regulatory body's decision on licensing 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 operation of the research reactor.

2.3. The content of the application for a licence should be based on the legal and regulatory framework of the State. Relevant requirements for the licensing process are established in Ref. [4]. 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 licensing process.

2.4. Licensing 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 licensing process should be understandable by interested parties and should be predictable (i.e. well defined, clear, transparent and traceable). The different stages of the licensing 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 in a logical order (Ref. [10], para. 2.5).

2.5. In all cases, the major stages of the licensing process for research reactors should encompass the regulation of:

(1) Siting and site evaluation;
(2) Design and construction;
(3) Commissioning;
(4) Operation, including utilization and modification[9];
(5) Decommissioning and release from regulatory control.

2.6. In some licensing regimes, consideration has been given to the adaptation of a 'prelicensing' process, such as steps that provide for early approval of siting, approval of the safety concept and design, and issuing of a construction licence. Such a licensing regime may help to minimize the duplication of effort through different stages of the licensing process. It may also allow for some stages to be conducted in parallel. It provides for the clear division of responsibilities for different stages between regulatory bodies, vendors and operating organizations; gives the public opportunities for early participation; and ensures that the most important safety issues are dealt with early in such a 'prelicensing' phase. A detailed demonstration of nuclear safety, including an adequate safety analysis, should be submitted by the operating organization, and should be reviewed and assessed by the regulatory body before the next stage is authorized. Detailed guidance on the licensing process is presented in Ref. [10].

2.7. 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

[9] Although utilization and modification of research reactors are processes comprising activities normally included in operation (Ref. [2], paras 7.85–7.92), they may be considered separate stages in the licensing process, because their safety implications lead to a large number of review and assessment activities that are repeated many times over the reactor's lifetime (see paras 2.40–2.43).

structure, a management system[10], and adequate resources to discharge its obligations and, as appropriate, its liabilities. The totality of the documentation that the operating organization uses in making this demonstration, some of which may not be included in the initial formal submission, should cover all appropriate topics, depending on the stage of the licensing process.

2.8. On the basis of the requirements for the governmental, legal and regulatory framework for safety with regard to the review and assessment of safety related documentation [4], the operating organization should submit to the regulatory body, in a timely manner, any information that the regulatory body has requested. It is the responsibility of the operating organization to make arrangements with the vendors to ensure the availability of information that has been requested by the regulatory body. It should also be the responsibility of the operating organization to keep the regulatory body informed of relevant new information and of any alterations to information submitted previously.

2.9. The format and content of documents submitted by the operating organization in support of a licence application should be based on the information presented in this Safety Guide. However, the regulatory body may require or may use additional information in the licensing process.

2.10. The review and assessment of information by the regulatory body is a continuous process. Sections of the safety analysis report or other documents should be submitted to the regulatory body at an early stage. A schedule for the review and assessment by the regulatory body should be discussed between the operating organization and the regulatory body.

2.11. The operating organization should revise all documentation associated with any modification or activity that may affect the safety of a research reactor (and all documentation having an indirect but significant influence on safety related aspects of a research reactor), as appropriate. The revised documentation should be submitted to the regulatory body to allow for its review and assessment (Ref. [4], para. 4.45), with the potential magnitude and nature of the associated hazards being taken into account (Ref. [4], Requirement 26).

[10] The term ‘management system’ reflects and includes the concepts of quality control, quality assurance and quality management. The management system is a set of interrelated or interacting elements that establishes policies and objectives, and enables those objectives to be achieved in a safe, efficient and effective manner. Further requirements and guidance are provided in Refs [11, 12].

2.12. 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 and decommissioned without undue radiation risks to site personnel, the public or the environment. On the basis of the documentation submitted, the regulatory body should be able to do the following:

(a) To acquire an understanding of the reactor design, the safety concepts on which it is based, the management system and the operating principles proposed by the operating organization.
(b) To 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 behaviour in normal operation, anticipated operational occurrences and accident conditions.

When necessary, modifications to the matters stated in (a) and (b) also have to be submitted at the request of the regulatory body.

2.13. The primary basis of 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 be sufficient for the regulatory body to decide on the following points:

— Whether the operating organization has provided the necessary and adequate information for the purpose and scope of the review and assessment (see para. 4.2).
— Whether this information is in compliance with the requirements of all applicable rules and regulations.
— Whether this information is accurate; this might be determined by means of independent checks of the design, including calculations, and by inspections of the programmes and facilities (e.g. design and review programmes or management system requirements and their implementation).

ACCEPTANCE CRITERIA[11]

2.14. In addition to the acceptance criteria established within the regulatory framework, the operating organization should develop additional acceptance criteria to demonstrate the adequate application of the principles and objectives of safe design and operation established in the IAEA safety standards. These principles include the Fundamental Safety Principles [1] and the radiation protection objectives stated in para. 2.2 of Ref. [2].[12]

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

— A set of numerical limits on the values of predicted parameters;
— A set of conditions for plant states during and after an accident;
— A set of performance requirements on systems;
— A set of requirements on the need for, and the ability to credit, actions by the operating organization.

2.16. The acceptance criteria should include additional margins beyond the basic acceptance criteria as established within the regulatory framework, to allow for uncertainties. These specific acceptance criteria may be defined by the designer or by the operating organization. The set of acceptance criteria should be satisfactory to the regulatory body.

2.17. In the development of the acceptance criteria, consideration should be given to the criteria listed below:

(a) Radiological criteria such as:
— Maximum allowable doses to the public;
— Dose limits (or design target doses[13]) for staff of the operating organization, including experimenters and workers at the reactor site;
— Dose limits for intervention in accident conditions to perform life saving actions and mitigatory actions;

[11] Practical examples of acceptance criteria are provided in Ref. [13].

[12] Requirements on radiation protection are established in Ref. [14], and additional recommendations and guidance are given in Ref. [15].

[13] Guidance on design target doses is provided in Ref. [15], paras 2.8 and 2.9.

— Authorized limits on releases to the environment during normal operation and acceptable limits on releases to the environment in accident conditions;
— Risk criteria (where applicable).

(b) Nuclear fuel performance criteria:
— 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;
— Frequency limits for significant damage to fuel cladding.

(c) Performance criteria, including:
— Limits on parameters to prevent damage of the primary coolant system boundary;
— Limits on parameters to prevent damage to safety relevant systems caused by in-core or out of core experimental facilities[14];
— Limits on parameters to prevent damage to the containment systems;
— Maintenance of core cooling;
— Frequency limits for certain anticipated operational occurrences and for particular accident conditions.

2.18. The detailed acceptance criteria should include the following:

— 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 a beyond design basis accident.
— There should be no consequential loss of function of the safety systems necessary to mitigate the consequences of an accident.
— Systems used for mitigation of the consequences of accidents should be designed and constructed to withstand the maximum loads and stresses and the most extreme environmental conditions for the accident analysed.

[14] An isotope production facility is regarded as an experimental facility.

INFORMATION REQUIREMENTS IN THE VARIOUS STAGES OF THE LICENSING PROCESS

2.19. 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. Guidance on the preparation and presentation of the safety analysis report is provided in Section 3, and guidance on its review and assessment is provided in Section 4. The following paragraphs provide a summary of the information that is normally required for each stage of the licensing process. The sequential request for information may lead to successive updating, with each version of the safety analysis report corresponding to a particular stage of the licensing process, as outlined in para. 2.5.

2.20. 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, as well as 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.

2.21. At various stages in the course of the design process (for example, before the start of construction or 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.

SUBMISSION OF INFORMATION TO THE REGULATORY BODY

Schedule for the submission of information

2.22. A schedule should be drawn up that indicates the timescale for the preparation of the different chapters of the safety analysis report. Since the approval of one stage is normally required before commencement of the next stage, the safety analysis report should be made available for review and assessment on a timescale that has been agreed upon by the regulatory body. An estimate of the size and scope of the analyses should be conveyed to the assessor. In this timescale, reasonable periods of time should be allotted 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

2.23. The operating organization should provide sufficient information to demonstrate to the regulatory body that the proposed site is suitable for the type and design of the proposed research reactor. Difficulties to be resolved during the subsequent stages of the licensing process should be identified. Information on the site itself, and preliminary information on the 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. If required in the State, an environmental impact study should be performed as a part of the licensing process.

2.24. The characteristics of the site, which may affect safety related aspects of the research reactor, should be investigated and assessed by the operating organization. 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 have to be considered for site evaluation are provided in section 5 of Ref. [2]. Additional guidance on siting and site evaluation is provided in the Appendix (see Chapter 3: Site characteristics) and in Ref. [10] and requirements on site evaluation are established in Ref. [16]. The details on siting that have to be addressed in the safety analysis report are presented under Chapter 3 of the Appendix.

Design and construction

2.25. Before authorization of the construction of the research reactor, features such as the physical layout and the type of construction of the reactor as well as the key elements of the 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 ageing mechanisms for materials and structures, systems and components, and to the effects of these ageing mechanisms on safety.[15] The operating organization should describe the

[15] Additional guidance on ageing management for research reactors is given in Ref. [17].

arrangements for the control of activities in construction, manufacture and installation. In addition, an outline plan for decommissioning, covering issues such as strategies to be applied, radiation doses to be expected and amounts of waste expected to be generated, should be prepared at the design stage. Information on the matters discussed in this paragraph should be submitted to the regulatory body for review and assessment.

2.26. 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 reactor and the associated safety systems and process systems. It should also present the results of the safety analysis to demonstrate the adequacy of the design of safety related structures, systems and components. This information should be submitted in the safety analysis report, which may be preliminary and subject to updating as the project proceeds.

2.27. 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 so that activities can proceed while the reactor is under construction. The information should be updated and resubmitted to the regulatory body as the detailed design and the construction of the reactor proceed. In some cases, revised versions of documents will be sufficient; in other cases, technical supplements may be appropriate. Additional guidance on the licensing process for this stage is given in Ref. [10].

2.28. 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 construction.

Commissioning

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

2.30. Reference [2] (para 7.46) requires that commissioning tests be arranged in functional groups and in a logical sequence. This sequence includes pre-operational tests, initial criticality tests, low power tests, tests to verify the

shutdown capabilities, power ascension tests and full 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 approved at the appropriate level of management before the subsequent test sequence is started. The commissioning programme should therefore be divided into stages, which are usually arranged according to the following sequence:

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

Commissioning should be carried out in accordance with the commissioning programme that has been reviewed and assessed by the regulatory body. Detailed guidance on the commissioning of research reactors is given in Refs [10, 18].

2.31. The updated safety analysis report should include the commissioning programme and should demonstrate its adequacy (Ref. [2], paras 7.42–7.44). 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.

2.32. The commissioning procedures for a commissioning stage should be reviewed before the start of the next stage and should be updated, where 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.

2.33. Stage A (tests prior to fuel loading) should ensure that the reactor has been constructed, manufactured and installed correctly and in accordance with the design documentation. If deviations from the design documentation have occurred, they should be recorded, and it should be shown that the safety analysis has not been compromised. The results of this stage should also confirm the operational features of the research reactor and should lead to the development of detailed instructions for operating personnel, which should be confirmed during Stages B and C.

2.34. Stage B (fuel loading tests, initial criticality tests, low power tests and tests to prove the shutdown capabilities with the introduction of fissile material) is a major step in the authorization process. The commissioning programme of the

previous stage, the organizational structure, the qualifications of operating personnel, the radiation protection programme, emergency preparedness and response, the operational limits and conditions for commissioning, and the preliminary operating procedures[16] should be taken into account at this commissioning stage. 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.

2.35. As power ascension test and full power test processes in Stage C move 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 routine operation is requested, the test results, any corrections of non-conformances, modifications to the design or modifications to the operational procedures, and any proposed changes to the operational limits and conditions should be submitted to the regulatory body for review and assessment.

2.36. The information referred to in paras 2.30–2.35 should be updated after each stage, and submission to the regulatory body should form the basis of the start of the next commissioning stage as a part of the licensing process.

Operation

2.37. In its application for an operating licence, the operating organization should submit all of the information referred to in the preceding sections. Additional information to prove the capability for safe operation should be submitted to the regulatory body. Some of this information is required in the licensing steps, and some information should be submitted after the formal licence has been obtained. Additional guidance on the licensing steps is given in Ref. [10], and detailed guidance on ensuring safe operation is given in Ref. [2] and the related Safety Guides [15, 19, 20].

2.38. The final version of the safety analysis report should be prepared for the stage of application for an operating licence. The results from the commissioning programme should be included and assessed to demonstrate that the design intentions have been achieved.

[16] Guidance on operating procedures is provided in Ref. [19].

2.39. A review of the safety measures for the operation of the research reactor should be undertaken periodically. While the need for reassessment may arise in a number of ways, systematic safety reassessments (i.e. periodic safety reviews) should be carried out by the operating organization at intervals to review important issues such as the cumulative effects of ageing of the research reactor. The nature of this review and the interval between reviews should reflect the risks that the research reactor presents. For this review, a comparison of the existing safety analysis report with informatin on operating experience should be made, including lessons from accidents and information on radiological aspects, modifications, experiments and other aspects of operation. If required as a result of the review of the safety measures for operation, the operating organization should submit to the regulatory body a request for an amendment of the licence. This request may include a revised safety analysis report.

Utilization and modification

2.40. 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 submission requirements will depend on the safety significance of the experiments and modifications. These requirements are set out in paras 7.86 and 7.88 of Ref. [2]. Specific guidance on the development of appropriate procedures for the control of experiments and modifications is provided in Refs [6, 19].

2.41. Experiments and modifications having major safety significance should be subjected to procedures for design, construction, commissioning and safety analysis that are equivalent to those for the research reactor itself. This safety analysis may need to be performed in stages. These stages could be: (1) design and procurement; (2) disassembly; (3) installation or implementation of the modification; (4) reassembly; (5) testing; (6) commissioning; and (7) validation of the design. The safety aspects of each phase of the project should be analysed and presented in a dedicated safety analysis report, or a revision of the appropriate chapters of the existing safety analysis report for the reactor should be prepared. The dedicated safety analysis report or the revised chapters should be 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.

2.42. 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 use in the safety analysis of the proposed experiment or modification.

2.43. Commissioning of the experiment or the modified research reactor should be conducted to demonstrate compliance with the design intention 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 the other safety analyses are still valid.

Decommissioning and release from regulatory control

2.44. The decommissioning process, such that regulatory control may be removed, which includes decontamination and the dismantling and/or removal of radioactive material, radioactive waste, components and structures, should require approval by the regulatory body. Detailed requirements on the subject are established in Ref. [2], paras 8.1–8.8. The operating organization should provide documentation that describes the intended decommissioning process[17] to demonstrate that remaining radiological hazards, if any, at the former site will be minimal, that any radioactive waste generated will be properly dealt with, and that any particular hazards associated with the decommissioning process have been adequately analysed and assessed. Further guidance is provided in IAEA safety standards on the management of radioactive waste.

2.45. 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 detailed report on the decommissioning process should be prepared. Further guidance on decommissioning is provided in Ref. [9].

[17] This documentation describing the intended decommissioning process is required when initiating this decommissioning process.

3. PREPARATION OF THE SAFETY ANALYSIS REPORT

PURPOSE AND SCOPE

3.1. The operating organization should make arrangements for preparing a safety analysis report to demonstrate the safety of the design of the research reactor. The safety analysis report should also 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 licensing 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, since the design of the 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 appreciation of the relevant design criteria, limitations and requirements, and in the evaluation of the hazards posed by the research reactor;
— To aid in the training 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 identify ageing mechanisms and their effects on safety for the development of an ageing management programme.

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

— The evolution of the design, operation and utilization of the research reactor and the related experimental facilities, and any modifications to and upgrades of the research reactor;
— The consequences of events that may have occurred during the lifetime of the research reactor and that may influence the actions that will need to be taken during the eventual decommissioning of the research reactor.

3.4. The safety analysis report should give a detailed description of the reactor site, the reactor itself, the experimental facilities and all other facilities with significance for safety. It should provide a detailed description of the general safety concepts and criteria, as well as of the codes and standards applied to the design for the purposes of protection of the reactor, the operating personnel, the public and the environment. The potential hazards associated with the operation of the reactor should also be addressed in the safety analysis report. The safety analysis report should contain or should refer to the safety analysis of accident sequences and of the safety features incorporated into the design to prevent accidents or to mitigate their consequences through the design and operating procedures.

3.5. The safety analysis report should provide a set of operational limits and conditions to be incorporated into the licence for operation, or should describe the content of the operational limits and conditions if they will be described in a separate document. It should also provide details of the conduct of operations intended by the operating organization, including its organization and the management system procedures established for the design and operation of the research reactor. The safety analysis report should also provide details of the emergency plan.

3.6. While the topics listed in paras 3.4 and 3.5 have been deliberately stressed, all topics treated in the Appendix to this Safety Guide should be adequately covered in the safety analysis report. All of these topics should be prepared in accordance with the corresponding recommendations in the Appendix. However, some of the topics may be discussed in separate documents (e.g. in the operational limits and conditions, operational procedures, physical protection plans or emergency plans). In this case, these topics are treated briefly in the safety analysis report and reference is made to the appropriate separate document.

SPECIFIC GUIDANCE

3.7. The operating organization should ensure that an independent verification of the safety assessment is performed by individuals or groups separate from those carrying out the design, before the design is submitted to the regulatory body (Ref. [2], para. 2.19).

3.8. The independent verification should be carried out under the responsibility of the operating organization by a team of experts who should be independent of the designers and of those performing the safety assessment. Personnel are

considered independent if they have not participated in any part of the design or the safety assessment. This independent verification is in addition to the reviews carried out within the design organization.

3.9. Whereas the safety assessment is a comprehensive study carried out by the designers 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.

3.10. In some States the proposal and the licence application for a research reactor may be subject to an open public debate.[18] 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.

3.11. The safety analysis report should present adequate references that may be necessary for the review and assessment process. This reference material should be freely available to the regulatory body and should 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 upon request.

3.12. 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.

3.13. 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 its staff, technical support organizations, consultants and advisory committees as well as 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

[18] Guidance on public participation is given in paras 2.42–2.45 of Ref. [10].

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.

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

3.15. The type of reactor, its site and its characteristics (design, power and utilization) may influence the extent of the information to be presented in the safety analysis report. Accident scenarios for reactors with higher power levels or with a significant inventory of radioactive material will usually 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.

3.16. For small, low risk facilities (such as critical assemblies or reactors with low power levels), these requirements are much less stringent. However, as the safety analysis report is often the only comprehensive document produced, every topic discussed in the Appendix to this Safety Guide 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 critical assembly facilities) may be much larger for small, low power facilities.

SELECTED POSTULATED INITIATING EVENTS

3.17. The following list of selected postulated initiating events is based on the appendix to Ref. [2]:

(1) Loss of electrical power supplies:
— Loss of normal electrical power.[19]

[19] Although it is not considered an initiating event, consideration should 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.

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

(3) Loss of flow:
- Failure of primary pump;
- Reduction of flow of primary coolant (e.g. due to the failure of a valve or a blockage in piping or a heat exchanger);
- Influence of the failure or mishandling of an experiment;
- Rupture of the primary coolant boundary leading to a loss of flow;
- Fuel channel blockage;
- Improper power distribution due to, for example, unbalanced rod positions, in-core experiments or fuel loading;
- Reduction of coolant flow due to bypassing of the core;
- Deviation of system pressure from specified limits;
- Loss of heat sink (e.g. due to the failure of a valve or a pump, or damage to a system).

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

(5) Erroneous handling or malfunctioning 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 the fuel);

— Failure of an emergency cooling system;
— Malfunctioning of the reactor power control;
— Criticality in fuel in storage;
— Failure of means of confinement, including the ventilation system;
— Loss of coolant during transfer or storage of fuel;
— Loss or reduction of proper shielding;
— Failure of experimental facilities or materials (e.g. due to loop rupture);
— Exceeding of fuel ratings.

(6) Special internal events:
— Internal fires or explosions, including internally generated missiles;
— Internal flooding;
— Loss of support systems;
— Dropping of heavy loads;
— Loss of integrity of pressurized vessels;
— Malfunction during a reactor experiment;
— Improper access by persons to restricted areas;
— Fluid jets and pipe whip;
— Exothermic chemical reactions;
— Electromagnetic compatibility;
— Security related incidents (see paras A.13.12 and A.13.13 in the Appendix).

(7) External events[20]:
— Earthquakes (including seismically induced faulting and landslides);
— Flooding (including failure of an upstream dam and blockage of a river);
— Tornadoes and tornado missiles;
— Sandstorms;
— Hurricanes, storms and lightning strikes;
— Tropical cyclones;
— Explosions;
— Aircraft crashes;
— Fires;
— Toxic spills;
— Accidents on transport routes;
— Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities or waste management facilities);
— Biological hazards such as microbiological fouling, structural damage or damage caused to equipment by rodents or insects;

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

— Extreme meteorological phenomena;
— Power surges or voltage surges on the external supply line;
— Security related external events (see paras A.13.12 and A.13.13 in the Appendix).

(8) Human errors.

DEVELOPMENT OF THE SAFETY ANALYSIS

3.18. The safety analysis, as part of the safety assessment used in the licensing 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 reactor as constructed.

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 operational 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 a range of postulated initiating events (such as disturbances in process parameters, malfunctions and failures of equipment, internal and external events, postulated design basis accidents and human errors). The safety analysis should also serve as a basis for the determination of the operational limits and conditions, as well as for designing specifications for systems and components.

3.20. The consideration of fault 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 consideration of fault conditions should also inform the operating instructions and procedures for operating personnel. In addition, the potential radiological consequences of fault conditions for workers, the public and the environment may be more severe than the radiological consequences in routine operation. For this reason, an important part of the effort in the peer review and verification by the operating organization should be directed to the safety analysis of fault conditions. This analysis should be performed in accordance with the magnitude and nature of the risks 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 development of the postulated initiating events in relation to system responses and their consequences;
— Comparison against acceptance criteria.

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

— That sufficient defence in depth has been provided and that the levels of defence are preserved in that potential accident sequences are arrested as early as possible.
— That the research reactor can withstand the physical and environmental conditions that it would experience. This would include extreme environmental conditions and other extreme conditions.
— That human factors and human performance issues have been adequately addressed.
— That long term ageing mechanisms that could detract from the reliability of structures, systems and components over the design lifetime are identified, monitored and managed (i.e. by upgrading, refurbishment or replacement), so that safety is not affected and risks do not increase.

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 auditable, as appropriate. Moreover, as complete as possible a listing of postulated initiating events should be provided. An important feature of the review and assessment process should be to consider whether the method of identification meets these requirements and whether the list of postulated initiating events is acceptable as the basis for the safety analysis. The use of hazard and operability (HAZOP) studies or failure modes and effects analysis (FMEA) could facilitate the selection process.

3.23. Postulated initiating events should be categorized in accordance with their anticipated system response. The purpose of this categorization is:

— To justify the basis for the range of events under consideration;
— To reduce the number of initiating events requiring 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 events that are associated with identical system performance (such as events that are

identical in terms of timing, plant systems response and radiological release fractions);

— To allow for different acceptance criteria for the safety analysis to be applied to different event classes.

3.24. Both internal and external initiating events of all types, for all operational states, including shutdown and fuel loading, should be considered in this process of event classification. The process of event classification should lead to a list of enveloping initiating events to be analysed. Failures in other systems such as experimental facilities, failures in the availability of off-site power or the total loss of off-site power, and failures in spent fuel storage and in storage tanks for radioactive liquids should also be considered.

3.25. In the preparation of the set of postulated initiating events for the analysis, the list given in para. 3.17, which is based on the appendix to Ref. [2], should form the basis of the postulated initiating events to be considered. Considerations on the methodology to be used are given in Annex I to this Safety Guide. Annex I also lists considerations for analyses of the event sequences triggered by the postulated initiating events and for analyses of external events and internal events. In particular, the analyses should clearly identify a number of assumed input parameters and initial conditions. These assumed input parameters and initial conditions should be presented in the safety analysis report and will provide the basis for the determination of the operational limits and conditions. Annex II to this Safety Guide gives examples of these parameters.

3.26. The general requirements in the development of the safety analysis are presented in Ref. [2], paras 6.72–6.78. To ensure that the safety analysis meets the intended objective, the detailed guidance on the preparation of the safety analysis as presented in the Appendix to this Safety Guide (Chapter 16: Safety analysis) should be taken into account.

3.27. The safety analysis should identify design basis accidents. In addition, accidents beyond the design basis that have more severe consequences may be analysed for purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident.

3.28. Annex I deals mainly with deterministic methods, which are normally used for safety assessments of research reactors. Deterministic techniques are characterized by conservatism and are based on defined sets of rules for event selection, analytical methods, and parameter specification and acceptance criteria. Through the use of these methods, reasonable assurance is provided that

the ultimate objective of preventing or limiting the release of radioactive material can be achieved without the need to perform complex calculations, because these methods tend to overestimate the amount of radioactive releases. The most severe of these releases (arising from the design basis accident or from a 'maximum credible accident') are taken into account in the selection of a site or in setting design requirements for engineered safety features for the reactor. The choice of these accidents is based on experience and engineering judgement, without the benefit of determining the probabilities of the event sequences.

3.29. Probabilistic techniques could be used to supplement the above mentioned safety assessments. Probabilistic methodologies use the assumption that all accidents are possible and that any number of simultaneous failures may occur, although the probabilities may be very low. Some postulated accidents or combinations of accidents may have less dramatic consequences than the postulated accidents used in the deterministic methodology. However, when they are weighted by their likelihood, they may represent a significant risk and may impose different demands on the design. In addition, the deterministic approach has difficulties in effectively treating system interdependences (e.g. common cause failure), which probabilistic methods can address analytically and quantitatively. Application of probabilistic techniques also leads to significant improvements in the understanding of system behaviour and interactions, and of the role of operating personnel under accident conditions. These techniques may be indicated for some specific cases, which could be discussed between the operating organization and the regulatory body.

3.30. A typical classification of postulated initiating events should be developed on the basis of initiating frequency, likelihood of system recovery and potential consequences of an initiating event, to determine the following:

(a) Postulated initiating events that are of high likelihood, which should be analysed to show that the research reactor has a robust tolerance for such events. Such a tolerance may be due to the provision of safety systems or because of an inherent behaviour tending (i) to restore the safe state, (ii) to prevent the release of radioactive material and (iii) to limit any such release to an acceptably low level.
(b) Postulated initiating events that are of low likelihood but that have potential severe consequences such that the research reactor should have safety systems in place to prevent the release of radioactive material, or to limit any release to an acceptable level.

Postulated initiating events that do not fall into these two groups (i.e. postulated initiating events with a low likelihood and in principle with low consequences) should also be evaluated, to ensure that small deviations from the scenarios for incidents will not cause unacceptable risks (e.g. cliff edge effects[21]) to the reactor or unacceptable risks to workers, the public, property or the environment.

3.31. The results of the safety analysis of the research reactor should be reflected in the safety analysis report by taking into account the guidance provided in the Appendix to this Safety Guide (Chapter 16: Safety analysis). The discussion of Chapter 16 of the safety analysis report also provides guidance on the comparison of the results with the acceptance criteria to determine the acceptability of the research reactor.

4. INFORMATION TO BE SUBMITTED FOR THE REVIEW AND ASSESSMENT PROCESS

PURPOSE AND SCOPE

4.1. The review and assessment process is an important appraisal, performed by the regulatory body, of information submitted by the operating organization to demonstrate the safety of the research reactor. Review and assessment are undertaken to enable the regulatory body to make a decision or a series of decisions on the acceptability of the research reactor in terms of safety. The process consists of examining the submissions of the operating organization on all aspects relating to the safety of the research reactor. It includes consideration of both normal operation and failures, and of events, including human errors, that have the potential to cause exposure of site personnel or the public, or radiological hazards to the environment. This safety analysis should be complete and should cover all the initiating events as agreed with the regulatory body, and one of the initial tasks of the review and assessment is to confirm its completeness. The review and assessment process includes checks on the site and

[21] 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.

elsewhere to validate the claims made in the submissions. Operating organizations often have external peer reviews conducted at their facilities by national bodies or international organizations. The results of such reviews could provide the regulatory body with additional insights into the activities of the operating organization.

4.2. The operating organization should include information in support of its licence application to facilitate the review process by the regulatory body. The regulatory body can then determine whether the proposed research reactor can be sited, constructed, commissioned, operated, utilized and modified, and eventually decommissioned, without undue radiation risks to site personnel, the public or the environment. The information submitted should include detailed information for the review and assessment in order:

(a) To determine whether the site is adequate for the type, power and use of the proposed research reactor;
(b) To determine, before construction, whether the proposed reactor design meets the regulatory body's requirements, and to impose any further requirements or conditions that may be deemed necessary by the regulatory body;
(c) To determine whether the operating organization has the necessary ability, reliability, resources, organizational structure and competent personnel to meet the regulatory requirements;
(d) To determine whether the construction remains consistent with the requirements of the regulatory body;
(e) To determine whether the commissioning programme is adequate and whether its results conform to the design intentions;
(f) To determine whether the operational limits and conditions are consistent with the regulatory requirements and whether an adequate level of operational safety can be ensured, including the provisions made for accident conditions;
(g) To determine whether the utilization and modification of the research reactor meet the requirements of the regulatory body;
(h) To determine whether the decommissioning programme meets the requirements of the regulatory body.

PROGRAMME FOR REVIEW AND ASSESSMENT

4.3. The operating organization should discuss with the regulatory body the programme for review and assessment, which should be established by the

regulatory body. The programme for review and assessment should take into account the stages of the licensing process as described in para. 2.5 and paras 2.22–2.43.

4.4. The programme for review and assessment should establish at an early date a schedule for the submission of documents for review and assessment. This schedule should be appropriate to the stages of the licensing process.

4.5. For more important submissions by the operating organization (such as the safety analysis report), it may be useful for the regulatory body to perform an acceptance review of the documentation. As a result of such an acceptance review, an application or submission that is deficient in certain areas may be returned to the operating organization for correction and resubmission (Ref. [21], para. 3.5).

4.6. A major feature of the submission by the operating organization will be its analysis of normal operational conditions as well as its analyses of the deviations from normal operation. However, the importance of the other aspects of the safety submission should be recognized: the safety of the research reactor should be based on sound engineering and good management, and the safety analysis should be a confirmation of the adequacy of the engineering and management and not a substitute for them. The value of safety analysis is in extending knowledge about and understanding of the research reactor and its behaviour, and in identifying shortcomings and areas in which safety can be improved.

4.7. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for the construction of the research reactor should include:

(a) The competence and capability of the operating organization to meet the licence requirements;
(b) The site characteristics, to confirm the acceptability of the site and the related data used in the design of the proposed research reactor;
(c) The basic design of the proposed research reactor, to confirm that it will meet the safety requirements, including requirements for occupational health and requirements for fire safety;
(d) The management systems of the operating organization and those of its vendors;
(e) The design features relating to physical protection that are important to safety;
(f) Information necessary for verification of the design.

4.8. The documents of the operating organization's case for the safety of the research reactor as presented in the safety analysis report, which should be submitted to the regulatory body for review and assessment in order to obtain authorization for commissioning Stage A (tests prior to fuel loading), should include:

(a) The 'as-built' design of the reactor;
(b) The commissioning programme;
(c) The operational limits and conditions for Stage A commissioning;
(d) The records and reporting systems;
(e) The management system, organizational structure and programme for operation.

4.9. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for commissioning Stage B (loading of fuel and initial criticality) should include:

(a) The records of the results of the previous commissioning stage, including non-conformances and, where appropriate, their associated corrective actions;
(b) The revisions to the commissioning programme, if any;
(c) The operational limits and conditions for Stage B commissioning;
(d) The provisions for radiological protection;
(e) The adequacy of the operating instructions, operating procedures, emergency procedures and administrative rules;
(f) The records and reporting systems;
(g) The training and qualification of research reactor personnel, including the levels of staff and their suitability for the work;
(h) The occupational health and fire safety aspects;
(i) The management system, organization and programme for operation;
(j) The emergency plan;
(k) The system of accounting for and control of nuclear material and radioactive material;
(l) The arrangements for physical protection of the reactor.

4.10. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for commissioning Stage C (power ascension tests and power tests) should include:

(a) The records and results of the commissioning tests of Stage B;
(b) The revisions to the commissioning programme, if any;

(c) The operational limits and conditions for Stage C commissioning;
(d) Any revised arrangements.

4.11. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for routine operation at full power should include:

(a) The records and results of commissioning tests of Stage C;
(b) Verification that the radiation dose rates in the reactor are as expected and verification of the adequacy of the shielding;
(c) The operational limits and conditions for normal operation;
(d) Any revised arrangements;
(e) The arrangements for maintenance, periodic testing, inspection, control of modifications and changes to specifications and surveillance.

4.12. Before starting the implementation of proposals for experiments and modifications that are of major safety significance or that may have a significant effect on safety, the operating organization should submit the appropriate documentation to the regulatory body for review and assessment. Detailed guidance on utilization and modification projects is provided in Ref. [6].

4.13. Before the authorization for eventual decommissioning and release from regulatory control can be obtained, the application submitted to the regulatory body for review and assessment should include:

— The records and results of operational experience;
— The decommissioning programme.

Detailed guidance on decommissioning is provided in Ref. [9].

Appendix

CONTENT OF A SAFETY ANALYSIS REPORT

The Appendix has been divided into 20 sections dealing with standard specific topics that are addressed in the safety analysis report for a research reactor. The section headings of the Appendix are, in general, the headings that may be appropriate for the different chapters of the safety analysis report. The areas in which basic information is required by the regulatory body — such as site characteristics, reactor descriptions (and safety system descriptions), conduct of operations, commissioning, safety analysis, operational limits and conditions, management system, radiation protection and emergency planning — are emphasized. In particular, considerable attention is given to the safety assessment of modifications and experiments as related to the usage of the reactor.

CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF THE RESEARCH REACTOR

A.1.1. This chapter of the safety analysis report should include an introduction to the report and general information regarding the research reactor and associated facilities, in order to provide an adequate overall picture of the research reactor.

General description of the research reactor

A.1.2. In this section, a summary of the principal characteristics of the research reactor and the site should be provided. The general arrangement and layout of the research reactor should be described, starting with the core and continuing with the secondary and tertiary systems and the reactor building, to convey an impression of the research reactor and its components. The reactor site and its environment should be briefly described. The features important to safety should be clearly identified. If the research reactor has novel features or involves unusual approaches to safety analysis, these should be outlined. A general description of the utilization and the experimental facilities that are foreseen should be included in this section.

Historical review

A.1.3. The operational history of the research reactor should be presented. For existing reactors, an overview of operational experience as well as of the major changes that have been made should be presented.

Comparison with other facilities

A.1.4. Any similarity with other facilities should be discussed. The design similarities, safety precedents and case histories from other facilities that will be referenced in the safety analysis report should be itemized.

Identification of the owner, the operating organization and representatives

A.1.5. The owner of the research reactor, the operating organization, the architect–engineer, the prime contractors and the consultants should be identified. It should be noted whether they have had previous experience with nuclear research facilities.

Safety features

A.1.6. This section should briefly state the safety principles adopted for the design, construction and operation of the reactor and the acceptance criteria to be used in the safety analysis. The safety features, components or systems incorporated into the research reactor that will be described in technical detail in the analysis should also be identified.

Experimental programme

A.1.7. This section should provide a brief description of the experimental programme to be pursued at the research reactor and the experimental facilities. The provisions needed for the experimental programme are addressed in Chapter 11 of the safety analysis report, and the safety analysis related to the experimental programme and the provisions is addressed in Chapter 16.

Material incorporated by reference

A.1.8. This section should tabulate reference information supporting the safety analysis report. This information may consist of, for example, computer codes and reports from reactor manufacturers and fuel manufacturers.

Requirements for further technical information

A.1.9. This section should identify those safety features or components for which further technical information, beyond that supplied in the safety analysis report, is required in support of the issue of a licence.

CHAPTER 2: SAFETY OBJECTIVES AND ENGINEERING DESIGN REQUIREMENTS

A.2.1. This chapter of the safety analysis report should identify, describe and discuss the safety objectives and the engineering design requirements of the structures, systems and components and other equipment important to safety.

Safety objectives and general design requirements

A.2.2. This section should describe the safety objectives and the general design requirements followed in the design of the reactor, in consideration of the requirements for normal operation, anticipated operational occurrences and the accidents taken into account in the design. Safety objectives and design requirements for accident mitigation should also be included. Other measures that can be used to mitigate accident conditions should be described in the appropriate chapters of the safety analysis report.

A.2.3. A statement of the overall safety objectives should be included. This should be followed by a brief description of the underlying safety objectives and general design requirements that are important to the design. Safety objectives are discussed in section 2 of Ref. [2], and general design requirements are discussed in section 6 of Ref. [2]. These objectives and requirements may include the following:

(a) Management system requirements;
(b) High standard of engineering design and, in particular, conservative design margins, engineered safety systems (features), barriers to radionuclide transfer and protection of these barriers;
(c) Inherent safety features (those relying only on physical properties);
(d) Passive safety features (passive features do not actively change state);
(e) The extent to which unique or unusual features that may affect the consequences or the probability of releases are incorporated;
(f) The extent to which redundancy, diversity and independence are applied in the design of engineered safety features;

(g) Fail-safe features;
(h) Defence in depth applied in the design;
(i) Accident prevention;
(j) Accident management;
(k) Proven engineering practice and use of generally accepted standards;
(l) Assessment of human factors and dependent failures;
(m) Radiation protection.

Emphasis should be placed on the principles used in design and not on a description of the reactor. The summary description of the reactor should be given in Chapter 5 of the safety analysis report.

Specific design requirements

A.2.4. The specific design requirements applied should be stated in this section. These requirements are discussed in detail in section 6 of Ref. [2] and include:

(1) Management system requirements for design, including codes of practice utilized in design.
(2) Monitoring of variables and control of reactor and system variables within their operating ranges.
(3) Reactor core integrity requirements.
(4) Protection against flow instabilities and suppression of power oscillations.
(5) Criteria for sharing of common structures, systems and components important to safety between facilities at the same site (e.g. emergency power supply, on-site fire brigade).
(6) Consideration of human factors and ergonomic principles to reduce the potential for human error and to relieve stress for the operating personnel.
(7) Requirements for design analysis with validated techniques, models or codes.
(8) Reactivity control and core design criteria, including:
    (a) Redundant reactivity control;
    (b) Reactivity limits;
    (c) Prevention of inadvertent criticality;
    (d) Shutdown margins;
    (e) Power peaking factors;
    (f) Maintenance of fuel design margins (e.g. burn-up level balancing with experimental requirements, residence time and water chemistry);
    (g) Design provisions to prevent, or to reduce the potential for, fuel loading errors.

(9) Core cooling criteria, including:
    (a) Requirements for adequate core cooling for all operational states and accident conditions;
    (b) Requirements for coolant system integrity and protection of the boundary from leakage.
(10) Fuel design limits and materials design criteria, including:
    (a) Fuel design bases for mechanical, chemical and thermal design;
    (b) Safety margins for fuel design parameters;
    (c) Methods of achieving a conservative safety margin for prototypical fuels;
    (d) Verification of fuel integrity;
    (e) Design bases for mechanical, thermal and chemical design of reactor materials important to safety.
(11) Design criteria for reactor utilization, including:
    (a) Radiation protection for all operational conditions;
    (b) Design requirements to ensure that safety system settings are not adversely affected (e.g. experiments influencing flux measurement);
    (c) Recognition of the interdependence between the reactor and any installed experimental equipment.
(12) Design criteria for the safety systems and, where required:
    (a) Provision of systems for shutdown, fuel cooling and control of radionuclide releases;
    (b) Operating requirements;
    (c) Separation requirements for safety system and control functions;
    (d) Single failure criteria;
    (e) Fail-safe mode requirements.
(13) Reliability requirements, including:
    (a) Operational (process) system reliability;
    (b) Reliability targets for safety systems;
    (c) Requirements for safety system redundancy and unavailability;
    (d) Segregation for independence or diversity;
    (e) Requirements for safety support systems.
(14) Design bases for equipment qualification for natural events, environmental conditions, fire protection and other hazards, for protection against loss or damage, and for protection against illicit acts such as theft of radioactive material or sabotage.
(15) Methods employed for protection against dependent failure.
(16) Capability for surveillance and maintenance of safety related equipment.
(17) Radiation protection in design, including:
    (a) Reduction of exposures through design features;
    (b) Control of radioactive releases;

(c) Control of radioactive material;
(d) Area classification and access control;
(e) Monitoring of fuel and waste storage areas.

Classification of structures, systems and components

A.2.5. If any scheme has been devised for the classification of structures, systems and components for purposes of analysis or design, such as for seismic safety or nuclear safety, the basis for the classifications and the list of classes should be presented in this section of the safety analysis report.

External events

A.2.6. In this section, the design criteria for the resistance of structures, systems and components to external events should be presented. These may include:

(a) Wind and tornado loadings;
(b) Water level (flood);
(c) Protection against missiles from internal and external sources, including aircraft;
(d) Seismic hazard and seismic analysis;
(e) Security related events, including attacks and theft of radioactive material or sabotage;
(f) Fire and explosions;
(g) Roof loadings from accumulated rain, snow, ice, dust or other natural materials.

Extreme weather conditions due to climate change should be taken into account for the determination of the external events. Additional information on siting requirements is presented in section 5 of Ref. [2].

Codes and standards

A.2.7. In this section, all codes and standards to be employed in the design of structures, systems and components should be listed. Justification for their use should be provided, particularly if they are relevant for nuclear safety.

A.2.8. If different codes and standards are used for different aspects of the same item or system, the consistency between them should be demonstrated. Typical areas covered by codes and standards are:

— Mechanical design, including stress analysis and fracture mechanics;
— Structural design;
— Earthquake resistant design;
— Selection of materials;
— Fabrication of equipment and components;
— Inspection of fabricated and installed structures, systems and components;
— Thermohydraulic and neutronic design;
— Electrical design;
— Design of instrumentation and control systems;
— Shielding and radiation protection;
— Fire protection;
— Maintenance, periodic testing and inspection as related to design;
— Design and production of fuel.

A.2.9. For items important to safety for which no appropriate established codes or standards exist, an approach derived from existing codes or standards for similar equipment should be applied. In the absence of such codes and standards, the results of experience, tests or analysis, or a combination thereof, may be applied, and an explanation of the results and their applicability should be given.

Technical design methods

A.2.10. This section should describe methods for design and analysis of structures, systems and components, including design transients, computer programs used, experimental stress analysis, and any programmes for dynamic testing and analysis of the mechanical systems and components. Particular attention should be paid to items important to safety.

Design for internal fire protection

A.2.11. This section should describe the design requirements for fire protection inside the research reactor. It should include passive features such as isolation, separation, selection of materials, building layout and zoning, location of fire barriers, and layout and protection of safety systems (including separation of safety related redundant systems). The fire protection system should be described in Chapter 10 of the safety analysis report (see para. A.10.8).

Qualification of components

A.2.12. This section should describe the design bases for qualification of components to resist such environmental factors as vibration, thermal expansion,

radiation, corrosion, dynamic effects, mechanical loadings and high pressure, high temperature, humidity, water, steam, chemicals, low temperature or a vacuum. Qualification tests and analyses that have been (or will be) performed should be described.

Conclusions

A.2.13. This section should provide the conclusion that the research reactor is designed to meet the overall safety objective and underlying safety objectives, and that appropriate external events, codes, standards and design methods have been considered in the design of the research reactor, including for the qualification of components.

CHAPTER 3: SITE CHARACTERISTICS

A.3.1. This chapter of the safety analysis report should provide information on the geological, seismological, hydrological and meteorological characteristics of the site and the vicinity, in conjunction with present and projected population distributions, land use, site activities and planning controls. The purpose is to indicate how these site characteristics have influenced the design of the research reactor and the operating criteria, and to show the adequacy of the site characteristics from the safety point of view. Additional information on siting is provided in section 5 of Ref. [2].

A.3.2. Information should be provided in sufficient detail to support the analysis and conclusions of Chapter 16 of the safety analysis report, to demonstrate that the reactor can be safely operated at the proposed site. For many low power research reactors, which present very limited hazards, the amount of detail provided in this chapter can be substantially reduced.

A.3.3. If a separate site evaluation report has been prepared, it should be referenced and only a summary should be presented in this chapter.

General site description

A.3.4. The location of the research reactor site should be specified and an area map should be provided that indicates:

(a) Research reactor property and boundary lines;
(b) Location and orientation of principal buildings and equipment;

(c) Location of any industrial, commercial or military facilities, and any institutional, recreational or residential structures;
(d) Nearby highways, roadways, airports, waterways and railway lines;
(e) Boundary lines of the area controlled by the operating organization;
(f) Boundaries for establishing release limits for effluents.

A.3.5. This section should describe the legal rights of the operating organization with respect to all areas that lie within the designated site area[22], as well as any activities unrelated to the operation of the research reactor that will be permitted in the site area.

External events

A.3.6. This section should describe the site related phenomena and characteristics, of both natural and human induced origin, that should be taken into account to assess the suitability of the site for the research reactor.

A.3.7. This section should describe the appropriate methods adopted for establishing the external effects that will constitute the postulated initiating events for important natural phenomena and human induced effects. Further information on design criteria for protection against these effects should be given in Chapter 2 of the safety analysis report (see para. A.2.6).

Geology and seismology[23]

A.3.8. The geology of the site and its environs should be described in this section in sufficient detail to identify effects that could present a hazard to the research reactor. A historical overview of reported earthquakes that could reasonably be expected to have affected the region surrounding the site should be presented.

A.3.9. Information that is used to establish the seismic design basis, such as earthquake return frequency and ground motion (including the static and

[22] The site area is the geographical area that contains an authorized research reactor, authorized activity or source, and within which the management of the authorized research reactor or authorized activity may directly initiate emergency actions [22].

[23] Requirements on site evaluation are established in Ref. [16].

dynamic stability of all soil or rock slopes, both natural and human made) should be presented in this section, as well as information for:

— Assessing the potential for surface faulting at the site;
— Defining the conditions and engineering properties of soil and/or rock supporting the reactor foundations;
— Assessing the potential for volcanic activity;
— Assessing the potential for liquefaction and ground motion.

Meteorology

A.3.10. This section should provide a meteorological description of the site and its surroundings, including wind speed and direction, air temperature, precipitation, humidity, atmospheric stability parameters and prolonged inversions. Seasonal and annual frequencies of weather phenomena — including, where applicable, hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, freezing rain, snow and ice, and sandstorms — should be provided.

Hydrology and oceanography

A.3.11. The surface and underground hydrology of the site and its environs should be described in this section, including the location, size, flow, water use and other characteristics of nearby freshwater courses. The location and characteristics of human made structures should be indicated, including dams, diversion channels and any flood control measures. Foreseeable changes in land use that may influence hydrology should be described, for example, changes in runoff characteristics resulting from urbanization, or realignment of drainage channels.

A.3.12. A description of the groundwater hydrology in the vicinity of the research reactor should be presented, including the main characteristics of the water bearing formations and their interaction with surface waters, and data on the uses of groundwater in the region.

A.3.13. If the reactor is to be built by the coast, oceanographic and hydrographic information, including a bathymetric map of the near-shore area in front of the location of the reactor, should be provided.

A.3.14. Natural phenomena to be considered in the safety analysis report may include, where appropriate:

— Flooding;
— Surges, seiches and wave action, including effects of ice ridges;
— Seismically induced phenomena such as tsunamis and dam failures.

Nearby industrial, transport and military facilities

A.3.15. All present or projected industrial, transport and military facilities that could pose a hazard to the research reactor should be described in this section; for example, significant manufacturing or chemical plants, refineries, storage facilities, mining and quarrying operations, military bases or sites, transportation routes (by air, land and water), transport facilities (railway lines, docks, anchorages, airports), oil and gas pipelines, drilling operations and wells, and underground storage facilities. The potential adverse effects that such facilities could have on the reactor (e.g. aircraft crashes or other transport accidents) should be described.

A.3.16. Foreseeable significant changes in land use should be considered, including expansion of existing facilities or activities, or the construction of high risk facilities.

Radiological impact

A.3.17. This section should describe radiological aspects and, in particular, the biological aspects of transfers of radioactive material to people. Most of these details may not be required for low hazard, low power reactors. In this case, only a brief summary should be given under each heading. If no radiological impact section is provided, justification should be provided for omitting this section of the safety analysis report.

A.3.18. Information should be included that, in combination with details of radioactive discharges and of the radionuclide behaviour and transfers presented in other chapters of the safety analysis report, will permit an assessment of the doses to individuals and to the population, and of any contamination of flora and fauna and food chains. This information should cover the entire region likely to be affected, with account taken of topographical, hydrological and meteorological characteristics.

Population distribution

A.3.19. The population distribution around the research reactor and in the region, including seasonal and daily variations, should be presented in this section. In

particular, information on existing or projected population distributions around the reactor should be collected and kept up to date during the lifetime of the research reactor.

Natural environment and land and water usage

A.3.20. The characteristics of the regional ecology and the uses of land and water should be summarized in this section, including:

(a) Land and bodies of water supporting wildlife;
(b) Land devoted to agricultural use;
(c) Land devoted to livestock or dairy farming;
(d) Land devoted to commercial, residential or recreational purposes;
(e) Bodies of water used for commercial or sport fishing;
(f) Bodies of water used for commercial purposes or recreation;
(g) Direct and indirect pathways for radioactive contamination of food chains.

Baseline radiological levels

A.3.21. This section should include a description of radioactivity due to both natural and artificial substances in air, water and ground (including below the surface), and in flora and fauna. If there was a nuclear installation on the site in the past, a brief description of any incidents that led to residual radioactive material at the site should be provided.

Atmospheric dispersion of radioactive material

A.3.22. This section should describe the models used to assess the atmospheric dispersion of radioactive material released under operational states and under accident conditions of the reactor, in accordance with the policies of the operating organization and the regulatory body. It should be stated whether the dispersion estimates are based on representative meteorological data or on conservative, worst weather assumptions. The scope of the models should include any unusual site and regional topographic features, and characteristics of the research reactor that may affect atmospheric dispersion. The accuracy and validity of the models, including the suitability of input parameters, the source configuration and the topography, should be discussed.

A.3.23. Where appropriate, this section may provide the results of calculations of atmospheric diffusion parameters at the site boundary and at off-site locations, or

may refer to radionuclide atmospheric concentrations and dose calculations, which should be presented in Chapters 12 and 16 of the safety analysis report.

Dispersion of radioactive materials through surface waters and groundwater

A.3.24. This section should indicate locations near the research reactor where radionuclides could be discharged or where they could enter surface waters or groundwater. The results of hydrological and hydrogeological investigations that have been carried out to assess, to the extent necessary, the dilution and dispersion characteristics of bodies of water should be presented.

A.3.25. The models used to evaluate the possible impact of the contamination of surface waters and groundwater on the population should be described. Where appropriate, the results of off-site dose calculations should be provided, or reference to such calculations should be made in Chapters 12 and 16 of the safety analysis report.

Adequacy of the site for emergency measures

A.3.26. This section should consider:

— Population distributions and projected population changes in the region surrounding the research reactor;
— Present and projected land use and water use in the region;
— Potential radioactive source terms, and doses to the population from direct exposure to radiation fields and from airborne radioactive material and aqueous pathways;
— Potential contamination of food chains;
— Potential exposures of site personnel;
— The need to control activities unrelated to research reactor operation in the controlled area or to evacuate persons engaged in these activities;
— The capability of the appropriate authorities to implement emergency measures if required;
— The feasibility of emergency plans (if they are required), with account taken of the population distribution, national and international boundaries, special groups (e.g. in hospitals), special geographical features (e.g. islands), the availability of evacuation routes and refuges for evacuees, and communication and transport provisions.

Monitoring of site related parameters

A.3.27. This section should define site related parameters that could be affected by the external events that have been taken into account for the analyses (e.g. parameters that could be affected by seismic, atmospheric, water and groundwater related events, and demographic, industrial and transport related factors). The strategy for monitoring, the provisions for monitoring and the use of the results in preventing, mitigating and predicting the effects of site related hazards should be described.

Conclusion

A.3.28. This section should provide the conclusion regarding the acceptability of the site for the research reactor under consideration. If further analysis is required to support the conclusion concerning acceptability, site characteristics should be identified and reference to the appropriate sections of the safety analysis report should be made. It should be stated that the radiological risks to the population from accident conditions, including those that may require implementation of mitigation measures, is acceptably low and in accordance with national requirements.

CHAPTER 4: BUILDINGS AND STRUCTURES

Reactor building

A.4.1. This section should contain a description of the reactor building and internal structures (e.g. reactor pools and internals, supporting structures, cranes, ventilation systems), emphasizing those characteristics of the building that assist in maintaining acceptable radiation levels on and off the site for all operational states. Information on the requirements of the reactor building is presented in Ref. [2] (paras 6.120–6.130 and 6.167–6.169).

A.4.2. The description should include the design basis of the building and internal structures, together with the design basis of the building penetrations (air locks, doors, etc.) in relation to their resistance to internal and external events (see paras A.2.11 and A.3.7).

A.4.3. The design and operation of the ventilation systems should be described, including requirements for containment or means of confinement, and including the ventilation exchange rates for the different operation modes. If applicable,

distinction should be made between the system used in normal operation and the system used for emergencies. The specific efficiencies of the air filters and iodine traps should be given.

A.4.4. The design and operation of reactor building subsystems should be described, such as a system for controlling the release of fission products.

A.4.5. The design and operation of cranes or other lifting devices should be described.

A.4.6. The descriptions required in paras A.4.1–A.4.5 should be supported by means of drawings, including flow and instrumentation diagrams.

A.4.7. Permissible limits as well as testing and inspection requirements for the subsystems should be described, in particular those for ensuring the prescribed leaktightness and leak rates.

Auxiliary structures

A.4.8. This section should include a description of auxiliary buildings and reactor structures important to safety.

CHAPTER 5: THE REACTOR

A.5.1. This chapter of the safety analysis report should provide all the necessary information to demonstrate that the reactor is capable of fulfilling its safety functions. These functions are:

- Shutting down the reactor and maintaining it in a safe shutdown condition for all operational states or accident conditions;
- Providing for adequate removal of heat from the core after shutdown, including in accident conditions;
- Containing radioactive material so as to minimize its release to the environment.

A.5.2. This chapter should provide information pertaining to operational states, including the parts of the safety analysis dealing with them. The consequences of failures and accidents are treated in Chapter 16 of the safety analysis report.

Summary description

A.5.3. The chapter should start with a summary of the functional, technical and operational characteristics of the reactor. Drawings, flow sheets and tables should be provided for illustration and support. Annex III presents items that should be considered in the description. The description should indicate the dependent and interrelated safety functions of the main reactor components.

Fuel elements

A.5.4. Basic information on fuel design and fuel properties should comprise:

(a) Fuel material, enrichment, composition and metallurgical state (oxide, alloy, etc.);
(b) Material (type, composition, etc.) of all other fuel parts, such as cladding, spacers and fittings, and burnable neutron absorbers;
(c) Fuel geometry, dimensions, tolerances, etc. (together with drawings);
(d) The material properties required for the analyses mentioned in paras A.5.5–A.5.8;
(e) The maximum temperatures to which the fuel elements can be subjected without deformation (due to blister formation or mechanical weakening);
(f) Fuel element instrumentation, if any.

A.5.5. An analysis should be provided that shows that the fuel elements can withstand the thermal conditions to which they are subjected throughout their normal operational life cycle. This life cycle should comprise not only nuclear applications in the reactor core but also the periods of storage, handling and transport.

A.5.6. An analysis should be provided that shows that the fuel elements can withstand the mechanical forces to which they are subjected (hydraulic forces, differential thermal expansion effects, etc.) without breach of mechanical integrity or undue deformation. The anticipated effects should be quantified.

A.5.7. An analysis should be provided that shows that the fuel element cladding can withstand the chemical environment to which is subjected during use and storage, with account taken of the effects of temperature and irradiation.

A.5.8. An analysis should be provided that shows that the intended irradiation conditions and limits (fission, density, total fissions at the end of lifetime, etc.) are acceptable and will not lead to undue deformation or swelling of components

that may contain fissile material. The anticipated upper limit of the eventual deformation (e.g. expressed as minimum cooling channel width) should be provided for the thermal safety analysis.

A.5.9. These analyses and this information should be supported by a report on experimental measurements and irradiation experience, and should include the entire fuel cycle (storage, transport, etc.).

Reactivity control system

A.5.10. Information should be provided that demonstrates that the reactivity control systems can fulfil their designated safety functions under all foreseeable operating conditions. Only the safety functions ensuring reactivity control (such as insertion capability) should be addressed here. All other aspects of reactivity should be treated in the section on nuclear design (see paras A.5.13–A.5.16). Incorporation of the protection system and power regulating systems is treated in Chapter 8 of the safety analysis report.

A.5.11. Basic information should be provided on the design of reactivity control systems, including materials, redundancy and diversity aspects, anticipated performance characteristics (such as drive speed and actuation and insertion times), fail-safe features, etc.

A.5.12. An analysis should be provided that shows that the reactivity control system will function properly in all operational states of the reactor and that it will maintain its reactor shutdown capability under all foreseeable accident conditions, including failures of the control system itself. Foreseeable ageing effects due to deterioration of properties as well as irradiation damage should be taken into account.

Nuclear design

A.5.13. An analysis should be provided that shows that the nuclear conditions in the reactor core are acceptable throughout its anticipated core cycle. The analysis should include the steady state and the dynamic nuclear and thermal characteristics of the reactor.

A.5.14. Basic information on the nuclear design should include:

(a) Core configuration and composition, such as the type and anticipated loading pattern of fuel elements, control elements and other components that affect the nuclear properties of the core. Since core configurations for research reactors may change with the changing experimental applications and requirements, the analysis may use a standard core configuration that has conservative properties with respect to all other configurations. An explanation of the intended fuel replacement strategy should complement this information. The information should be supported by drawings.
(b) Horizontal and vertical distributions of the neutron flux in the core at thermal neutron and fast neutron energy levels.
(c) Basic reactivity characteristics of the core such as the infinite and the effective multiplication factors; the anticipated effectiveness and the position of control elements during core lifetime; minimum shutdown capacity; reactivity feedback properties with regard to temperature, void, etc.; and reactivity worth of individual core components (fuel elements, irradiation devices, etc.).

A.5.15. The basic information should be supported by reference to the calculational methods and codes used, experimental verification of the basic input data, or other information that supports the validity of the nuclear properties, details of which are supplied in this section.

A.5.16. An analysis should be provided that shows that the effectiveness, speed of action and shutdown margin of the reactor shutdown system[24] are acceptable, and that a single failure in the shutdown system will not prevent the system from completing its safety functions when required. A sufficient shutdown margin should be provided so that the reactor can be brought to and maintained in a subcritical state in all operational states and accident conditions.

Thermohydraulic design

A.5.17. Information should be provided to prove that, in all operational states, adequate capacity for core cooling will be available to keep the reactor fuel in a

[24] For reactor designs that feature more than one shutdown system, the analysis should cover all of them.

thermally safe condition, and that an adequate thermal safety margin will be maintained to prevent or to minimize fuel damage under accident conditions.

A.5.18. Basic information on thermal and hydraulic core design should include:

(a) All safety related hydraulic characteristics of individual core components and of the core as a whole (such as average and local coolant velocities, and coolant pressures, as appropriate) for operational states during forced and natural convection cooling;
(b) The power distribution, including power peaking factors, in all core components that may contain fissile materials, as derived from the nuclear design characteristics provided in para. A.5.14(b).

A.5.19. The information should be qualified by reference to the analyses, experimental measurements, fabrication specifications, etc., from which it is derived, thus providing a quantitative assessment of the uncertainties for each of the safety relevant parameters that have been quantified.

A.5.20. An analysis should be provided that proves that the maximum thermal load to which any fuel element in the reactor is subjected in any operational state does not exceed the available cooling capacity, whether cooling by forced convection or natural convection. The limiting criteria that are to be applied for this analysis may be related to nucleate boiling, flow instability, inlet vortexing, departure from nucleate boiling, etc. (depending on the reactor type and operating conditions), and should be verified and qualified. All correlations used to determine the thermohydraulic load and void fractions should be clearly described, together with the justification for their applicability.

A.5.21. The analysis should lead to the determination of a thermal safety margin for the core, both for 'best estimate' conditions (based upon nominal thermohydraulic conditions) and for 'conservative' conditions (with account taken of the uncertainty values as derived in para. A.5.19).

A.5.22. The assessment should take into account changes to safety relevant fuel parameters that may be caused by mechanical deformation, irradiation swelling, etc., as mentioned in paras A.5.6 and A.5.8.

Reactor materials

A.5.23. Information should be provided that shows that all materials that have been selected for the construction of safety relevant structures and components

can withstand the nuclear, thermal and chemical environments to which they will be subjected, without unacceptable worsening of the performance of the safety functions of such structures and components. Ageing effects due to the deterioration of properties as well as irradiation damage should be included. Materials with low activation properties should be considered in the process of selection of materials.

A.5.24. Items that should be considered include:

(a) Core support and hold down structure;
(b) Safety relevant reactor internals such as guides of the reactivity control mechanism;
(c) The reactor tank and related components constituting the primary coolant boundary;
(d) Support structures for the reactor tank, safety instrumentation, irradiation facilities, beam tubes, etc.

The information may be given as a list of all relevant materials, their safety specifications and anticipated conservative values of essential material properties at the end of their service life.

A.5.25. The information should be validated by reference to experimental measurements and experience. If such validation cannot be given, a material surveillance programme (periodic testing and inspection) carried out to verify essential material properties should be described.

CHAPTER 6: RESEARCH REACTOR COOLING SYSTEMS AND CONNECTED SYSTEMS

A.6.1. This chapter of the safety analysis report should provide a description of the reactor cooling systems that remove the heat from the reactor. The description should contain the main design characteristics and performance characteristics. It should be supported by schematic flow diagrams and an elevation drawing of the cooling systems.

Primary cooling system

A.6.2. The design and operation of the primary cooling system should be described in detail. The design and performance characteristics of the main components (pumps, valves, heat exchangers, piping) should be tabulated. A

flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of and the effects of irradiation on these materials should be specified. The reactor vessel, together with in-service environmental factors such as corrosion, fatigue, thermal stress cycling and ageing effects, should be described.

A.6.3. Methods utilized for leak detection and measures to minimize the loss of the primary coolant should be described. The potential consequences of a loss of primary coolant should be discussed.

A.6.4. The chemistry data for the primary coolant should be presented, including the effects of irradiation of the primary coolant.

Secondary cooling system

A.6.5. The design and operation of the secondary cooling system should be described in detail. The design and performance characteristics of the main components (pumps, valves, heat exchangers, cooling towers, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of and corrosion control measures should be specified. Ageing effects should also be discussed.

A.6.6. If the reactor uses a closed intermediate cooling system between the primary cooling system and the ultimate heat sink, this should also be described.

Moderator system

A.6.7. The design and operation of the moderator system should be described in detail. The calculation of the heat generated in the moderator should be presented. The design and the performance characteristics of the main components of the moderator cooling system should be tabulated. A flow and instrumentation diagram of this system should be included, as well as drawings of the main components. The materials the components are made of should be specified; the effects of irradiation and corrosion should be discussed. Ageing effects should also be discussed.

Emergency core cooling system

A.6.8. The design and operation of the emergency core cooling system should be described in detail. The accidents for which this system is designed should be

mentioned, and analyses should be provided to demonstrate that the system fulfils the requirements. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of should be specified, the effects of irradiation, if any, should be discussed, and any environmental effects and ageing effects should also be discussed. The procedures for inspection and testing of the emergency core cooling system should be described.

Decay heat removal system

A.6.9. The design and operation of the decay heat removal system, including the ultimate heat sink, should be described in detail. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of should be specified; the effects of irradiation, if any, and any corrosion and ageing effects should be discussed, as well as unfavourable environmental conditions for the ultimate heat sink.

Primary purification system

A.6.10. The design and the operation of the primary purification system should be described in detail, including the procedures for exchange of resins and the shielding used to protect personnel during this operation. This may be described in this section, or reference may be made to Chapter 10 of the safety analysis report.

A.6.11. The design and performance characteristics of the main components (pumps, valves, filters, resins, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of should be specified. The means for monitoring performance and renewing the system's ability to purify the coolant should be described.

Primary coolant make-up system

A.6.12. The design and operation of the coolant make-up system may be described here, or reference may be made to Chapter 10 of the safety analysis report. The relevant chemistry control and chemistry data of the coolant should be presented (e.g. details of new water treatment, degassing and demineralizing processes).

CHAPTER 7: ENGINEERED SAFETY FEATURES

A.7.1. This chapter of the safety analysis report should identify and provide a summary of the types, locations and functions of the engineered safety features provided in the research reactor. Examples of engineered safety features are an emergency core cooling system and a containment system or a means of confinement. The requirements of these systems and supplementary features are discussed in paras 6.115–6.130 of Ref. [2].

A.7.2. The design basis and various modes of operation of the engineered safety features should be discussed in detail. The accidents for which these systems are designed should be presented, and analyses should be provided to demonstrate that the systems fulfil the requirements. The subsystems that are essential for the proper operation of the engineered safety features should be described (e.g. uninterruptible power supply for the emergency core cooling system). The extent to which the engineered safety features are automated and the conditions for which manual override is warranted should be clearly indicated.

A.7.3. Information should be provided on:

(a) Component reliability, system interdependence, redundancy, diversity of fail-safe characteristics and physical separation of redundant systems;
(b) Evidence that the material used will withstand the postulated accident conditions (radiation levels, radiolytic decomposition, etc.);
(c) Provisions for tests, inspections and surveillance (including those performed under simulated accident conditions) to ensure that the feature will be dependable and effective upon demand;
(d) Effects of ageing on the operability of the engineered safety feature.

A.7.4. Reference should be made to the relevant chapters of the safety analysis report or to other documents where the engineered safety features are described further.

CHAPTER 8: INSTRUMENTATION AND CONTROL SYSTEMS

A.8.1. This chapter of the safety analysis report should provide information regarding the instrumentation and control systems of all safety systems and safety related items and systems. The information provided should emphasize those instruments and associated equipment that affect reactor safety. The

requirements for instrumentation and control systems are established in paras 6.136–6.144 of Ref. [2].

A.8.2. All instrumentation and control systems and supporting systems (with emphasis on safety systems and safety related systems), including alarm, communication and display instrumentation, should be listed, and considerations of instrumentation errors should be included. Adequate schematic diagrams should also be provided.

A.8.3. Information on provisions for testing the instrumentation and control system should also be included. It should be demonstrated that ageing effects and obsolescence of components have been considered in the design, especially for those components that cannot readily be replaced.

Reactor protection system

A.8.4. The requirements for the reactor protection system are discussed in paras 6.95–6.105 of Ref. [2]. The reactor protection system, including all its components, should be described in detail. A schematic diagram should show how the parameters for initiating protective actions are derived from monitored process variables such as neutron flux, temperatures and flow, and how these parameters are logically combined.

A.8.5. The adequacy of the protection system to shut down the reactor in a safe manner (e.g. by providing redundancy) and to bring the reactor into a safe condition should be described. It should be demonstrated that the protection system will perform its function on demand, especially in cases of common cause and common mode failures, as well as with single failures.

A.8.6. For computer based digital protection systems, evidence of software verification and validation should be included.[25]

A.8.7. The means for detecting failures within the reactor protection system should be described.

A.8.8. This section should describe the methods used to prevent adverse environmental conditions (e.g. conditions of temperature, humidity, high voltage,

[25] Guidance on verification and validation of software is provided in Ref. [23].

electromagnetic fields) from influencing the reactor protection system, as well as methods to protect against tampering.

Reactor power control system

A.8.9. All elements of the reactor power control system should be described (including the design criteria and functionality). Any interfaces between the power regulating system and the reactor protection system should be identified and analysed to confirm that they do not lead to a degradation of safety.

Other instrumentation and control systems

A.8.10. All other instrumentation systems required for safe operation should be described, such as:

— The fire protection system;
— The experimental control system;
— The ventilation control system;
— The secondary cooling system;
— The coolant chemistry control system;
— The radiation monitoring system;
— The seismic monitoring system;
— The monitoring system for external meteorological and hydrological conditions.

Alarm system

A.8.11. The alarm system that indicates an abnormal status of the research reactor and failures within the safety systems should be described.

Interlocks

A.8.12. All interlocks that are provided for reactor operation and the relevant logic should be listed and described.

Control room

A.8.13. This section should include a description of the instrumentation systems that are provided in the reactor control room for indicating the status of the protection system, the reactor power regulation system and other important systems.

A.8.14. It should be demonstrated that sufficient information and means are available in the reactor control room to enable the operating personnel to carry out the required actions.

A.8.15. The information required in emergencies, including information available in the emergency control room, where provided, should be discussed.

CHAPTER 9: ELECTRIC POWER

A.9.1. This chapter of the safety analysis report should describe the AC and DC power supplies, with the emphasis on their dependability and their relationship to safety. The descriptions should be supported by adequate diagrams. The adequacy of each power supply should be demonstrated, and ageing effects that could affect safety should be discussed.

Off-site power supply

A.9.2. This section should describe the off-site power supply and should emphasize the design and performance characteristics.

Emergency power supply

A.9.3. This section should describe the design and operation of the emergency power supply and should emphasize the connection to the off-site power supply.

A.9.4. The description should include:

(a) The dependability of the system;
(b) The starting load requirements of the equipment powered by the system;
(c) The starting time of the system and the time sequence for connecting loads;
(d) The starting method (automatic or manual);
(e) The duration of operation with and without diesel backup.

Uninterruptible power supplies

A.9.5. The design and operation of the AC and DC uninterruptible power supplies, including the connection to the emergency power supplies, should be described. The capacities of the power source should be specified and compared with the requirements of the safety related loads.

Cables and routeing

A.9.6. Information should be provided on the types of cable used. The adequacy of the measures employed to separate the cables so as to maintain redundancies, to prevent interference between cables and to provide fire protection should be demonstrated.

CHAPTER 10: AUXILIARY SYSTEMS

A.10.1. This chapter of the safety analysis report should provide information concerning the auxiliary systems included in the research reactor. The description of each system, the design bases for the system and for critical components, a safety assessment demonstrating how the system satisfies the requirements of the design basis, information on the testing and inspection to be performed to verify the capability and dependability of the system, and information on the instrumentation and control system required should be provided. In cases where auxiliary systems are not related to the protection of the public against exposure to radiation, enough information should be provided to allow understanding of the design and function of the auxiliary system; emphasis should be placed on those aspects that might affect the reactor and its safety features or that might contribute to the control of radioactive material inside the research reactor. For those systems, foreseeable ageing effects that could affect safety should also be discussed.

Fuel storage and handling

A.10.2. This section should describe systems for storing fresh fuel and spent fuel, for cooling and cleaning the spent fuel pool (where applicable), and for handling and, if necessary, cooling the fuel during transfer within the research reactor. The quantity of fuel to be stored and the means for maintaining subcriticality, even during adverse seismic conditions, should be provided.

A.10.3. Fresh fuel handling and storage, including the tools and systems used, should be described. A brief description of the operating procedures for fuel handling should also be given (see para. A.13.10).

A.10.4. Information concerning the management of irradiated fuel should be provided (i.e. the activity, decay rate, fuel burnup history, refuelling frequency, and inspection and storage requirements), including the management of damaged fuel, as appropriate.

Water systems

A.10.5. Each water system of the research reactor that has not been described previously should be discussed in this section. These may include the service water system, the cooling system for reactor auxiliaries and the makeup system for demineralized water. In each case, the information provided should include the design bases, a system description, flow and instrumentation diagrams, a safety assessment if required, testing and inspection requirements, and instrumentation requirements.

Process auxiliaries

A.10.6. All auxiliary systems associated with the reactor process system and the experimental facilities, such as compressed air systems, process sampling systems, or equipment and floor drainage systems, should be discussed in this section. The discussions should include the design bases, a system description, a safety assessment, testing and inspection requirements, and instrumentation requirements.

Air conditioning, heating, cooling and ventilation systems

A.10.7. The ventilation systems for all areas except the reactor building (see Chapter 4 of the safety analysis report) should be discussed in this section. A system description should also be provided.

Fire protection

A.10.8. A description and a safety analysis of the fire protection system should be provided in this section, including information on procedures and maintenance activities. Reference can also be made to the design methods (see para. A.2.11).

Other auxiliary systems

A.10.9. In this section, the design bases, system descriptions and safety analysis should be provided for the other auxiliary systems, such as general communication systems, sanitary provisions, sewerage systems and gas service systems.

CHAPTER 11: RESEARCH REACTOR UTILIZATION

A.11.1. This chapter of the safety analysis report should describe the expected experimental use of the research reactor and should provide information demonstrating that provisions have been made to ensure that the experimental facilities and experiments are within the safety criteria established for the research reactor, the staff and the public. Requirements are established in Ref. [2], and guidance is provided in Ref. [6].

Experimental facilities

A.11.2. This section should provide a description of the design basis and of the design, as far as appropriate, as well as a safety analysis for all experimental facilities associated directly or indirectly with the research reactor. Such facilities may include the beam tubes, the thermal column, in-core or moderator facilities, boreholes and experimental loops. Ageing effects that could affect safety should also be discussed.

A.11.3. The method of review and approval for new experimental facilities together with the administrative procedures and controls to be employed should be described. Special attention should be given to the methods that will be utilized to review and approve new experimental facilities that are outside the scope of the facilities discussed in the safety analysis report.

A.11.4. For experimental facilities not yet defined in detail, the design basis should be presented. A dedicated safety analysis report for these facilities should be developed and approved at a later stage.

A.11.5. Materials that will not be allowed to be used in experiments in or near the reactor core should be specified, together with materials that may be utilized only under additional safety conditions.

CHAPTER 12: OPERATIONAL RADIATION SAFETY

A.12.1. This chapter of the safety analysis report should describe, for normal operational conditions:

(a) The radiation protection programme, including the radiation protection policies and objectives of the operating organization;
(b) Sources of radiation at the research reactor;

(c) Research reactor design for radiation safety;
(d) Waste management systems[26];
(e) Dose assessment for normal operation;
(f) Conclusions.

A.12.2. The estimated radiation exposure of the staff and the public for accident conditions should be analysed in Chapter 16 of the safety analysis report. Exposure from anticipated operational occurrences should be within the bounds laid down in the accident analysis and, therefore, should also be described in Chapter 16. Planning for a radiological emergency is described in Chapter 20, and management of irradiated fuel should be treated in Chapter 10 of the safety analysis report.

Radiation protection programme

Radiation protection policy and objectives of the operating organization

A.12.3. This policy statement should endorse the radiation protection objective as stated in paras 2.2 and 2.3 of Ref. [2]. In particular, this section should summarize the authorized dose limits for both occupationally exposed personnel and the public, as well as the operational emission limits based on these dose limits. The regulatory requirements for maintaining exposures and releases of radioactive material, including radioactive waste and effluents, below the authorized limits should be described. The reference levels of doses and releases established by the operating organization to assist the research reactor management in applying the optimization principle[27] to ensure that radiation doses and operational emissions are as low as reasonably achievable and are below the authorized limits should also be described. The records that should be kept to prove that exposure to radiation is justified should also be specified.

A.12.4. The programme for radiation protection established and implemented by the operating organization of the research reactor, including the application of the optimization principle, should be described. The policy and arrangements for emission control at the research reactor, including the organizational policy

[26] In some cases, waste management systems and operational radiation safety are discussed separately.

[27] Guidance on the optimization principle can be found in Ref. [15].

concerning control and monitoring of releases and the evaluation of trends, should also be described.

Organization, staffing and responsibilities

A.12.5. This section should describe the administrative organization of the management and staff responsible for radiation protection, including the authority and responsibility associated with each position identified and the experience and qualifications of the personnel responsible for the health physics programme. As appropriate, the functional responsibilities of the health physics group in areas such as advising on radiation protection, support, training, monitoring, dosimetry and laboratory services, and administrative control of radioactive material should be included. Reference should also be made to the relevant management system procedures that are applicable to the activities in radiation protection.

Facilities, equipment and instrumentation

A.12.6. The health physics facilities and equipment, such as laboratories for analysis of radioactive material, equipment for contamination control and decontamination facilities, should be described, including the locations of these facilities, as well as the arrangements for maintenance and calibration of health physics instruments and for personnel monitoring (e.g. film badges, thermoluminescence dosimetry services).

A.12.7. This section should describe the radiation and contamination monitoring stations, including fixed hand and foot monitors, portal monitors (where used) and portable activity monitors located at these stations. The equipment and instrumentation, both portable and located in the laboratory, for performing radiation and contamination surveys, for contamination control between different access zones, for monitoring and sampling of airborne radioactive material, and for personnel monitoring should also be described.

A.12.8. Information should be provided on the protective clothing and equipment routinely used at the research reactor, including respiratory protective equipment.

A.12.9. Special equipment available for use in an emergency when high dose rates may prevail, and any special training of research reactor personnel in the use of this special equipment, should be described in the emergency plan (see para. A.20.3).

A.12.10. If separate documentation has been prepared to describe the health physics programme, this documentation may be referred to, with only a brief summary being given in this section.

Procedures and training

A.12.11. An overview of the written procedures for the radiation protection programme should be provided. Such procedures should be prepared in accordance with the relevant management system requirements and may include:

— The policy, methods and frequencies for conducting radiation surveys and air sampling;
— Effluent monitoring;
— Administrative measures for controlling access to or occupancy times in controlled areas;
— Control of contamination of personnel and equipment;
— Control of compliance with applicable regulations for the transport of radioactive material;
— The methods and procedures for personnel monitoring, including methods for recording, reporting and analysing results;
— The programme for assessment of internal radiation exposure, such as bioassay or whole body counting, and other related medical surveillance of personnel, in particular in cases of overexposure;
— The issue, selection, use and maintenance of protective equipment such as respirators;
— The methods for handling and storage of sources, radioisotopes or other radioactive material;
— The handling and disposal of radioactive waste.

A.12.12. Reference should be made to the operating procedures, which include provisions for controlling the doses to operating personnel in normal operation and during work for maintenance, in-service inspection and refuelling. Reference should also be made to the operating procedures, which include provisions for the monitoring of systems that collect, contain, store or transport radioactive liquids, gases or solids. Any procedures relating to experimental facilities, isotope production or laboratory activities should be referenced.

A.12.13. This section should describe the methods and procedures for controlling and evaluating the exposure of experimenters and other personnel (e.g. contractors and students) who are likely to have only a cursory knowledge of radiation protection procedures at the research reactor.

A.12.14. Reference should also be made to emergency operating procedures in Chapter 20 of the safety analysis report for emergencies at the research reactor during which dose rates may be high.

A.12.15. This section should give a brief description of the radiation protection training programme for the management and staff responsible for radiation protection, and for other personnel, including contractors and students.

Effluent monitoring programme

A.12.16. This section should describe the effluent monitoring programme carried out on the site and off the site. If off-site monitoring of effluents is done by the operating organization of the research reactor, the arrangements and responsibilities should be discussed.

Audit and review programmes

A.12.17. This section should describe the provisions for controlling the conduct of the radiation protection programme and its review.

Radiation sources at the research reactor

A.12.18. All normal potential radiation sources (contained sources and airborne radioactive material) due to reactor operation and all potential radiation sources throughout the research reactor that can be identified should be catalogued in this section. These sources are used as bases for shielding calculations, the design of ventilation systems, dose assessment, waste management and the determination of effluent releases.

A.12.19. For typical radiation sources that are shielded or contained, information should be provided on the form, location, geometry, isotopic content and activity. For typical liquid and airborne radioactive material, information should be provided on the form, location, isotopic content and concentrations.

A.12.20. Examples of sources of radiation or radiation fields can be found in Annex IV.

A.12.21. This section of the safety analysis report should provide drawings of the research reactor, showing the location of all typical sources.

Research reactor design for radiation safety

A.12.22. In the description of the design considerations for the research reactor and equipment, it should be demonstrated that possible external and internal radiation exposures of personnel and the public are based on the radiation protection policy described in para. A.12.23. A description should be included of how the design philosophy reduces the exposure of personnel, minimizes the undesirable production of radioactive material, reduces the need for and the time spent on maintenance and operational activities with the possibility of causing internal or external exposure, and keeps releases of radioactive material to the environment as low as reasonably achievable.

Access control and zoning

A.12.23. This section should describe how the layout of the research reactor provides for the necessary segregation of radioactive material from personnel and the public, and how it prevents other hazards. This layout may include zones that are classified according to their potential for contamination and/or exposure. Drawings should be provided showing the research reactor layout with the controlled and supervised areas. The section should also describe the access control measures that guard against personnel approaching areas with high radiation fields and potentially contaminated areas, or the control measures that prevent the placement of a radiation source (e.g. spent fuel or activated or irradiated material) in an area where personnel are present.

Shielding and protective features

A.12.24. The shielding required for the research reactor, associated facilities (e.g. beam tubes) and the radiation sources identified in paras A.12.18–A.12.21 should be described. The description should include the radiation levels external to the shielding at locations where occupancy may be required, as well as the materials, the criteria for penetrations of the shielding and the calculational methods used. The section should also describe other protective features, such as geometric arrangements (e.g. for distance) or remote handling methods to ensure that the exposures of research reactor personnel and of the public are within the relevant requirements and are based on the optimization principle. The description should include the methods for ensuring that beam tubes and other experimental facilities are adequately shielded against radiation streaming during experimental use.

Ventilation for radiation protection

A.12.25. This section should discuss the radiation protection aspects of the ventilation system on the basis of the description of the system in Chapter 4 or Chapter 7 of the safety analysis report.

Radiation monitoring systems

A.12.26. This section should describe the permanent monitoring systems for controlled and supervised areas, for effluents and for airborne radioactive material, including information on:

— Locations of monitors and detectors;
— Types of monitor and instrumentation (stationary or mobile, sensitivity, type of measurement, range, accuracy and precision);
— Types and locations of local and remote alarms, annunciators, readouts and recorders;
— Alarm or controller set points;
— Provision of emergency power supplies;
— Requirements for calibration, maintenance and testing;
— Automatic actions initiated or taken.

A.12.27. This section should describe the criteria and methods for ensuring that representative samples are obtained from the areas being monitored.

A.12.28. The radiation monitoring system or other systems that could be used in accident conditions should be described. Reference should be made to Chapter 16 of the safety analysis report for use of the system in the safety analysis, and to Chapter 20 for emergency measures regarding the application of monitoring under accident conditions.

Radioactive waste management systems

Solid radioactive waste

A.12.29. This section should describe the treatment of solid radioactive waste including, as applicable:

(a) The types and class of radioactive waste, the origins and quantities of solid radioactive waste, including the physical form, volume and isotopic compositions, and the measured or estimated activity;

(b) For wet radioactive waste, the methods of dehydration;
(c) The methods of collection, processing, packaging, storage and transport of radioactive waste.

Liquid waste

A.12.30. This section should describe the treatment of liquids that are considered to be radioactive waste, including:

(a) The types and quantities of liquid radioactive waste, and the origins, locations, forms and estimated activities of liquid radioactive waste;
(b) Diagrams of flow paths and flow rates, process equipment, storage tanks and release points for releases to the environment;
(c) Measures to separate radioactive effluents and non-radioactive effluents;
(d) Release targets;
(e) Requirements for the system capacity, redundancy and flexibility, and for the capability of the system to facilitate maintenance, reduce leakage and prevent uncontrolled releases to the environment.

A.12.31. The criteria for determining whether processed liquid radioactive waste will be recycled or discharged should be described, including the expected effluent concentrations tabulated by radionuclide released and the total annual radioactive releases to the environment. The dilution factors upon release should be given.

Gaseous waste

A.12.32. This section should describe the treatment of gaseous radioactive material that is considered to be waste, including:

(a) The types and quantities of gaseous waste, and the sources, locations, forms and calculated quantities of radionuclides;
(b) Diagrams of flow paths and flow rates, process equipment and release points for releases to the environment;
(c) Measures to separate radioactive and non-radioactive effluents;
(d) Release targets;
(e) Requirements for the system capacity, redundancy and flexibility, and for the capability of the system to facilitate maintenance, reduce leakage and prevent uncontrolled releases to the environment.

A.12.33. If applicable, design provisions to handle gaseous material with a potential for explosion should be described.

Dose assessment for normal operation

Doses to the public

A.12.34. This section should demonstrate that the combined effects of direct radiation and of releases of radioactive material from the research reactor do not result in off-site doses to the public that exceed authorized limits. In addition, measures to reduce the exposures on the basis of the optimization principle should be described.

A.12.35. If previous sections of this chapter of the safety analysis report have demonstrated that radioactive releases are a small fraction of the operational emission limits and are acceptable, and that both direct and indirect exposure to radiation are also within acceptable limits, this section should provide only a summary of all pathways of radiation exposure: airborne radioactive material, liquid radioactive material, and direct and indirect exposure to radiation.

A.12.36. If radioactive releases have not been treated in terms of operational emission limits, then this section should include a calculation of the individual doses, at the research reactor site boundary and at off-site locations, due to the effects of all releases. A description of the calculational assumptions, methods and tools should also be presented. It should be shown that the combined effects of all releases meet regulatory requirements for doses to the public.

A.12.37. This section should state the criteria to be used for determining that gaseous and liquid radioactive releases are at an acceptable rate. The effluent concentrations tabulated by radionuclide released and the total annual radioactive releases to the environment should be included, together with the methods, parameters and assumptions used in calculating these quantities.

A.12.38. In addition, for gaseous effluents, all points of release of radioactive material to the environment should be identified, providing for each quantity:

(a) The height of the release;
(b) The effluent temperature and the exit velocity;
(c) Assumptions made concerning the transport and dilution of the gases in the environment.

Occupational exposure

A.12.39. This section should present a diagram showing the radiation fields in normally occupied areas of the research reactor and in areas where maintenance activities will be performed. Estimated annual occupancy data for the controlled areas of the research reactor should be used to show that the expected doses are acceptable for the major functions, such as research reactor operation, conduct of experiments, normal maintenance, radioactive waste management, refuelling and in-service inspection. An estimate of the annual dose at the boundaries of the controlled area should be provided.

A.12.40. This section should demonstrate that the estimated radiation exposure of personnel due to inhalation in areas with airborne radioactive material is acceptable. If data are available, a summary of the annual doses to research reactor personnel should be provided.

Conclusion

A.12.41. This section should give a conclusion regarding the acceptability of the operational radiation safety programmes and the design features at the research reactor.

CHAPTER 13: CONDUCT OF OPERATIONS

A.13.1. This chapter of the safety analysis report should describe the organizational structure and the way in which the operating organization will conduct the operations of the research reactor. This should include the staffing, review and audit of operations of the research reactor; operating procedures; maintenance; testing and inspection; security aspects; and records and reports. Requirements on these topics are established in Ref. [2].

Organizational structure

A.13.2. The structure of the operating organization should be described in this section. The key personnel and the groups at the various operating levels of the research reactor should be illustrated in an organizational diagram. The functions, authority and responsibility of key personnel in the operating organization should be described.

A.13.3. Organizational functions for which it is planned to use off-site or external groups should be indicated.

A.13.4. This section should provide data on the personnel required in the different operational states of the research reactor.

Staff qualification and training

A.13.5. This section should describe the qualifications of key personnel.

A.13.6. This section should indicate the type of training required for various personnel and how often the required training will be provided. Any licensing or qualification requirement for the staff should be discussed. Training requirements for research reactor users and instructions for visitors, if any, should be given. If a simulator is available, the use of the simulator in the training and qualification of the staff should also be described in this section.

Review and audit

A.13.7. This section should describe the method for the review and audit of the safety aspects of research reactor operations. It should also describe the composition and qualifications of the review and audit group; the rules for group meetings; the items to be reviewed by the group, such as changes to the licence, to the operational limits and conditions, to the procedures and to the research reactor itself; modifications; new tests; experiments and procedures; and evaluation of unplanned events.

A.13.8. Information on the audit function of the group should be provided, including the items to be audited, the intervals between audits, and the ways in which audit findings will be addressed by the research reactor management within the management system programme for operation (see Chapter 18 of the safety analysis report).

Operating instructions and procedures

A.13.9. This section should describe the operating procedures or provide an overview of the operating manual that contains these procedures.

A.13.10. These written instructions and procedures (see also Ref. [19]) should include information on the following items:

— Reactor startup, operation and shutdown;
— Loading, unloading and movement of fuel and irradiated material;
— Inspection and testing of items important to safety, in particular the safety systems;
— Setting up, testing and performance of experiments with safety significance;
— Maintenance, in particular concerning major components or systems important to safety;
— Radiation protection;
— Response to anticipated abnormal occurrences, failures of systems or components, and accident conditions;
— Effluent monitoring and environmental monitoring;
— Emergencies;
— Physical protection (see paras A.13.12 and A.13.13);
— Fire protection.

The safety analysis report should describe how to perform major, minor and temporary modifications to procedures.

Maintenance, periodic testing and inspection

A.13.11. This section should describe the conduct of the maintenance, periodic testing and inspection programme for equipment and components of the research reactor, which should be based on the guidance provided in Ref. [24]. An overview is sufficient if the detailed programme is given in supplementary documents. The maintenance, periodic testing and inspection programme should provide information on:

(a) The system or equipment to be inspected or tested;
(b) The inspection or testing criteria;
(c) The inspection or testing intervals;
(d) The persons responsible for the maintenance, testing or inspection;
(e) Approval of maintenance work;
(f) Resumption of normal operation after maintenance.

Physical protection[28]

A.13.12. The measures taken to protect the research reactor against unauthorized access and sabotage, and to protect against unauthorized removal of fissile and radioactive material, should be described, including procedures for access to the site and to the research reactor, and the physical protection systems.

A.13.13. The physical protection measures of the research reactor should be kept confidential and therefore may be described in a separate document.

Records and reports

A.13.14. This section should provide information on the system for controlling records, data and reports that are important to safety. The records may comprise data on:

(a) Reactor operation (logbooks, strip charts, checklists, automatic data readout);
(b) Operational status (type and number of operational components and of components out of service);
(c) Maintenance, testing and inspection protocols;
(d) Records of modifications;
(e) Irradiation of samples and radionuclides produced;
(f) Movement of fissile material;
(g) Radiation levels;
(h) Radiation exposure (external and internal), radiation doses to personnel and records of medical examinations;
(i) Effluent monitoring and environmental monitoring results;
(j) Failures of and other events involving safety related components;
(k) Documents on training and retraining.

A.13.15. This section should give the minimum time interval for which records are to be stored in accordance with the management system for the operation of the research reactor (see Chapter 18 of the safety analysis report).

[28] Guidance on nuclear security is provided in the IAEA Nuclear Security Series [25–28]; see also Refs [29, 30].

Feedback of operational experience

A.13.16. This section should describe the process for the evaluation and feedback of operational experience, including the evaluation of trends in operational disturbances, trends in malfunctions, near misses and other incidents that have occurred at the research reactor and, as far as applicable, at other nuclear installations.

CHAPTER 14: ENVIRONMENTAL ASSESSMENT

A.14.1. This chapter of the safety analysis report should provide a summary of the environmental report for licensing actions including construction, operation, modification and decommissioning of the research reactor.

A.14.2. This chapter should briefly discuss the following points, in connection with the related information included in Chapter 3 of the safety analysis report:

(a) The environmental impact of the licensing action;
(b) Unavoidable adverse environmental effects;
(c) Alternatives to the licensing action that were considered;
(d) Irreversible and irretrievable commitments of resources;
(e) An analysis providing a balance of the environmental effects of the licensing action and the alternatives available for preventing or mitigating environmental effects, as well as a summary of the environmental, economic, societal, technical and other benefits deriving from the research reactor.

A.14.3. Some licensing actions may have little or no environmental effect. In these cases, the decision to take such actions should be stated and briefly justified.

CHAPTER 15: COMMISSIONING

A.15.1. This chapter of the safety analysis report should describe the technical aspects of the commissioning programme. For a research reactor under construction, this chapter should describe the commissioning programme in sufficient detail to show that the functional requirements of structures, systems and components will be adequately verified. For an existing research reactor this chapter should describe the commissioning programme that has been carried out

and the main results of the commissioning programme in sufficient detail to show that the functional requirements of structures, systems and components have been adequately verified. Complete details of the commissioning programme and the results of the commissioning, if completed, may be provided in a separate commissioning document.

A.15.2. The commissioning programme should describe the different stages, which are usually arranged according to the following sequence:

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

Research reactors under construction

A.15.3. This section should provide the following information concerning the commissioning programme:

(a) A summary of the programme and objectives;
(b) Details of the commissioning organization, including training requirements;
(c) An outline of the management system procedures for commissioning (see Chapter 18 of the safety analysis report);
(d) A summary schedule of the major phases of the programme;
(e) A summary of the operational limits and conditions for commissioning and of the commissioning procedures.

A.15.4. This section should contain a description of how information on the commissioning of similar operational facilities will be utilized. The method for reporting the results of commissioning to the regulatory body should be described, including resolutions regarding non-conformances or unexpected results.

A.15.5. This section should describe the method for updating the safety analysis report, if required, to include the results of commissioning tests.

Research reactors after commissioning

A.15.6. After commissioning of the research reactor, the paragraph on commissioning should be updated with the following information concerning the commissioning programme:

(a) A summary of the results;
(b) A summary of the major technical and organizational changes during the commissioning process;
(c) A summary of the accepted non-conformances and, where appropriate, their associated corrective actions;
(d) An overview of possible modifications of structures, systems and components, the safety analysis and the safety analysis report, procedures, etc.

Existing research reactors

A.15.7. For existing research reactor facilities, this section should provide the following information concerning the commissioning programme:

(a) A summary of the programme and objectives;
(b) A summary of the results;
(c) A summary of the accepted non-conformances and, where appropriate, their associated corrective actions;
(d) The method for updating the safety analysis report, if required, to include the results of commissioning tests of modifications.

Commissioning of modifications

A.15.8. The information outlined in paras A.15.1–A.15.7 should also be included in a safety analysis report involving modifications to existing research reactor facilities.

CHAPTER 16: SAFETY ANALYSIS

A.16.1. The safety analysis presented in this chapter forms the focal point of the safety analysis report. In previous chapters, it is stated that the research reactor design, and especially the design of structures, systems and components important to safety, should be evaluated for the susceptibility of structures, systems and components to malfunctions and failure. In this chapter, the effects of anticipated process disturbances and postulated component failures and human errors (postulated initiating events) should be described, including their consequences, to evaluate the ability of the research reactor to control or to accommodate such situations and failures.

A.16.2. To ensure completeness of presentation and to facilitate the review and assessment by the regulatory body, this chapter of the safety analysis report should contain the following information:

(1) Introduction — the general approach and methods used in the safety analysis (paras A.16.3–A.16.4);
(2) Research reactor characteristics — the reactor parameters and initial conditions used in the safety analysis (paras A.16.5–A.16.9);
(3) Selection of initiating events — the spectrum of initiating events considered in the safety analysis (paras A.16.10–A.16.12);
(4) Evaluation of individual events sequences — the results of the safety analysis (paras A.16.13–A.16.46);
(5) Summary — a summary of significant results and conclusions regarding acceptability (paras A.16.47–A.16.48).

Introduction

A.16.3. This section should provide an overview of the methods and approaches used in the safety analysis. The information provided should be sufficient for a reviewer to obtain a basic understanding of the methods used and of the general nature of the criteria used to assess the acceptability of the results. Annex I of this Safety Guide may be of some assistance in completing this section, but the level of detail of Annex I is not required here.[29]

A.16.4. This section should provide a brief summary, under the following headings:

(1) Methods of identification, selection and justification of initiating events.
(2) Methods of analysis, including where appropriate:
    (a) Event sequence analysis;
    (b) Transient analysis;
    (c) Evaluation of external events and special internal events;
    (d) Qualitative analysis;
    (e) Radiological consequence analysis.
(3) Acceptance criteria.

[29] Additional information is provided in Ref. [13].

Research reactor characteristics

A.16.5. This section should summarize the reactor parameters and initial conditions used in transient analysis (paras A.16.19–A.16.24). These parameters and permitted boundaries of operation will form the basis for the operational limits and conditions in Chapter 17 of the safety analysis report.

Core parameters

A.16.6. A summary should be given of the research reactor parameters and ranges for specified operating conditions considered in the safety analysis. Although these values may be tabulated in various other sections of the safety analysis report, they should be summarized here to assist in the review and assessment of the safety analysis. Such parameters should include, but are not limited to:

(a) Core power;
(b) Core inlet temperature;
(c) Fuel element cladding temperature;
(d) Reactor system pressure;
(e) Core flow;
(f) Axial and radial power distribution and hot channel factor;
(g) Power peaking factor;
(h) Excess reactivity;
(i) Reactor kinetics;
(j) Fuel reactivity coefficient and moderator temperature reactivity coefficient;
(k) Void reactivity coefficient;
(l) Available shutdown reactivity worth;
(m) Insertion characteristics of reactivity control and safety devices.

A.16.7. A range of values should be specified for reactor parameters that vary with fuel burnup, refuelling or other factors.

A.16.8. The permitted boundaries of operation for the system parameters should be specified, including permitted fluctuations in a given parameter and associated uncertainties. The most adverse conditions within the boundaries of operation should be used as initial conditions for transient analysis.

Functions of the research reactor protection system

A.16.9. The settings of all protection system functions that are used in the safety analysis should be listed. Typical protection system functions are reactor trip, isolation valve closures and provision of backup cooling.

Selection of postulated initiating events

A.16.10. This section should list the postulated initiating events that are treated in the safety analysis. The starting point of the safety analysis is the identification of the list of postulated initiating events. The list should be comprehensive, and justification for rejection of particular initiating events should be provided. Annex I to this Safety Guide provides some information on methodologies. The points mentioned in paras A.16.11–A.16.12 should be considered in the selection.

A.16.11. Each postulated initiating event should be assigned to one of the following categories, or grouped in some other manner consistent with the type of research reactor under study:

(a) Loss of electric power supplies;
(b) Insertion of excess reactivity;
(c) Loss of flow;
(d) Loss of coolant;
(e) Erroneous handling or failure of equipment;
(f) Special internal events including failure of experiments;
(g) External events;
(h) Human error.

A.16.12. The initiating events in each group should be evaluated to identify the events that would be bounding, and the events selected for further analysis should be indicated and justified. The events selected for further analysis should include those having potential consequences that are bounding for all other initiating events in the group.

Evaluation of individual events

A.16.13. The detailed information listed below should be given for each initiating event selected in para. A.16.12. This information is organized under the following headings:

(a) Identification of causes;
(b) Sequence of events and systems operation;
(c) Transient analysis;
(d) Classification of damage states;
(e) Derivation of source terms;
(f) Evaluation of radiological consequences.

A.16.14. The extent of the quantitative information that should be given for these topics will differ for the various initiating events and will depend on the type of research reactor. For those situations in which a particular initiating event is not bounding, only the qualitative reasoning that led to that conclusion should be given, together with a reference to the section presenting an evaluation of the more bounding initiating event. Furthermore, for those initiating events that require a quantitative analysis, it may not be necessary to provide such an analysis for each topic. For example, there are a number of events initiating a reactor transient that result in minimal radiological consequences. The safety analysis report should merely present a qualitative evaluation to show that this is the case. A detailed evaluation of the radiological consequences should not be performed for each such initiating event.

Identification of causes

A.16.15. For each event evaluated, a description of the causes that led to the initiating event under consideration should be included, both for initiating events due to equipment failure and for initiating events due to human error.

Sequence of events and systems operation

A.16.16. The step by step sequence of events, from event initiation to the final stabilized condition, should be described. The following should be provided for each event sequence:

(a) Identification of significant occurrences on a timescale, for example, flux monitor trip or start of insertion of control rods;
(b) Indication of the proper functioning of normally operating reactor instrumentation and controls, and of their failure to function;
(c) Indication of proper functioning of reactor protection and safety systems, and of their failure to function;
(d) Indication of the required operator actions;
(e) Evaluation of dependent failures and human errors;

(f) Qualitative evaluation of sequence probabilities (if employed);
(g) Justification for exclusion of sequences that are outside the design basis.

A.16.17. Not every postulated initiating event needs to be completely analysed and described. In the analysis of event sequences, logical models should be constructed for groups of initiating events to identify the fault sequences. These logical models start with the fundamental safety function and consider the required safety functions for the group of initiating events, the safety systems and the individual components of the safety systems. The bounding event sequences in each group that have been selected for further analysis should be indicated.

A.16.18. A systematic assessment should be carried out to identify the failures of safety system equipment that could occur following the initiating event. These failures should be included in the logical model.

Transient analysis

A.16.19. A detailed analysis of core and system performance should be described in this section. The methods used to characterize the performance of the reactor core and of the system under accident conditions should be discussed, and the important results of the analysis should be presented. The discussion should include, where appropriate, an evaluation of the parameters that may affect the performance of barriers that restrict the transport of radioactive material from the fuel to the environment (e.g. fuel–cladding interaction and fuel failure modes, the primary coolant system and the building or systems providing confinement).

Computational models

A.16.20. The computational models employed, including computer codes or analogue simulations used in the analyses, should be identified. It should be confirmed that the models are applicable for the expected range of operational parameters, that they yield conservative predictions, that they represent all important physical phenomena and that they have been properly validated. This section should provide only a summary of mathematical models and computer codes or lists used, referring to detailed descriptions in documents available to the regulatory body. The following should also be provided:

(a) A general description of the model, including:
    (i) The purpose of the model and its range of application, including the extent or range of variables investigated;

(ii) A summary description of the analytical models and empirical correlations used;
(iii) Any simplifications or approximations introduced in the analysis;
(iv) The degree of conservatism of the methods and correlations;
(v) The numerical accuracy of the model, including the estimated accuracy of results and factors contributing to the uncertainties;
(vi) The method combining these codes (if a set of codes is used).

(b) A brief description of input data for each model should be provided, including:
(i) The method of selection of input parameters, including their applicability and their degree of conservatism;
(ii) A listing of input data for each model;
(iii) The sensitivity of the model to particular input parameters.

(c) A summary of results of validation studies, including:
(i) Comparisons of model predictions with results of experiments or operation, or with other models that have also been compared with results of experiments or operation;
(ii) Demonstration of adequate numerical accuracy or of the degree of conservatism;
(iii) Confirmation that the modelling represents all important physical phenomena;
(iv) Confirmation that the empirical correlations are conservative, are based on experiment (where practicable) and are appropriate for the range of operational parameters.

Input parameters and initial conditions

A.16.21. The input parameters and initial conditions used in the analysis should be clearly identified. Annex II to this publication provides a list of examples of these items. However, the initial values of other variables and additional parameters should be included in the safety analysis report if they are used in the analysis of the event being analysed.

Results

A.16.22. The results of the analysis should be presented and described in the safety analysis report. Key parameters should be given as a function of the time of the transient or accident. The following are examples of parameters that should be included:

— Reactivity;
— Thermal power;
— Heat flux;
— Power distribution;
— Reactor cooling system pressure;
— Minimum critical heat flux ratio or departure from the nucleate boiling ratio, as applicable;
— Nuclear heating;
— Core coolant flow rates;
— Coolant conditions (inlet temperature, average core temperature, hot channel exit temperature);
— Core temperature (maximum fuel centre line temperature, maximum cladding temperature) and maximum fuel enthalpy;
— Reactor coolant inventory (total inventory and coolant level in various locations in the reactor coolant system);
— Parameters of the secondary heat exchanger system (inventory and level, enthalpy, temperature, mass flow rate).

A.16.23. Uncertainties in the results should be pointed out and discussed.

A.16.24. The margins between the predicted values of various core parameters and the values of these parameters that would represent the boundaries of acceptable conditions should be provided.

Classification of damage states

A.16.25. The transient analysis may show that the fuel design limits have been exceeded, resulting in some damage to fuel cladding. An estimate of the type of damage, the quantity of fuel affected and other factors (e.g. fuel and cladding temperatures, coolant characteristics, chemical interactions) should be provided.

A.16.26. Some event sequences may result in different radiological hazards, including failures of experiments or of irradiation and/or activation facilities and mechanical damage to the cladding of the irradiated fuel. An estimate of the form and content of the hazardous material, together with any physical parameters that further characterize its nature, should be provided. Any regrouping of the sequences within the class according to the type and the extent of radiological hazard should be described. Sequences that result in no hazard should be excluded, and the remaining sequences that are bounding or limiting for each category of hazard should be selected for analysis of the releases of radioactive material.

Derivation of source terms[30]

A.16.27. The source terms, if any, for each bounding sequence mentioned in the previous section of the safety analysis report should be described. Such a description should include the quantity of radioactive material that might be released from the research reactor, its physical and chemical form, and any other factors necessary to completely specify its potential dispersion in the environment. Factors which affect the source term, including the volatility of radionuclides, releases from the fuel, retention of fission products within the reactor coolant and retention of fission products inside the reactor building or means of confinement, should be taken into account.

A.16.28. This section should indicate whether detailed calculations of realistic release fractions have been performed or whether conservative release fractions have been employed, such as an arbitrary source term that is larger than expected for probable accident sequences (e.g. to demonstrate the effectiveness of the building or means of confinement, or to show that the resulting doses to critical groups would meet regulatory requirements).

A.16.29. Mathematical models used in determining and analysing the source term should be summarized, and information on validation should be presented. The information given in paras A.16.30–A.16.32 should be provided for each limiting event sequence, where appropriate.

Assessment of releases to the reactor building

A.16.30. The radionuclides released inside the building, the quantity of the specific radionuclides and other physical factors characterizing the releases should be described for each relevant sequence. The parameters and assumptions used in the analysis should be presented, including:

(a) The fission product inventory (or radionuclide inventory for accidents not involving fuel damage);
(b) The nature of the fuel element damage, and the fraction of the fuel cladding damaged;
(c) The fractions of the fission product released from the fuel;
(d) The retention factors and plateout of radionuclides in water and on surfaces.

[30] Additional information is provided in Ref. [31].

Assessment of releases from the reactor building

A.16.31. The radionuclides released to the environment, the quantity of the specific radionuclide and other physical factors characterizing the release should be given for each of the event sequences that results in releases to the reactor building. Releases of both airborne and aqueous radioactive material should be considered. The parameters and assumptions used in the analysis should be presented, including:

(a) Removal of radionuclides by liquid and gaseous hold-up systems, recirculation systems and ventilation systems, including filter efficiencies;
(b) Surface deposition and resuspension;
(c) Radionuclide hold-up time, decay time and precursor production;
(d) Reactor building leak rate or liquid effluent release rate;
(e) Release mode (single puff, intermittent, continuous);
(f) Release point (stack, ground level, etc.).

Assessment of other hazards

A.16.32. Descriptions should be given of accidents that might result in significant direct exposure of personnel or the public to radiation fields associated with any releases that are contained within the reactor building (see also para. A.16.38). Examples include:

— Inadvertent criticality;
— Releases from an experiment or the research reactor that are contained but that present a radiation hazard;
— Aqueous spills or other releases of radioactive material that are contained locally;
— Loss of shielding.

Evaluation of the radiological consequences

A.16.33. This section should discuss the calculational methods used to determine the possible radiological consequences of representative event sequences and should summarize the results of dose calculations. The information should be sufficient to substantiate the results and to allow an independent review to be performed by the regulatory body.

A.16.34. If no possible radiological consequences are associated with a given event sequence, this section should simply contain a statement to that effect.

Methods for analysis of the possible radiological consequences

A.16.35. The methods used to analyse the possible radiological consequences that might result from incidents should be presented in this section. The assumptions and methods used in determining the possible radiological consequences should be supported by providing adequate information, where appropriate; by referring to other sections within the safety analysis report; or by referring to other documents.

A.16.36. Information on the modelling of possible radiological consequences should include the following:

— A description of the mathematical or physical models employed, including any simplifications or approximations introduced into the analysis;
— A description of the meteorological data used to perform the calculations;
— A summary of the computer codes or analogue simulations used in the analyses, with reference to detailed descriptions;
— Information on the validation of the calculational methods used, including the restrictions and limitations on their utilization;
— Consideration of uncertainties in the calculational methods used, the performance of equipment, instrumentation response characteristics or other intermediate effects that were taken into account in the evaluation of the results.

Dose results

A.16.37. This section should present the results of the dose calculations giving the effective dose at the site boundary or the exclusion boundary[31] and, if necessary, the effective dose to the public at greater distances from the site. In these cases, the dose to the most highly exposed member of the public should be given, as well as the doses, in an accident, to the control room personnel and to personnel in other places on the site, where appropriate.

[31] The exclusion boundary is the boundary of the deliberate exclusion from the scope of regulatory control of a particular area of exposure of the research reactor on the grounds that it is not considered amenable to control by means of regulatory requirements.

External exposure

A.16.38. Consideration should be given to external exposure due to radiation arising from both aqueous and atmospheric releases, and to the possibility of ground contamination and gamma radiation from radionuclides deposited on the ground (‘ground shine’).

Radiation fields

A.16.39. Radiation fields associated with releases that occur within the research reactor and that could result in radiation doses due to external exposure should be described, together with estimates of doses to critical groups. The parameters and assumptions used in the analysis should be justified, including:

— The quantity of radionuclides released and the timescale of the release;
— Radionuclide decay and precursor production;
— Shielding parameters, buildup factors and scattering (e.g. for gamma radiation from radionuclides in an airborne plume (‘cloud shine’));
— Distance to critical groups and the timescale over which doses are calculated.

Aqueous releases

A.16.40. This section should summarize the assessment of aqueous releases and, where appropriate, dispersion in surface waters and groundwater, contamination of the flora and fauna and food chains, and the consequent doses to individuals and to the population. Reference should be made to paras A.3.11–A.3.14 for data on hydrological and hydrogeological characteristics of surface water and groundwater. The discussion of potential hazards should include:

— Radiation from released fluids;
— Evaporation or airborne radioactive material caused by resuspension of radionuclides from the released fluids;
— Ground contamination;
— Contamination of aquifers and reservoirs on and off the site.

A.16.41. Parameters and assumptions used in the analysis should be justified, including:

— Radionuclide removal by liquid hold-up systems or recirculation systems;
— Potential discharge points, the inventory of radionuclides released, their concentrations in the fluid, the release rate and the mode of release (single, continuous or intermittent release);
— Radionuclide decay and precursor production;
— Dilution and dispersion characteristics, including migration and retention characteristics of soils, radionuclide movement in hydrogeological formations, the reconcentration ability of sediments and biota, and other effects that may be needed to determine radionuclide movement and exposure pathways;
— Direct and indirect pathways for contamination of the food chain;
— Radionuclide uptake by humans and the consequent doses.

A.16.42. Special attention should be paid to ascertaining those characteristics important for the determination of food chain transport.

A.16.43. If the possibility of aqueous releases to surface water or groundwater aquifers is judged to be credible, the provisions for the containment of any liquid releases within the research reactor should be described and the possibility of failure of these provisions should be discussed.

Atmospheric releases

A.16.44. This section should present the doses to research reactor personnel and to the public after a release of airborne radioactive material from the research reactor, with account taken of atmospheric dispersion, where appropriate.

A.16.45. The parameters and assumptions used in the analysis should be presented and shown to be conservative, including:

— The source term, characterizing it in terms of the radionuclide inventory, the physical and chemical forms, and any other factors necessary to completely specify the dispersion of radioactive material to the environment, including buoyancy;
— Mode and characteristics of the release (single, intermittent or continuous release, release duration);
— Location of release and characteristics, including height and diameter of the stack;
— Distance to receptors and intervening terrain;
— Meteorological data, including wind speed and wind direction, and data on inversions and other atmospheric stability factors;

— Wake effects of the building;
— Diffusion parameters;
— The physical and chemical forms of radionuclides at the receptor location, and whether they are airborne or deposited;
— Results of dose calculations (for doses due to inhalation, ingestion and ground shine).

Ground contamination

A.16.46. This section should discuss possible ground contamination, either by direct dispersion of particulate radioactive material or by deposition from releases of airborne or aqueous radioactive material. The surface contamination by radionuclides should be estimated, and the doses (due to ingestion and ground shine) should be assessed.

Summary

A.16.47. This section should summarize the important results of the safety analysis, including a brief description of the dominant accident sequences. Significant conclusions arising from the analyses should be presented. The effect of uncertainties in the results should be discussed and evaluated.

A.16.48. The results of the analyses should be compared with the appropriate acceptance criteria. It should be shown that the criteria discussed in paras 2.14–2.18 have been met. An evaluation of the results should demonstrate that the design is acceptable and should confirm the validity of the operational limits and conditions discussed in Chapter 17 of the safety analysis report.

CHAPTER 17: OPERATIONAL LIMITS AND CONDITIONS

A.17.1. This chapter of the safety analysis report should contain the operational limits and conditions important to safe reactor operation that have been derived from the safety analysis. The operational limits and conditions represent an envelope of parameters, developed by the operating organization, that will protect the research reactor and that will protect personnel and the public from exposure and the environment from contamination if they are not exceeded. The operational limits and conditions should be understood by the responsible operating personnel. The operational limits and conditions include safety limits, safety system settings, limiting conditions for safe operation, and surveillance

and administrative requirements. Requirements are established in paras 7.29–7.41 of Ref. [2], and guidance is provided in Ref. [19].

A.17.2. The operational limits and conditions are based on an agreement between the operating organization and the regulatory body, and they form an important part of the requirements for authorization by the regulatory body of the operation of the research reactor. Changes to the operational limits and conditions should require a revision of the safety analysis report, and assessment and approval by the regulatory body.

A.17.3. Because of the important role of the operational limits and conditions in ensuring safe operation, each operational limit and condition should be selected and appropriately substantiated by a written statement of the reason for its adoption. This information should either be presented in a separate document or be included in this chapter of the safety analysis report. In the first case, the information on the operational limits and conditions given in the safety analysis report could be a summary of this separate document. In both cases, the information on each operational limit and condition should cover the following points:

(a) The objectives to be met by the establishment of operational limits and conditions (e.g. prevention of situations that might lead to accident conditions).
(b) The applicability of the operational limits and conditions, for example, to physical variables related to physical barriers, such as the fuel cladding temperature or pool water level, or to the conditions of these barriers. Sometimes the applicability refers to the equipment set-up, such as the minimum number of measuring channels that are operable.
(c) The specification(s) of the operational limit and condition; for example, the value that may not be exceeded, or specific conditions on equipment.
(d) The bases for these topics, in particular for the adopted specifications. These are normally the design calculations or safety calculations included in the safety analysis, which allow for margins in engineering and measuring uncertainties. However, these bases are sometimes simple conservative assumptions from previous operational experience, or they are based on the results of proposed experiments.

Safety limits

A.17.4. The safety limits for important process variables or parameters should be stated and justified by the analyses provided in the safety analysis report. Safety

limits normally involve operational parameters such as fuel temperatures, fuel cladding temperatures, reactor coolant temperature, reactor pressure, reactor power, coolant flow rates and, for pool reactors, the water level above the core. These safety limits are derived primarily from Chapters 5 and 16 of the safety analysis report.

Safety system settings

A.17.5. Safety system settings should be provided for those process variables and parameters that, if not controlled, could result in a safety limit being exceeded. This section should identify the safety system settings and should provide an analysis showing that the safety limits will not be exceeded. In determining safety system settings, consideration should be given to items such as calibration error, possible inaccuracies in measurement and system response times. Safety system settings are derived primarily from Chapters 5 and 16 of the safety analysis report.

Limiting conditions for safe operation

A.17.6. This section should present the limiting conditions for safe operation, which should provide acceptable margins between normal operating values and safety system settings. In many cases the limiting conditions that are established by the operating organization set constraints on equipment and operational characteristics. These constraints are identified in the safety analysis report as being important to safety and should be adhered to during operation of the research reactor. In some cases, when process variables or parameters reach a limiting condition for safe operation, they may initiate alarms to enable the operating personnel to take appropriate action to prevent safety system settings from being exceeded. Some examples of limiting conditions for safe operation are as follows:

— Core configurations and design limitations (e.g. reactivity coefficients, burnup limits, minimum and maximum number of the fuel elements and reflector elements, their geometrical arrangements, inspection);
— Minimum number, design and performance of reactivity control mechanisms;
— Fuel design parameters (e.g. enrichment, fuel type, cladding type);
— Maximum reactivity insertion rate;
— Minimum operational measurement systems and control systems for the reactor and safety set points;
— Equipment required to provide confinement or containment;

— Operations that require means of confinement or containment;
— Minimum operating equipment for ventilation systems;
— Equipment and performance of the emergency power supply systems;
— Minimum operational equipment for radiation monitoring systems and effluent monitoring systems, and their safety set points for the different operational stages (e.g. shutdown, operation, fuel handling);
— Limits on effluent releases;
— Limitations on experiments (e.g. reactivity, materials);
— Other design limitations important to safety.

Surveillance requirements

A.17.7. This section should discuss the surveillance requirements regarding the frequency and scope of tests, showing that the performance levels set by the safety limits and the limiting conditions for safe operation are being met. The requirements for monitoring, inspection, operability checks and calibrations should be included, and the actions to be taken if a system fails should be described. The conditions for continuing operation during repair work or the acceptability of the substitution of replacement equipment for failed equipment should be stated. Guidance is presented in paras 3.27–3.32 of Ref. [19].

Administrative requirements

A.17.8. This section should contain the administrative and organizational requirements, as well as the organizational structure and responsibilities, the staffing requirements, the review and audit of research reactor operating procedures, the review of operational events, reports and records, and the radiation protection area classifications. These limiting conditions and administrative requirements are derived primarily from Chapter 13 of the safety analysis report.

CHAPTER 18: MANAGEMENT SYSTEMS

A.18.1. The IAEA safety standards use the term ‘management system’ rather than ‘quality assurance’. The concept of ‘management system’ reflects and includes the initial concepts of ‘quality assurance’ and ‘quality control’ (controlling the quality of products) and reflects its evolution through ‘quality assurance’ (the system for ensuring the quality of products) and ‘quality management’ (the system for managing quality). The management system is an integrated set of interrelated or interacting elements that establishes policies and

objectives, and that enables those objectives to be achieved in a safe, efficient and effective manner.[32]

A.18.2. The operating organization is responsible for the developmend and use of a management system that will ensure conformance with the requirements for every aspect of safety. The objectives and scope of the management system should be established in accordance with the requirements of Ref. [2] and with national standards.

A.18.3. This section should describe the management system or should refer to a description of it. A summary should be provided of the items, services and processes to which the management system should apply, and of the organizational structure within which the activities are to be planned and implemented. The level of control and verification of quality should also be defined, and the means available for achieving this level should be described.

A.18.4. This section should describe or should refer to the particular parts of the management system that have been established for the phases of design, procurement, construction, commissioning or operation, as appropriate. The management system procedures should be consistent with the requirements of the research reactor project and its objectives, status and characteristics, and the management system should be acceptable to the regulatory body.

Management system procedures

A.18.5. This section should describe or refer to the planning, implementation and control of essential activities relating to the management system procedures to ensure that the specific requirements — such as regulatory requirements, design and construction criteria, and acceptance criteria — are correctly applied and fulfilled. In particular, the responsibilities and authorities of the personnel concerned under the management system should be specified.

A.18.6. This section should describe the procedures covering specific activities under the management system, such as resolution of non-conformances, design changes, design deviations and concessions, and the analysis of their impacts on safety requirements.

[32] In Refs [11, 12], requirements are established and guidance is provided on the management system.

A.18.7. This section should describe the procedures covering the operating activities performed under the management system. Examples are activities relating to reactivity management and criticality management, thermal safety of the core, safety of experimental devices, reactor modifications, procurement and storage of components and materials, manipulations of core elements and experimental facilities, and human surveillance.

A.18.8. This section should describe how the safety analysis report and supporting documents are identified and filed, and how long the documents are retained, or a reference to such a description should be given.

CHAPTER 19: DECOMMISSIONING

A.19.1. This chapter of the safety analysis report should provide information on the design provisions and the operational procedures to facilitate the decommissioning process. The design basis relating to decommissioning should be described.

A.19.2. Those aspects of the research reactor design that facilitate decommissioning should be discussed, such as selection of materials to reduce activation and to provide for easy decontamination, detachment and handling (remotely where required) of activated components, and adequate facilities for the processing of radioactive waste.

A.19.3. This chapter should discuss the aspects of research reactor operation that facilitate decommissioning, such as operational practices to reduce activation of material and maintenance of records of the construction and contamination of the research reactor. The safety analysis report should provide evidence that modifications will not have an adverse impact on the decommissioning of the research reactor.

CHAPTER 20: EMERGENCY PLANNING AND PREPAREDNESS

Emergency plan

A.20.1. This section of the safety analysis report should contain or refer to an emergency plan, which will provide reasonable assurance that actions can and will be taken in response to a nuclear or radiological emergency that might occur at the research reactor. However, safety precautions taken in the design and

operation of the reactor will greatly reduce the possibility of an accident. Requirements for the emergency plan are established in Ref. [2].

A.20.2. This section should demonstrate that the emergency plan is based on accidents analysed in the safety analysis report.

A.20.3. This section should provide information on actions to be taken in the reactor building, on the site and off the site. Since the off-site emergency plan is required to be established in cooperation with the responsible authorities, the emergency measures that are to be taken off the site could be presented in a separate plan and referenced in this section. The information should cover the following items:

(a) The emergency response arrangements, giving clear instructions regarding authorities and responsibilities;
(b) The process for identifying and classifying an emergency;
(c) The agreements made with off-site agencies that will help in an emergency;
(d) Notification of on-site personnel and, if necessary, off-site personnel;
(e) Notification of government authorities and local authorities;
(f) Reliability of communications between the emergency control room, if available, and outside locations;
(g) Protective actions;
(h) Equipment items available to deal with an emergency and their location;
(i) Arrangements with medical facilities to treat contaminated victims;
(j) Training of personnel;
(k) Frequency and scope of exercises and drills;
(l) Adequacy of resources to implement the emergency plan.

Emergency procedures[33]

A.20.4. This section should demonstrate that the emergency plan will be implemented by means of emergency procedures. The emergency procedures should include the specific actions that will be taken to mitigate the consequences of a nuclear or radiological emergency.

A.20.5. This section should contain information on the arrangements for periodic review of the emergency plan, the emergency procedures and their

[33] Guidance is provided in paras 5.53–5.56 of Ref. [19].

implementation, to ensure that the requirements of new experiments or research reactor modifications are included.

A.20.6. The emergency procedures should contain guidance on limits to doses to emergency workers performing rescue missions or taking protective actions.

REFERENCES

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

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors, IAEA Safety Standards Series No. SSG-22, IAEA, Vienna (2012).

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Annex I

APPROACH TO AND METHODS OF SAFETY ANALYSIS

I–1. This annex presents some considerations for developing a safety analysis for a research reactor. The well accepted basic approach to developing a safety analysis is to consider initiating events for credible accidents, using a deterministic[1] method to estimate the maximum possible releases to the environment. Probabilistic methods may be used to evaluate which accident sequences are of a higher likelihood; they will also be useful for evaluating relative rankings of risks, and hence for determining countermeasures. They may also be used for identifying any latent weaknesses in the design and for quantifying the value of possible improvements or modifications. However, probabilistic safety assessment is not treated in this Safety Guide, and consequently only deterministic methods are discussed here.[2]

I–2. These considerations cover a wide spectrum of research reactors and thus may contain information that is not applicable to all research reactors but that is provided for additional guidance.

METHODS FOR IDENTIFICATION AND SELECTION OF INITIATING EVENTS

I–3. Postulated initiating events are possible occurrences that may lead to reactor fault sequences or to accident scenarios. They might originate from component failures, system malfunctions, human errors or external events and particular internal events.

I–4. The method used to identify postulated initiating events and to select sets of particular events for further analysis has to be established. This method has to ensure that the list of initiating events is as complete as possible, that initiating events are grouped in a logical fashion to simplify the analysis, and that limiting or bounding initiating events in each group are selected for further analysis. Such a method could include one or more of the following:

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[1] More detailed information on the development of deterministic safety analysis can be found in Ref. [I–1].

[2] For further information on applications of probabilistic safety assessment to research reactors, see Refs [I–2, I–3]. More detailed information on the development of probabilistic safety analysis can be found in Refs [I–4, I–5].

(a) Lists of initiating events in research reactors. A list of possible initiating events in research reactors is given in para. 3.17 of this publication.
(b) Engineering evaluation. Potential sources of radiation and types of radiological hazard within the research reactor are identified, and a systematic review of the research reactor design, operations and site factors is made to identify occurrences that could lead to radiological hazards.
(c) Operational experience. Past experience from the research reactor or from similar facilities, including experience derived from the examination of safety reports and the IAEA's Incident Reporting System for Research Reactors (IRSRR) database, can be used to develop or to supplement the list of initiating events.
(d) Logical analysis. An example is a top-down logical model known as a master logic diagram, which is similar to a fault tree.

I–5. Methods used to reject particular initiating events and to exclude them from further analysis need to be determined and justified. Such methods could lead to rejection of the following initiating events:

(a) Incredible initiating events. Initiating events that are not possible for the research reactor under study.
(b) Very rare initiating events. Initiating events whose frequency of occurrence may be so low that they could be candidates for rejection on a probabilistic basis (e.g. aircraft crashes) using statistical data or conservative estimates. Combinations of mutually independent initiating events, each having a low frequency of occurrence, would also fall into this category.

I–6. Certain methods can be used to group initiating events as follows:

(a) Initiating events that require similar safety functions, which determine the design parameters of the safety systems;
(b) Initiating events that have a similar influence on reactor behaviour or on structures, systems or components, for which similar calculational models are used;
(c) Initiating events that can assist in the selection of limiting cases for analysis in each group;
(d) External initiating events that have the potential for a common cause impact on the research reactor.

One possible grouping is shown in para A.16.11 of the Appendix to this publication.

I–7. To simplify the analyses for each group of postulated initiating events, a method could be used to select for further analysis those limiting initiating events that are limiting for all other initiating events in the group.

METHODS FOR EVENT SEQUENCE ANALYSIS

I–8. A clearly defined method will facilitate the evaluation of the step by step sequence of events, from the initiation of the event to the final stabilized condition. The rules or conventions regarding the extent to which reactor systems, including the reactor protection system, are assumed to function are the basis for this method. If there is a possibility of fuel cladding failure, then other barriers to prevent the spread of radioactive material have to be considered, not only if all systems function correctly but also if some of them fail. Consideration has to be given to the types of event that will be evaluated by using this method, and the types of event that will be evaluated by other methods (see paras I–15–I–19).

I–9. The sequences have to include the response of the reactor and the reactor systems, as well as human interactions. Possible sequences for the case in which a system fails need to be described in detail. The following points need to be considered:

(a) Use of structured techniques, such as event trees or event sequence diagrams;
(b) Identification of significant occurrences on a timescale, for example, flux monitor trip and start of insertion of control rods;
(c) Indication of correct and incorrect functioning of normally operating reactor instrumentation and controls;
(d) Evaluation of the three principal safety functions (shutting down the reactor, cooling the fuel and maintaining confinement of radioactive material), including an indication of both the correct functioning of reactor protection and safety systems and their possible failure;
(e) Required operator actions;
(f) Frequency or probability evaluations to be carried out in assessing the sequence of events;
(g) Conditions for termination of the analysis, including, for example, situations in which stable conditions are reached (no exposures or releases), or if the likelihood of the sequence becomes so low that further analysis is not warranted, or if all levels of defence against the initiating event are exceeded and the sequence leads to significant exposure of personnel or to the release of radioactive material.

I–10. Rules or conventions have to be established to determine the response of reactor systems. These rules or conventions need to refer to:

(a) The effect of single, random failures;
(b) System qualification (or lack of qualification) under accident conditions;
(c) Safety and protection systems, including reliability in quantitative terms, if applicable;
(d) Support systems, such as normal and emergency electric power and cooling;
(e) Redundant trip parameters;
(f) Actions of systems that are independent;
(g) Operator action (e.g. response time, display of information on a console);
(h) Carrying out of frequency or probability evaluations to assess the system response, the extent to which such evaluations will be used and the methods to be employed (including validation).

I–11. Rules or conventions have to be developed to determine those event sequences that are beyond the design basis and thus excluded from further analysis. Such rules could be based on:

(a) Qualitative arguments justifying the exclusion of events whose occurrence is impossible, or events that are considered not to be credible for the research reactor under study;
(b) Qualification of the research reactor or research reactor systems against the effects of the event; or
(c) Quantitative frequency or probability arguments.

I–12. The effects of dependent failures (e.g. common cause or cross-linked effects) and human error that have to be considered include:

(a) Investigations carried out to identify the specific causes of dependent failures or human error;
(b) Evaluation of the effect of human error on either initiating an accident or worsening the development of accident sequences;
(c) Assessments of the validity of any assumptions or rules concerning the response of research reactor systems during accident sequences.

I–13. The frequency or probability of event sequences may be evaluated; this would help to determine which sequences could be excluded from the design basis or to assess the relative risk presented by various sequences. This evaluation includes:

(a) The known or estimated frequency of the initiating event, for example, loss of electrical power supply and failure of a pump or rupture of pipe work.
(b) Methods for estimating the probability of failure of each of various safety or safety support systems.
(c) Rules regarding the subdivision of event sequences to avoid (or to accommodate) an arbitrary subdivision at the systems level, as well as an arbitrary subdivision of initiating events (e.g. a set of similar pipe breaks rather than the generic event, specific meteorology) that can lead to many similar event sequences and that may have a low cumulative probability.
(d) Conventions for determining the likelihood of event sequences, with due regard to the effects of a dependent failure. For example, the probability of a safety function loss might be determined as the product of the failure probability of the associated systems and the cumulative probability of similar initiating events if these systems and events are independent.

I–14. Limiting or bounding event sequences in each class could be selected for further analysis, to reduce the number of events to be analysed using analytical methods of core transients. Consideration is to be given to:

(a) Conservative assumptions made in the classification of events to provide a safety margin (e.g. uncertainty allowances and not taking full credit for mitigating actions of systems or of operator response) or to ensure that all sequences in a class have been covered, starting from all permitted states in the operating envelope;
(b) The methods used to choose bounding sequences in a group of events, which represent the entire class and not just specific sequences, including those sequences that have the most severe consequences.

METHODS FOR EVALUATION OF EXTERNAL EVENTS AND SPECIAL INTERNAL EVENTS

I–15. General methods used to evaluate particular external and internal events, such as earthquakes, tornadoes or a sudden, catastrophic rupture of reactor pressure retaining components or reactor internals, are presented in the appropriate chapter of the safety analysis report. It may be difficult to model the effects of such events, or analyses may be highly speculative. Further guidance on protection against such events is given in Chapters 2 and 3 of the safety analysis report as set out in the Appendix to this Safety Guide.

I–16. In general, design qualification is an accepted practice for protection against external events once siting questions have been resolved (i.e. if the site

does not present hazards for which there is no adequate protection). The method for establishing the design bases for particular external phenomena can be summarized as follows:

(a) The potential of an event at the research reactor site for each phenomenon is assessed. If such a potential exists, historical data are evaluated to determine both the intensity and the frequency of occurrence of the phenomenon.
(b) The relevant physical parameters associated with the different degrees of severity of each external phenomenon are identified.
(c) A relationship between the severity of the phenomenon and the frequency of occurrence is determined, or a model appropriate to the phenomenon in the site region is constructed.
(d) A particular design basis frequency of occurrence is established (the defined recurrence frequency, often in the range of 10^{-3} per year) for which protection is provided to preserve essential safety related structures, systems and components.
(e) The design basis parameters for the phenomenon are evaluated, corresponding to the design basis frequency of occurrence.

I–17. Design qualification may prevent failure of pressure retaining components. In this case, the safety analysis report has to describe the design and construction standards used (e.g. acceptable engineering codes and practices) to prevent structural failures and to maintain the required safety functions. Reference may be made to the appropriate chapters of the safety analysis report (see Chapters 2 and 3 of the safety analysis report as set out in the Appendix to this Safety Guide).

Qualitative evaluations

I–18. Consideration has to be given to the conditions under which qualitative evaluations are used in the safety analysis to treat particular event sequences; for example:

(a) Treatment of fault sequences that are not limiting (e.g. they are bounded by other initiating events);
(b) Justification of design measures to prevent certain fault sequences or to demonstrate that the events would not be considered credible;
(c) Justification of administrative measures to reduce the probability of occurrence of faults.

I–19. Such qualitative arguments need to be used with caution and after consultation of the regulatory body concerning their acceptability.

ACCEPTANCE CRITERIA

I–20. The significant results of the safety analysis have to be compared with the acceptance criteria (see paras 2.14–2.18 of this Safety Guide).

I–21. Not only the acceptance criteria appropriate to the safety analysis but also the results of the comparisons referred to in para. I–20 are to be presented in the safety analysis report.

REFERENCES TO ANNEX I

[I–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Analysis for Research Reactors, Safety Reports Series No. 55, IAEA, Vienna (2008).

[I–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Manual on Reliability Data Collection for Research Reactor PSAs, IAEA-TECDOC-636, IAEA, Vienna (1992).

[I–3] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Component Reliability Data for Research Reactor PSA, IAEA-TECDOC-930, IAEA, Vienna (1997).

[I–4] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

[I–5] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

Annex II

EXAMPLES OF INPUT PARAMETERS AND INITIAL CONDITIONS

II–1. Examples of input parameters and initial conditions to be identified in the safety analysis are:

— Moderator (and coolant) temperature coefficient of reactivity;
— Moderator void coefficient of reactivity;
— Fuel temperature coefficient of reactivity;
— Effective prompt neutron lifetime;
— Delayed neutron fraction(s);
— Average heat flux;
— Maximum heat flux;
— Minimum departure from nucleate boiling ratio;
— Minimum critical heat flux ratio;
— Margin to onset of significant void;
— Margin to onset of flow instability;
— Axial power distribution;
— Radial power distribution;
— Hot channel factor;
— Core coolant flow rate;
— Core coolant inlet and exit temperatures;
— Core coolant inlet and exit pressures;
— Hot channel coolant exit temperature;
— Maximum fuel centre-line temperature;
— Fuel cladding temperature;
— Reactor coolant system inventory;
— Coolant level in reactor vessel or tank;
— Coolant level in the components (e.g. delay tank);
— Heat exchanger mass flow rate and temperature;
— Fuel burn-up (exit burn-up, ratio of peak to average burn-up);
— Control rod worth (differential and total, shutdown margin);
— Reactivity insertion rate in an emergency.

Annex III

ITEMS TO BE CONSIDERED IN THE DESCRIPTION OF THE RESEARCH REACTOR

SUMMARY DESCRIPTION

III–1. A brief description of the following aspects of the research reactor needs to be provided:

(a) Purpose of the research reactor (neutron source, irradiation facilities, material testing).
(b) Type of research reactor (pool, tank, etc.):
  — Type of fuel;
  — Moderator;
  — Reflector;
  — Core configurations (fuel elements, reflector elements, reactivity control mechanisms);
  — Reactivity control mechanisms for power regulation (control or shim rods);
  — Reactivity control mechanisms for shutdown (safety rods).
(c) Coolant.
(d) Mechanical reactor design:
  — Reactor vessel, reactor pool;
  — Core support structures;
  — Reactor bridge;
  — Beam tubes, in-core test facilities;
  — Natural circulation provisions (flapper valves, coolant gate, etc.).
(e) Shielding.
(f) Summary table of main design and performance characteristics:
  — Rated power;
  — Neutron flux;
  — Core coolant flow;
  — Core inlet and outlet temperatures;
  — Power density.

REACTOR STRUCTURES

III–2. A detailed description of the following items is required:

(a) Reactor pool and/or vessel;
(b) Core support, grid plate;
(c) Reactor bridge;
(d) Reflector;
(e) Shielding (including movable shielding);
(f) Supports for core instrumentation;
(g) Beam tubes;
(h) In-core test facilities;
(i) Provisions for natural circulation.

The description needs to include materials and dimensions, supported by drawings. The effects of corrosion, fatigue and neutron irradiation on the lifetime of safety related mechanical components need to be discussed.

REACTIVITY CONTROL SYSTEM, REACTOR SHUTDOWN SYSTEM

III–3. The function of the mechanical design and the electrical design is described here. The description includes the materials and dimensions, and is supported by drawings. The reactivity control mechanisms and their instrumentation, such as their position or status (coupled and/or decoupled), are presented, together with the insertion time and interlocks. The effects of corrosion, fatigue and neutron irradiation on the lifetime of the mechanical and electrical components are also discussed in this section. The safety related design parameters to be presented are:

— Speed of control rods;
— Insertion time of shutdown rods;
— Maximum number and heights of withdrawals of rods.

Measures to avoid ejection of the control rods and shutdown rods also need to be described.

FUEL ELEMENTS

III–4. The fuel used, including the uranium enrichment and the type of fuel, needs to be specified. The description of the fuel element, supported by drawings, and the main characteristics of the fuel elements are to be presented, such as:

(a) Thickness of cladding;
(b) Length of active zone;
(c) Width of coolant channel;
(d) Number of fuel plates and/or pins;
(e) Cladding material;
(f) Uranium loading.

If fuel elements are used that contain channels for the movement of neutron absorbing blades or neutron absorbing rods, they are to be described in the same section. A summary of the experience with the fuel is a part of the section regarding the fuel elements.

REACTIVITY CONTROL SYSTEMS

III–5. In addition to the description of the reactivity control systems, supported by drawings, the main dimensions and information on the neutron absorber material used and on the experience with these or with similar reactivity control systems need to be provided.

Annex IV

TYPICAL RADIATION SOURCES AND RADIATION FIELDS IN A RESEARCH REACTOR

IV–1. Examples of possible radiation sources or radiation fields in a research reactor are:

— The fission product inventory of the reactor core;
— Spent fuel storage;
— Concentration of fission products, activation products and corrosion products in the pool or the coolant system and in related systems such as the purification system;
— Equipment, systems and piping containing activation sources;
— Solid and liquid radioactive waste and radioactive waste management facilities, and leakage or spills from these facilities;
— Gaseous radioactive material from the pool, coolant systems, cover gas systems, reflector systems and experimental facilities connected to ventilation systems, or any leakage from these systems;
— Filters from the ventilation systems;
— Airborne radioactive material in areas normally occupied by personnel;
— Experimental facilities with the potential to generate activated material or other radioactive material, or facilities for the storage and handling of such material, including sample activation and/or irradiation facilities, in-core experiments and hot cells;
— Material irradiated by the research reactor;
— Neutron startup sources;
— Sources for testing and calibration of radiation monitoring equipment.

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