DS396 Draft 8

Date: June 2010

IAEA SAFETY STANDARDS

for protecting people and the environment

Status: **Member States comments processed**
NUSSC processed

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

DRAFT SAFETY GUIDE
DS396

New Safety Guide

IAEA
International Atomic Energy Agency

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide was developed under the IAEA programme for safety standards for research reactors, 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. Seven of these principles[1] are directly addressed in this Safety Guide. Guidance is also given which analysis, acceptance criteria, verifications and evaluations should be performed to prove that the safety objectives will be met to fulfill 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 the IAEA Safety Requirements on the Safety of Research Reactors [2].

1.2. This publication supersedes Safety Series No. 35-G1[2]. The main changes and adaptations were related to the consistency with NS-R-4, the other recently published safety guides for research reactors and other relevant Safety Standards. Where applicable references to the safety standards were incorporated.

1.3. Owing to the particular characteristics of research reactors, the safety aspects related to all stages in the lifetime of a research reactor have been given special emphasis and have been incorporated in Ref. [2]. These characteristics include the large variety of designs, the wide range of powers, the different modes of operation and purposes of utilization, the particularities of siting and differences among organizations operating research reactors, in particular, concerning their resources. These characteristics require a graded approach ([2], paras 1.11-

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

1.14) 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 the general public, the site personnel and the environment have a number of responsibilities which are interrelated. Most important are the performance of the safety analysis by the operating organization and its review and assessment by the regulatory body, as well as the production, submission and evaluation of other safety related documents during the licensing process or in other special circumstances, such as modifications and experiments. The present Safety Guide develops the general concepts in these areas which are presented in Ref. [2] and therefore, this Safety Guide should be read in conjunction with it. Furthermore, this publication considers 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 modifications, which are also discussed in other publications.

1.5. The use of the terms safety assessment, safety analysis, and review and assessment in this Guide requires explanation. Safety assessment is used synonymously with safety evaluation, i.e. a process to evaluate the safety of the reactor. Therefore, the safety assessment of the reactor may include many and varied activities during the licensing process, with iteration between the design and confirmatory analytical activities, such as analyses, preparation and submission of documents for review, and may involve several organizations. The safety assessment should continue during all the stages of the reactor lifetime and according to the potential magnitude and nature of the hazard associated with the particular research reactor or activity. The term safety assessment is in general used in the Safety Standards for research reactors instead of safety analysis, which in this Guide has a more specific meaning. Safety analysis is the analysis of incidents or hazards or risks. 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. Therefore, the term review and assessment is specifically used in this Safety Guide in the frame of the licence application to describe the regulator's responsibility of assessing safety documentation (requirements 25 and 26 of Ref. [24]).

[3] The terms ‘safety specifications’ or ‘technical specifications (tech specs) for safe operation’ and ‘general operating rules’ are sometimes used in Member States These terms are used by operating organizations and regulatory bodies for nuclear reactors. These expressions usually cover safety limits, safety system settings, limiting conditions for safe operation, and surveillance requirements and administrative requirements.

1.6. This Safety Guide provides guidance to the operating organizations in carrying out independent verification of the safety assessment of a new research reactor with new or already existing design. Although this Safety Guide provides guidance to the operating organization it is also suitable for designers performing a safety assessment of a new or existing research reactor.

OBJECTIVE

1.7. The objective of this Safety Guide is to provide recommendations to meet the requirements for the safety assessment in the licensing process, such as responsibilities and functions of the organizations involved during the licensing process ([2], paras 3.2-3.3 and 4.1-4.4) and the steps towards the issue of the licence ([2], paras 3.4-3.5) and the requirements for conducting the safety assessment of facilities and activities [27]. This Safety Guide also provides recommendations on the performance of the safety analysis ([2], paras 6.72-6.78) and the preparation of the safety analysis report ([2], paras 3.6-3.10). Guidance is also given which analysis, verification and evaluation should be performed to prove that the safety objectives will be met to fulfill the safety requirements for the Operating Organization. Finally, recommendations are also provided which 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 design modifications, as well as for the independent verification of the safety assessment of new research reactors with a new or already existing design. The guidelines are also applicable to a revised and updated safety assessment of an existing reactor.

1.9. This Safety Guide provides only recommendations in relation to the utilization[4] (experiments or experimental facilities) in direct relation with the safety analyses of the safety analyses report for the reactor. Guidance on the safety analyses for experiments and experimental facilities is provided in [5].

SCOPE

1.10. The recommendations provided in this Safety Guide are applicable to any type of research reactor. However, the particular characteristics of research reactors, as set out in para. 1.3, require that a graded approach ([2], paras 1.11-1.14) is applied in implementing the

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

recommendations in this Safety Guide. Therefore, it is 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 discussions and agreements between the operating organization and the regulatory body, with the final decision to be made by the latter.

1.11. The Safety Guide focuses mainly on research reactors of capacity of up to a few tens of megawatts. Therefore, the amount of detail required in the safety analysis report for small research reactors (below a few tens of kilowatts) and critical assemblies may be substantially reduced. Nevertheless, when using the graded approach all items included in this Safety Guide should be assessed. On the other hand, additional guidance for the safety analyses, preparation of the safety analyses report and licensing process of high powered or otherwise advanced or complex research reactors is provided in the IAEA safety publications for power reactors[5]. The use of the safety publication for power reactors requires also that a graded approach ([2], paras 1.11-1.14) is applied in implementing the recommendations based on the potential hazard of the research reactor.

1.12. Although this Safety Guide focuses mainly on newly designed and constructed research reactors, its content is applicable to any re-licensing process or reassessment for the 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[6] 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 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 of the research reactor and its utilisation. The principle of a graded approach should be used ([2], paras 1.11. – 1.14.) applying the recommendations and guidance of this 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 should be taken into account from the design stage and an appropriate compatibility should be achieved. However security aspects are subject to confidentiality requirements and are not discussed in this Safety Guide, see also A13.12 and A.13.13.

[5] Further guidance on the preparation of safety analysis for research reactors with bigger potential hazard can be found Refs [3] and [4].
[6] Guidance on decommissioning activities is provided in ref [15].

STRUCTURE

1.14. 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 toward the licensing of a research reactor. The main reason for presenting these two topics together in a single Safety Guide is their interrelationship and joint importance in the licensing process.

1.15. Section 2 describes the licensing process by which the safety of the research reactor and the issue of licences is controlled and determined.

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

1.17. Section 4 provides general recommendations on the conduct of the review and assessment of the safety of the research reactor by the regulatory body.

1.18. The Appendix is a comprehensive guide on the preparation of a safety analysis report for a research reactor having the characteristics discussed in paras 1.3 and 1.9-1.12. It gives recommendations for the standard content of the safety analysis report.

1.19. Annexes I and II outline, and give information on, application of a basic approach to performing the safety analysis of a research reactor by using mainly deterministic methods[7] to analyse accidents, including their radiological consequences. Annex III deals with specific aspects of the reactor to be described in its safety analysis report. Finally, Annex IV gives a list of typical radioactive sources to be considered in a research reactor and described in the safety analysis report.

[7] Annex I deals mainly with deterministic methods which are normally used for safety evaluations of research reactors. Probabilistic techniques could be used to supplement the above mentioned evaluations, see also paras 3.27 and 3.28.

2. SAFETY ASSESSMENT IN THE LICENSING PROCESS

RESPONSIBILITIES

2.1. In accordance with Principle 1 ([1], para. 2), it should be assured that for a research reactor to be built (or to undergo a major modification), the highest safety standards which can reasonably be achieved should be met to protect the people and environment around the site where it operates. This assurance comes from 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 ([24], requirements 1 and 2). The establishment of an independent regulatory body is an important requirement of an adequate legal and regulatory framework. The IAEA Safety Requirements for Research Reactors [2] establishes general requirements for the framework of the system for ensuring safety, including the licensing process. Further guidance on the development of a governmental regulatory body has been provided in [24]. These requirements should be met 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 keeps the prime responsibility to demonstrate to the satisfaction of the regulatory body that this responsibility has been and will continue to be adequately discharged. The prime responsibility for safety can not be delegated. One of the ways in which the operating organization demonstrates that it has achieved adequate safety is through the information normally incorporated in a safety analysis report. This information also constitutes the prime basis for the regulatory decision on licensing the research reactor. Close liaison should be maintained between the regulatory body and the operating organization throughout the entire process of regulatory supervision of the research reactor.

2.3. The content of the application of a licence should be based on the legal and regulatory system of the Member State[8]. The information provided in support of a licence application should be commensurate with the potential magnitude of the hazard associated with the research reactor and its utilization, and 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 from regulatory control of the research reactor. The licensing process should be understood by interested parties and

[8] Further guidance on the subject can be found in Ref. [24]

predictable (well-defined, clear, transparent and traceable). The different stages of a licensing process should be established in a coherent yet flexible way in order to achieve most efficiency. These stages should be discrete and in a logical order ([26], 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 "pre-licensing" process, e.g. steps that provide for early approval of siting and approval of the safety concept and design and the issue of a construction licence. Such a licensing regime may help minimize duplication of effort through different stages and may allow for some stages to be conducted in parallel, provides for clear division of responsibilities for different stages between regulators, vendors and operating organization, gives the public opportunities for early participation and ensures that the most important safety issues are dealt upfront in such a "pre-licensing" phase. A detailed demonstration of nuclear safety, which includes adequate safety analysis, should be submitted by the operating organization, and reviewed and assessed by the regulatory body before the next stage is authorized. Detailed guidance on the licensing process is presented in [26].

2.7. At all stages, the operating organization should be able to demonstrate that it is in control of the research reactor and has an adequate organizational structure, management system[10], and adequate resources to discharge its obligations and, as appropriate, its liabilities. The totality of the documentation which the operating organization uses in making this demonstration, some of which may not be in the initial formal submission, should cover all appropriate topics, depending on the stage of the licensing process.

[9] Utilization and modification of research reactors, although processes comprising activities normally included in operation ([1], paras 7.87 –7.94), may be considered separate stages in the licensing process because their safety implications lead to a large number of review and assessment activities which are repeated many times during the reactor's lifetime (see paras 2.48-2.51 of the present Safety Guide).

10 The term management system reflects and includes the concept 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 which enables those objectives to be achieved in a safe, efficient and effective manner. Further guidance is given in Refs [6] and [7].

2.8. Based on the requirements of a Governmental, Legal and Regulatory Framework regarding review and assessment of safety related documentation [24] the operating organization should submit timely to the regulatory body 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 is also the responsibility of the operating organization to keep the regulatory body informed of any new information and alterations to previously submitted information.

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

2.10. The review and assessment of the 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 might be agreed upon 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 (or having an indirect but significant influence on safety related aspects) as appropriate. The revised documentation should be submitted to the regulatory body to allow for review and assessment ([24], para 4.45) with the potential magnitude and nature of the associated hazard being taken into account ([24], requirement 26).

2.12. The operating organization should submit information to the regulatory body on which the regulatory body could determine whether the proposed research reactor can be sited, designed, constructed, commissioned, operated, utilized, modified and decommissioned without undue radiological risk to the site personnel, the general public and the environment. Based on the documentation submitted the regulatory body should be able to:

(a) Acquire an understanding of the reactor design, the safety concept on which it is based, the management system and the operating principles proposed by the operating organization.

(b) Perform a review and assessment of the operating organization's technical submissions. This review and assessment proceeds from an overall survey of the reactor to an in-depth review and assessment of the design of individual systems, structures and components and their behaviour during normal operation, anticipated operational occurrences and accident conditions.

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

2.13. The primary basis for the review and assessment of the safety aspects of the proposed 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 (para 4.2);
- Whether this information is in compliance with the requirements of all applicable rules and regulations;
- Whether this information is accurate; this may be done by independent checks of the design, including calculations, and by inspections of the programmes and facilities (e.g. design- and review programmes, management system requirements and implementation);

ACCEPTANCE CRITERIA[11]

2.14. In addition to the acceptance criteria established within the regulatory framework, the operating organization should develop acceptance criteria to demonstrate an adequate application of the principles for safe design and operation embodied in the IAEA Safety Standards. These principles include the radiation protection objectives as stated in para 2.2 of [2], which refer to the IAEA Basic Standards for Radiation Protection and Recommendations of International Commission on Radiological Protection.[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 reactor and the accident conditions considered in the design of the research reactor. They may be:

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

[11] Practical examples of acceptance criteria are provided in Ref. [19].

[12] See International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources [22], and ICRP publication 103 [23], for the radiation protection principles. Additional guidance for the radiation protection is also given in Ref. [18].

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 the operating organization. The set of acceptance criteria should be satisfactory to the regulatory body.

2.17. During the development of the acceptance criteria considerations should be given to those listed below:

(a) Radiological criteria such as:
- Maximum allowed doses to the public;
- Dose limits (or targets) for staff of the operating organization, including experimenters, workers at the reactor site;
- Dose limits for intervention during accident conditions to perform life saving and mitigating actions;
- Release limits to the environment both during normal operations and during accident conditions; and
- Risk criteria (where applicable).

(b) Fuel performance criteria
- Maximum cladding temperature below blistering temperature;
- Maximum heat flux not exceeding critical heat flux (CHF) during a transient;
- Maximum heat flux not exceeding onset of significant voiding (OSV) during a transient;
- Flow conditions not exceeding onset of flow instability (OFI);
- Frequency limits for significant fuel cladding damage.

(c) Performance criteria, including:
- Limits on parameters to prevent damage of the primary coolant system boundary;
- Limits on parameters to prevent damage of safety relevant systems by in-core or out-core experimental facilities[13]
- Limits on parameters to prevent damage of the containment systems;
- Maintenance of core cooling; and
- 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 subsequently more serious condition of the research reactor without the occurrence of a further independent failure. Thus the anticipated

[13] Isotope production facility is regarded as an experimental facility

operational occurrence by itself should not generate a DBA, and such an accident by itself should not generate a beyond design basis accident.

- There should be no consequential loss of function of the safety systems needed to mitigate the consequences of an accident.
- Systems used for accident mitigation should be designed and constructed to withstand the maximum loads, stresses and environmental conditions for the accident analyzed.

INFORMATION REQUIREMENTS DURING THE STAGES OF THE LICENSING PROCESS

2.19. The operating organization should provide all relevant information on the safety of the research reactor. This information is normally presented in a safety analysis report and is described comprehensively in the Appendix of this Safety Guide. Guidance for the preparation and presentation of the safety analysis report is specified in Chapter 3 and guidance for its review and assessment is given in Chapter 4. The following paragraphs provide a summary of the information which is normally required for each of the stages of the licensing process. The sequential request of information may lead to successive updating, with each version of the report corresponding to a particular stage in the licensing process, as outlined in para 2.5.

2.20. The production 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 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 during the course of the design process (for example, before start of construction or operation) the status of the design should be described in a safety analysis report and should describe 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 which indicates the time-scale for the production of the different chapters of the safety analysis report. Since the approval of one stage is normally required before commencement to the next stage, it is important that the safety analysis report is

made available for review and assessment on a time-scale that has been agreed upon by the regulatory body. Some estimate of the size and scope of the analyses should be conveyed to the assessor. In this time-scale, 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 4.3-4.4).

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 that must 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 impact on the site personnel, on the population in the surroundings and on the environment should be provided. In some Member States an Environmental Impact Study should be performed as a part of the licensing process.

2.24. The characteristics of the site, which may affect the safety aspects of the research reactor, should be investigated and assessed by the operating organization. The objective of the assessment is to assess how the site characteristics will influence the design and operation of the research reactor and to demonstrate the adequacy of the characteristics of the site from the safety viewpoint. The requirements for the initial evaluation and selection of a site, the general criteria for site evaluation and the external events which have to be considered for site evaluation are given in Ref. ([2], chapter 5). Additional guidance on siting and site evaluation is provided in chapter 3 of the Appendix and in [25] and [26]. The details on siting which have to be addressed in the SAR are presented in Chapter 3 of the Appendix.

Design and construction

2.25. Before authorization of construction of the research reactor, features such as physical layout and the construction of the reactor and the key process elements should be carefully considered, and their effects on the safety of the research reactor throughout its lifetime should be assessed. In this stage due consideration should be given to the ageing[14] mechanisms of materials and SSCs and their effects on safety. The operating organization should describe the 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 used, radiation doses to be expected and amounts of waste to be produced should be prepared in the

[14] Additional guidance for ageing management for research reactors is given in Ref. [35]

design stage too. The information discussed in this para should be submitted to the regulatory body for review and assessment.

2.26. To obtain a construction licence or an agreement for the start of construction, the operating organization should submit 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 and process systems, and it should present the results of the safety analysis which demonstrate the adequacy of the design of safety related structures, systems and components. This information should be submitted in the form of a safety analysis report, which may be preliminary and subject to updating as the project proceeds.

2.27. Those aspects of the design, which should be submitted to the regulatory body for review and assessment before the design is finalized, should be identified in order that the activities proceed smoothly while the reactor is under construction. The information should be updated and submitted to the regulatory body as the detailed design and construction of the reactor proceed. In some cases revised versions of the documents will be sufficient; in other cases, technical supplements may be appropriate. Additional guidance for the licensing process for this stage is given in [26].

2.28. The safety analysis report is the major document provided at this stage for review and assessment by the regulatory body, which will authorize the progress of detailed design and construction.

Commissioning

2.29. When the construction is in 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 submitted to the regulatory body for review and assessment in order to obtain the required authorizations for commissioning.

2.30. Ref. ([2], para 7.46) requires that commissioning tests have to be arranged in functional groups and in a logical sequence. This sequence includes pre-operational tests, initial criticality tests, low power tests, tests to prove the shutdown capabilities, power ascension and full power tests. Every test sequence has to be completed successfully and the results should be submitted to the regulatory body for review and assessment. The test results should be approved by the appropriate management level before the following test sequence should be started. The

commissioning programme should therefore be divided into stages which are usually arranged according to the following sequences:

– Stage A: tests prior to fuel loading;

– Stage B: fuel loading tests, initial criticality tests, 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, which has been reviewed and assessed by the regulatory body. Detailed guidance for the commissioning of research reactors is given in [12] and [26].

2.31. The updated revised safety analysis report should include the commissioning programme and demonstrate its adequacy ([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 above mentioned version of the safety analysis report.

2.32. The commissioning procedures for a commissioning stage should be reviewed before the start of the next stage and updated where necessary. The 'as-built' situation and the results of the previous commission stages should be taken into account. The updated commissioning procedures should be submitted to the regulatory body for review and assessment in order 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 this 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 the 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; radiation protection programme, emergency preparedness; the operational limits and conditions for commissioning; and the preliminary

operating procedures[15] should be taken into account in 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 the Stage C commissioning "Power ascension tests and full power tests" processes move closer to completion, this stage should concentrate on how the research reactor will be operated, utilized and maintained, and on procedures for controlling and monitoring operation and 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, modifications of 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 the paras 2.30 – 2.35 should be updated after each stage and submission to the regulatory body will form the basis for the start of the next commissioning stage as a part of the licensing process.

.

Operation

2.37. In the application for an operating licence, the operating organization should submit all of the information referred to in the preceding sections. To prove safe operation additional information should be submitted to the regulatory body. Some of this information is required during the licensing steps and some information should be submitted after the formal licence is obtained. Additional guidance for the licensing steps is given in [26] and detailed guidance to assure safe operation is given in Ref. [2] and the related Safety Guides [8], [11] and [18].

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

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 (periodic safety reviews (PSRs)), 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 depend on the potential risks the research reactor presents. For this review, a comparison of the existing safety analysis report with the operating experience should be made, including

[15] Guidance on operating procedures is given in Ref. [11]

accidents, radiological information, modifications, experiments and other aspects of the operation. If required as a result of the above mentioned review, 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 may affect reactor safety. The specific submission requirements will depend on the safety significance of the experiments and modifications. These requirements are described in para 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 [5] and [11].

2.41. Experiments and modifications having major safety significance should be subjected to design, construction, commissioning and safety analysis procedures equivalent to those for the 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 it is noted that the safety analysis report provides boundaries of operational limits and conditions which 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 submit them to the regulatory body for review and assessment in order to justify the safety of the proposed experiment or modification.

2.43. Commissioning of the experiment or the modified research reactor should take place to demonstrate compliance with the design intentions in the safety analysis report. In addition, if changes to the safety analysis report are made, it will be necessary to show that the original analysis is still valid.

Decommissioning and release from regulatory control

2.44. The decommissioning process, such that regulatory controls may be removed, includes decontamination and dismantling and/or removal of radioactive materials, radioactive waste,

components and structures should require agreement by the regulatory body. Detailed requirements on the subject can be found in ([2], paras 8.1.-8.8.) The operating organization should describe decommissioning process [16] to demonstrate that the 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 special hazards associated with the decommissioning process have been adequately analysed and assessed. Further details are provided in other relevant IAEA Safety Series.[17]

2.45. At some point in the decommissioning process (e.g. after removal of all fuel from the site) the safety analysis report ceases to be a major working document and a detailed decommissioning report should be prepared. Additional guidance on decommissioning is given in [28].

3. PREPARATION OF THE SAFETY ANALYSIS REPORT

PURPOSE AND SCOPE

3.1. The operating organization should make the arrangements for preparing a safety analysis report to demonstrate the safety of the design. The safety analysis report should also be the basis for the safe operation of the research reactor. The safety analysis report is 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 reactor design 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 design are consistent;
- To aid appreciation of the relevant design criteria, their limitations and requirements, and in the evaluation of the hazards posed by the research reactor;

[16] This documentation is required when initiating the decommissioning process. For the SAR different information on decommissioning should be included. (see paras A19.1.-A19.3. of the Appendix).
[17] See IAEA's Radioactive Waste Management Safety Standards.

- To aid training of operating personnel and familiarization with the research reactor;
- To aid the establishment of operational limits and conditions on certain parameters, which have to be met at all stages in the life of the reactor in order to ensure adequate protection of the margins of safety for the reactor;
- To identify ageing mechanisms and their effect on safety for the evolution 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, utilization and the related experimental facilities, and modification/upgrade of the research reactor;
- The consequences of events that may have occurred during the lifetime of the research reactor and which may impact on 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 safety significance. It should provide a detailed description of the general safety principles and criteria as well as codes and standards applied to the design for the protection of the reactor, the operating personnel, the general public and the environment. The potential hazards associated with the operation of the reactor should also be addressed in the safety analysis report. It should contain or refer to safety analysis of accident sequences and of the safety features incorporated in the design to avoid or to minimize accidents, or to mitigate their consequences through 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.–3.5. have been deliberately stressed, all topics treated in the Appendix of this Safety Guide should be adequately covered by the safety analysis report. All of these topics should be prepared in accordance with the corresponding recommendations in the Appendix. However, some of them may be discussed in separate documents (e.g. operational

limits and conditions, operational procedures, physical protection, emergency planning). In this case these topics are treated briefly in the safety analysis report, and a 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 ([9], para 3.13).

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 those performing the safety assessment. Personnel are considered independent if they have not participated in any part of the design and 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 would be carried out by or on behalf of the operating organization.

3.10. In some Member States the proposal and licence application for a research reactor may be subjected 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 which can be understood by the general public.

3.11. The safety analysis report should present adequate references which 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 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 to be provided upon request.

3.12. Some regulatory bodies request 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

[18] More guidance on public participation is given in paras 2.37-2.40 of Ref. [26]

right of individuals to privacy, in accordance with national law and regulations. Such confidential information should be made available as necessary without restriction to the regulatory body; that is, 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 inform the operating organization formally which consultants and advisors should be involved on behalf of the regulatory body. Those to whom such information is entrusted should be advised of its confidential nature and should be obliged, consistently with national law and regulations, to protect its confidentiality. If consultants, technical support organizations and external advisory committees should have confidential documents to their disposal a process to ensure confidentiality should be in place.

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

3.15. The reactor type, site and characteristics (design, power, and utilization) may influence the extent of the information that must be presented in the safety analysis report. Accident scenarios for reactors with higher power levels or significant inventory of radioactive material will usually require more details on the site and on the safety features to protect against any significant releases 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 of this 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 the operating personnel against overexposure in critical assembly facilities) may be much larger for small, low power facilities.

TABLE I. SELECTED POSTULATED INITIATING EVENTS

The following list of selected PIEs is based on the appendix to Ref. [2].

1. Loss of electrical power supplies
   - Loss of normal electrical power[19]
2. Insertion of excess reactivity
   - Criticality during fuel handling and loading (due to an error in fuel insertion)
   - Start-up 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 D_2O into H_2O systems);
   - Influence by experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material);
   - Insufficient shutdown margin;
   - Inadvertent ejections of control rods;
   - Maintenance errors with reactivity devices;
   - Spurious control system signals.
3. Loss of flow
   - Failure of primary pump;
   - Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger);
   - Influence of the failure or mishandling of an experiment;
   - Rupture of the primary coolant boundary leading to a loss of flow;
   - Fuel channel blockage;

[19] Although it is not an initiating event, consideration should be given to the loss of normal power followed by the loss of emergency power in order to ensure that the consequences would be acceptable under emergency conditions.

- Improper power distribution due, for example, to unbalanced rod positions, in-core experiments or fuel loading;
- Reduction in 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 pump or a system rupture).

4. Loss of coolant
   - Rupture of the primary coolant boundary;
   - Damaged pool;
   - Pump-down of the pool;
   - Failure of beam tubes or other penetrations.
5. Erroneous handling or malfunction of equipment or components
   - Failure of the cladding of a fuel element;
   - Mechanical damage to core or fuel (e.g. mishandling of fuel and dropping of a transfer flask onto the fuel);
   - Failure of an emergency cooling system;
   - Malfunction of the reactor power control;
   - Criticality in fuel in storage;
   - Failure of means of confinement, including the ventilation system;
   - Loss of coolant to fuel during transfer or storage;
   - Loss or reduction of proper shielding;
   - Failure of experimental facilities or materials (e.g. loop rupture);
   - Exceeding of fuel ratings.
6. Special internal events
   - Internal fires or explosions, including internally generated missiles;
   - Internal flooding;
   - Loss of support systems;
   - Drop of heavy loads
   - Loss of integrity of pressurized vessels
   - Malfunction in reactor experiment;
   - Improper access by persons to restricted areas;
   - Fluid jets and pipe whip;
   - Exothermic chemical reactions;
   - Electromagnetic compatibility
   - Security related incidents (see A.13.12 and A.13.13.

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;
- Tropical cyclones;
- Explosions;
- Aircraft crashes;
- Fires;
- Toxic spills;
- Accidents on transport routes;
- Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);
- Biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects;
- Extreme meteorological phenomena;
- Lightning strikes;
- Power or voltage surges on the external supply line;
- Security related external events (see A.13.12 and A.13.13).

8. Human errors

[20] The extreme weather conditions due to climate change should be taken into account for the determination of the external events

DEVELOPMENT OF THE SAFETY ANALYSIS

3.17. The safety analysis, part of the safety assessment used in 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 programme progresses so that the final safety analysis reflects the final design of the reactor as constructed.

3.18. The safety analysis is mainly used to enable the operating personnel to understand the basis for 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 and mitigation of their consequences. 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 base accidents and human error). The safety analysis also serves as a basis for the determination of the operational limits and conditions, as well as to design specifications for components and systems.

3.19. The consideration of fault conditions determines the design of the research reactor and the limits for the safety systems and for most SSCs needed for the operation of the research reactor. It will strongly influence the operational instructions and procedures that operating personnel should follow. In addition the potential radiological consequences for workers, the public and the environment of fault conditions may be more severe than those in routine operation. For this reason, an important part of the review and assessment effort should be directed to the safety analysis of fault conditions. It should be performed in accordance with the potential magnitude and nature of the risks associated with the particular research reactor. Safety analysis can be considered to consist of the following major steps:

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

3.20. The safety analysis should assess whether:

- Sufficient defence in depth has been provided and that the levels of defence are preserved in that the potential accident sequences are arrested as early as possible.
- The research reactor can withstand the physical and environmental conditions it would experience. This would include extremes of environmental and other conditions.
- Human factors and human performance issues have been adequately addressed.
- Long term ageing mechanisms that could detract from the reliability of the SSCs over the designed lifetime are identified, monitored and managed (i.e. by upgrade, refurbishment or replacement) so that safety is not affected and risk does not increase.

3.21. The identification and selection of the PIEs should be the first step in the safety analysis. The selection method used should be systematic and auditable as appropriate. Moreover, as complete as possible a listing of PIEs 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 PIEs 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.

3.22. PIEs 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 with identical system performance (such as in terms of timing, plant systems response and radiological release fractions).
- To allow for differing acceptance criteria for the safety analysis to be applied to differing event classes.

3.23. This process of event classification, in which initiating events of all types, both internal and external but also normal operation, shutdown and fuel loading, should be considered, should lead to a list of enveloping initiating events to be analysed. Failures in other systems, such as experimental facilities, the availability of off-site power or the total loss of off-site power, spent fuel storage and storage tanks for radioactive liquids should also be considered.

3.24. In the preparation of the set of postulated initiating events for the analysis, the list given in the Appendix of Ref. [2] should form the basis of the postulated initiating events to be considered. This list has been reproduced in Table I. Further detailed guidance on the methodology used is given in Annex I of this Guide. This Annex also gives guidance for the analysis of the event sequences triggered by the PIEs and for the analyses of external events and internal events. In particular, the analyses should clearly identify a number of assumed input parameters and initial conditions; these should be presented in the safety analysis report and will determine the basis for the determination of the operational limits and conditions. Annex II of this guide gives examples of these parameters.

3.25. 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 fulfil the intended objective, the detailed guidance for the preparation of the safety analysis as presented in Chapter 16 (Safety Analysis) of the Appendix of this Safety Guide should be taken into account.

3.26. The safety analysis should identify the design basis accidents (DBAs). In addition, accidents beyond the design basis with more severe consequences may be analysed for purposes of emergency planning and the measures to be defined and taken to mitigate the consequences of an accident.

3.27. Annex I deals mainly with deterministic methods which are normally used for safety evaluations 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 - to limit release of radioactive materials - can be achieved without performing complex calculations, because these methods tend to overestimate the amount of radioactive release. 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 siting 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.

3.28. Probabilistic techniques could be used to supplement the above mentioned evaluations. Probabilistic methodologies assume that all accidents are possible and that any number of simultaneous failures may occur, although the probabilities may be very low. Some accidents or accident combinations may have less dramatic consequences than those used in deterministic methodology but, when they are weighted by their likelihood, they may represent a real risk and

impose different demands on design. In addition, the deterministic approach has difficulties in treating effectively system interdependences (e.g. common cause failure) which the probabilistic methods can address analytically and quantitatively. Application of these 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. Detailed guidance has been developed in the IAEA-TECDOCs 636 [16] and 930 [17].

3.29. A typical PIE classification based on the initiating frequency, system recovery likelihood and potential consequences of an initiating event should be developed to determine the following:

(a) PIEs that are of high likelihood, which should be analysed to show that the research reactor has a robust tolerance for them owing to the provision of safety systems or an inherent behaviour tending (i) to restore the safe state, (ii) to prevent the release of radioactive material, (iii) to limit any such release to an acceptably low level.

(b) PIEs that are of low likelihood but that have severe potential 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.

PIEs which do not fall into these two groups, e.g. PIEs with a low likelihood and in principle low consequences, should also be evaluated to ensure that small deviations from the incident scenarios will not cause unacceptable risk (cliff edge effect) to the reactor or the environment.

3.30. The results of the safety analysis of the research reactor should be reflected in the safety analysis report, taking into account the guidance given in Chapter 16 (Safety Analysis) of the Appendix of the present Guide. Chapter 16 also provides guidance for the comparison of the results with the acceptance criteria to determine the acceptability of the reactor.

4. PERFORMANCE OF THE REVIEW AND ASSESSMENT

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 in order to enable the regulatory body to make a decision or 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 should include consideration of both normal operation and failures, and events, including human errors that have potential for causing exposure of workers or the public or radiological hazards to the environment. This safety analysis should be complete and 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 should include checks on the site and elsewhere to validate the claims made in the submissions. Operating organizations often have external peer reviews conducted at their facilities by national 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 information submitted by the operating organization in support of its licence application should be subjected to the review process of the regulatory body to determine whether the proposed research reactor can be sited, constructed, commissioned, operated, utilized and modified and decommissioned without undue radiological risk to the site personnel, the general public and the environment. Within this overall objective, specific objectives of the review and assessment are:

(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 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 assured, including the provisions made for accident conditions;

(g) To determine whether the utilization and modifications 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 THE REVIEW AND ASSESSMENT

4.3. The Operating Organization should discuss the programme for the review and assessment with the regulatory body. This programme, which should be established by the regulatory body, should take into account the stages in the licensing process as described in para 2.5 and paras. 2.22-2.43.

4.4. The programme 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 in 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 this acceptance review, an application or submission that is grossly deficient in certain areas may be returned to the operating organization for correction and re-submittal. ([10], Para 3.5)

4.6. A major feature of the submission of the operating organization will be its analysis of both the 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 is based on sound engineering and good management, and the safety analysis is a confirmation of the adequacy of these and not a substitute for them. The value of safety analysis is an extending knowledge and understanding of the research reactor and its behaviour and in identifying shortcomings in areas in which safety can be improved.

4.7. The documents, which 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 occupational and fire safety aspects;

(d) The management systems of the operating organization and its vendors;

(e) The design features related to physical protection which are important to safety;

(f) Information necessary for design verification.

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 authorizing 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, organization and programme for operation.

4.9. The documents, which should be submitted to the regulatory body for review and assessment in order to obtain authorizing for commissioning Stage B: “loading of nuclear fuel and initial criticality”, should include:

(a) The records of the results of previous commissioning step, including non-conformances and, when appropriate, their 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 operating instructions and procedures, especially the operating and emergency procedures, and of the administrative rules;

(f) The records and reporting systems;

(g) The training and qualification of research reactor personnel, including levels of staff and their suitability for the work;

(h) The occupational and fire safety aspects;

(i) The management system, organization and programme for operation;

(j) The emergency plan;

(k) The accounting measures for nuclear and radioactive materials;

(l) The physical protection arrangements important for safety.

4.10. The documents, which should be submitted to the regulatory body for review and assessment in order to obtain authorizing for commissioning Stage C "Power ascension tests and power tests", should include:

(a) The records and results of 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) Revised arrangements.

4.11. The documents, which should be submitted to the regulatory body for review and assessment in order to obtain authorizing 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) Revised arrangements.

(e) The periodic testing, maintenance, inspections, 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 having 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 [5].

4.13. Before the authorization for 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 program

Detailed guidance for decommission projects is provided in [15]

APPENDIX

The Appendix has been divided into 20 chapters dealing with standard specific topics that are addressed in a safety analysis report. The chapter headings of the Appendix are, in general, the headings that may be appropriate for the safety analysis report. The areas in which basic information is required by the regulatory body, such as site characteristics, reactor (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.

CONTENT OF A SAFETY ANALYSIS REPORT

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 reactor building to convey an impression of the research reactor and its components. The reactor site should 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, which are already foreseen, should be described in this section.

Historical review

A.1.3. The operational history of the research reactor should be presented. For existing reactors an overview of the operational experience as well as 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 owners or agents

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

Safety features

A.1.6. This section should briefly state the basic safety principles adopted for the design, construction and operation of the reactor and the nuclear safety criteria for acceptance. The safety features, components or systems incorporated into the research reactor, which will be described in more 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, and the safety analysis related to the experimental programme and the provisions are 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 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 is required in support of the issue of a licence but which has not been supplied in the safety analysis report.

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, components, equipment and systems 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, considering 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 which 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 NS-R-4. 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 on intrinsic physical principles);
(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, separation, 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; and
(m) Radiation protection.

Emphasis should be placed on the principles used in design and not 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, fire brigade, etc);
(6) Consideration of human factors and ergonomic principles to reduce the potential for human error and to relieve stress of 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) Maintaining of fuel design margins (e.g. burn-up level balancing with experimental requirements, residence time and water chemistry) and

(g) Design provisions to prevent, or 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; and

(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; and

(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;

(c) Design requirements to ensure that safety system settings are not adversely affected (e.g. experiments influencing flux measurement); and

(d) 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: and

(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; and

(e) Requirements for safety support systems.

(14) Design bases for equipment qualification for natural events, environmental conditions, fire

protection and other hazards including theft of radioactive material or sabotage;

(15) Methods employed for protection against dependent failure;

(16) Capability for surveillance and maintenance of safety related equipment; and

(17) Radiation protection in design, including:

(a) Reducing exposures by design features;

(b) Control of releases;

(c) Control of radioactive materials;

(e) Area classification and access control; and

(f) 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 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:

(1) Wind and tornado loadings;

(2) Water level (flood);

(3) Protection against missiles from internal and external sources, including aircraft;

(4) Seismic hazard and seismic analysis;

(5) Security related events, including terrorist attacks and theft of radioactive material or sabotage;

(6) Fire and explosions; and

(7) Roof loadings from accumulated rain, snow, ice, dust, or other natural materials.

The extreme weather conditions due to the 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 systems, structures and components;
- Thermo hydraulic and neutronic design;
- Electrical design;
- Design of instrumentation and control systems;
- Shielding and radiological protection;
- Fire protection;
- Inspection, testing and maintenance 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, 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, the building layout and zoning, the location of fire barriers, and the safety system layout and protection (including separation of safety related redundant systems). The fire protection system should be described in Chapter 10 of the safety analysis report, see in 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 vacuum. Qualification tests and analyses that have (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 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 distribution, land use, site activities and planning controls. The purpose is to indicate how these site characteristics have influenced the research reactor design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint. Additional information on siting can be found 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 (Safety Analysis) 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 which indicates:

(a) Research reactor property and boundary lines;
(b) Location and orientation of principal buildings and equipment;
(c) Location of any industrial, commercial, military facilities, institutional, recreational or residential structures;
(d) Nearby highways, roadways, airports, waterways and rail lines;
(e) Boundary lines of the area controlled by the operating organization; and
(f) Boundaries for establishing effluent release limits.

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, as well as any activities unrelated to the research reactor's operation that will be permitted in the site area[21].

External events

A.3.6. This section should describe the site related phenomena and characteristics, of both natural and man induced origin, which must 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 which will constitute the design basis events for important natural phenomena and man induced effects; further information on design criteria for protection against these effects should be given in Chapter 2, see para A.2.6.

[21] 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.

Geology and seismology[22]

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 the reported earthquakes that could reasonably affect 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 dynamic stability of all soil or rock slopes, both natural and man-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; and
- Assessing the potential for volcanic activity; and
- .Assessing the liquefaction and ground motion potential.

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 should be provided, including, where applicable, hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, freezing rain, snow and ice, and sandstorms.

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 man made structures should be indicated, including darns, diversion channels and any flood control measures. Foreseeable changes in

[22]). Additional guidance on siting and site evaluation is provided in [25] and [26]

land use that may influence hydrology should b 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 given.

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; and
- Seismically induced phenomena such as tsunamis and dam failures.

Nearby industrial, transport and military facilities

A.3.15. All present or projected industrial, transport, or military facilities which 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 (air, land, water), transport facilities (rail lines, docks, anchorages, airports), oil and gas pipelines, drilling operations and wells, and underground storage facilities. The potential adverse effects that these facilities can 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 materials on human. 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 a radiological impact section is not provided, justification for deleting this safety analysis report section should be provided.

A.3.18. Information should be included which, in combination with details of radioactive discharges and of the radionuclide behaviour/transfers presented in other chapters, will permit an assessment of the doses to individuals and the population, and of contamination of biological chains and food chains. This information should cover the entire region likely to be affected, taking into account 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, 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) Lands 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 and recreational purposes;
(e) Bodies of water used for commercial and sport fishing;
(f) Bodies of water used for commercial purposes and recreation; and
(g) Direct and indirect pathways for radioactive contamination of the food chain.

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 underground), and in flora and fauna. If there was a nuclear installation on the site, a brief description of any incidents that have led to residual activity at the site should be provided.

Atmospheric dispersion of radioactive materials

A.3.22. This section should describe the models used to assess the atmospheric dispersion of radioactive material released under operational states and 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 research reactor characteristics that may effect atmospheric dispersion. The accuracy and validity of the models, including the suitability of input parameters, source configuration and 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 off-site locations, or refer to radionuclide atmospheric concentrations and dose calculations should be presented in Chapter 12 (Operational Radiological Safety) and Chapter 16 (Safety Analysis).

Dispersion of radioactive materials through surface waters and groundwater

A.3.24. This section should indicate locations near the research reactor where radionuclide releases could be discharged or where they could enter surface waters or groundwater. The results of hydrological and hydro-geological investigations 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 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 Chapter 12 (Operational Radiological Safety) and Chapter 16 (Safety Analysis) of the safety analysis report.

Site adequacy 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 and water use in the region;
- Potential radioactive source terms, and doses to the population from direct radiation fields and from airborne/aqueous pathways;

- Potential contamination of the food chain;
- Potential doses to 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; and
- The feasibility of emergency plans (if they are required), taking into account the population distribution, national and international boundaries, special groups (e.g. hospitals), special geographical features (e.g. islands), availability of evacuation routes and refuges for evacuees and communication and transport facilities.

Monitoring of site related parameters

A.3.27. This section should define site related parameters, which could be affected by the external events which have been taken into account for the analyses, e.g. affected by seismic, atmospheric, water and groundwater related, demographic, industrial, and transport. The strategy for monitoring, the provisions for monitoring and the use of the results in preventing, mitigating and forecasting 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 of 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 risk to the population from accident conditions, including those which 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 system), emphasizing

those characteristics of the building, which assist in maintaining acceptable radiation levels on and off site during all operational states. Information on the requirements of the reactor building is presented in Section 6 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.) regarding 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, including requirements for containment or means of confinement, including the ventilation exchange rates at the different operation modes, should be described. If applicable, distinction should be made between the system used during 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, such as a system for controlling the release of fission products should, be described.

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 drawings, including flow and instrumentation diagrams.

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

Auxiliary structures

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

CHAPTER 5: 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 heat removal from the core after shutdown, including accident conditions;

- Containing radioactive material in order to minimize its release to the environment.

A.5.2. This section should provide information pertaining to operational states, including the portions of the safety analysis dealing with them. The consequences of failures and accidents are treated in Chapter 16 (Safety Analysis).

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 which 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, burnable poisons;
(c) Fuel geometry, dimensions, tolerances, etc. (together with drawings);
(d) The material properties required for the analyses mentioned in paras 5.5-5.8;
(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 which 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 which 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 which shows that the fuel element cladding can withstand the chemical environment to which is subjected during use and storage, taking into account the effects of temperature and irradiation.

A.5.8. An analysis should be provided which shows that the intended irradiation conditions and limits (fission, density, total fissions at the end of life, etc.) are acceptable and will not lead to undue deformation or swelling of components which 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 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 which demonstrates that the reactivity control systems can fulfil their designated safety functions during all foreseeable operating conditions. Only the safety functions ensuring reactivity control (such as insertion capability) should be addressed here. All the other reactivity aspects should be treated in the section on nuclear design, see paras A.5.13.–A.5.16; the incorporation of the protection and power regulating systems is treated in Chapter 8 (Instrumentation and Control).

A.5.11. Basic information should be provided on the design of reactivity control systems, including the materials, redundancy and diversity aspects, anticipated performance characteristics (such as drive speed, actuation and insertion time), fail-safe features, etc. Ageing effects due to deterioration of properties, as well as irradiation damage should be considered too.

A.5.12. An analysis should be provided, which shows that the reactivity control system will function properly under 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 which 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 which affect the nuclear properties of the core. Since research reactor core configurations may change with the changing experimental applications and requirements, the analysis may use a standard core configuration, which has conservative properties with respect to all other configurations. An explanation of the intended fuel replacement strategy should complement the information. This information should be supported by drawings.

(b) Horizontal and vertical distributions of the neutron flux in the core at thermal 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 life, minimum shutdown capacity, reactivity feedback properties with regard to temperature, void, etc., and reactivity worth's of individual core components (fuel elements, irradiation devices, etc.).

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

A.5.16. An analysis should be provided, which shows that the effectiveness, speed of action and shutdown margin of the reactor shutdown system[23] 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 made and maintained subcritical under all operational states and accident conditions.

Thermalhydraulic design

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

[23] Some reactor designs feature more than one shutdown system. The analysis should cover all of them.

an adequate thermal safety margin will be available to prevent or minimize fuel damage in 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, 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 which may contain fissile materials, as derived from the nuclear design characteristics provided in para A.5.14 (b).

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

A.5.20. An analysis should be provided which proves that the maximum thermal load to which any fuel element in the reactor is subjected during any operational state does not exceed the available cooling capacity, whether by forced 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 thermal hydraulic load and void fractions should be clearly described along with the justification for there 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) as well as for ‘conservative’ conditions (taking into account 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, which shows that all materials which have been selected for the construction of safety relevant components and structures can withstand the nuclear, thermal and chemical environment to which they are subjected, without unacceptable worsening

of the performance of the safety functions of such components and structures. 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 selection process of materials.

A.5.24. Items which 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) Reactor tank and related components constituting the primary coolant boundary; and
(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 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 which remove the heat from the reactor. The description should contain the main design 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 the 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 of which the components are made 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 the 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 the 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 from which the components are made and corrosion control measures should be specified. The 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, from which the components are made, should be specified; the effects of irradiation and corrosion should be discussed. The 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 with which this system must cope 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 of which the components are made should be specified, the effect of irradiation, if any, should be discussed, as well as any environmental effects and the ageing effects should also be discussed. The procedures of 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 material of which the components are made should be specified; the effects of irradiation, if any, the corrosion and the 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 resins exchange and the shielding used to protect persons during this operation. This may be done in this section, or reference may be made to Chapter 10.

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 of which the components are made should be specified. The means for monitoring the 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. The relevant chemistry control and chemistry data of the coolant should be presented, e.g. details of new water treatment and degassing and demineralising 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 engineered safety features (ESFs) provided in the research reactor. Examples of ESFs are an emergency core cooling system and means of confinement or containment system. 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 ESFs should be discussed in detail. The accidents with which these systems must cope should be presented and analyses should be provided which demonstrate that the systems fulfil the requirements. The subsystems which are essential for the proper operation of the ESFs should be described (e.g., uninterruptible power supply for the emergency core cooling system). The extent to which the ESFs are automated and the conditions for which manual override is warranted should be clearly indicated.

A.7.3. Information should be provided on:

(1) Component reliability, system interdependence, redundancy, diversity of fail-safe characteristics and physical separation of redundant systems;
(2) Evidence that the material used will withstand the postulated accident condition (radiation levels, radiolytic decomposition, etc.);
(3) 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;
(4) Effect of aging on the operability of the ESF.

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

CHAPTER 8: INSTRUMENTATION AND CONTROL

A.8.1. This chapter of the safety analysis report should provide information regarding the instrumentation and control (I&C) systems of all safety systems and safety related items and systems. The information provided should emphasize those instruments and associated equipment which affects reactor safety. The requirements for I&C are contained in paras 6.136.-6.144. of Ref. [2].

A.8.2. All I&C 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 I&C 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 which cannot be replaced easily.

Reactor protection system

A.8.4. The requirements for the 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 failure as well as with single failure.

A.8.6. For computer based digital protection systems, evidence of software verification and validation should be included.[24]

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 (temperature, humidity, high voltage, electromagnetic fields, etc.) from influencing the reactor protection system, and 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 interface between the power regulating system and the

[24] Guidance for software verification and validation can be found in Ref. [14].

reactor protection system should be identified and analysed to confirm that they do not lead to degradation of safety.

Other instrumentation and control systems

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

- Fire protection system;
- Experimental control;
- Ventilation control;
- Secondary cooling system;
- Coolant chemistry control;
- Radiation monitoring system;
- Seismic monitoring system, and
- Monitoring system for external meteorological and hydrological conditions.

Alarm system

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

Interlocks

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

Control room

A.8.13. This section should include a description of the instrumentation systems, which 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, if 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 emphasis on 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 which could affect safety should be discussed

Off-site power supply

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

Emergency power supply

A.9.3. This section should describe the design and operation of the emergency power supply, emphasizing 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/manual); and
(e) Duration of operation with or without diesel back-up.

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 routing

A.9.6. Information should be provided on the types of cables used. The adequacy of the measures employed to separate the cables in order to maintain redundancies, prevent cross-talk and provide fire protection should be demonstrated.

CHAPTER 10: AUXILIARY SYSTEMS

A.10.1. This chapter 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 evaluation demonstrating how the system satisfies the requirements of the design basis, the testing and inspection to be performed to verify system capability and dependability, and the required instrumentation and control 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 contribute to the control of radioactivity inside the research reactor. For those systems also foreseeable ageing effects, which could affect safety should be discussed.

Fuel storage and handling

A.10.2. This section should describe systems for storing fresh and spent fuel, for cooling and cleaning the spent fuel pool (if 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 tools and the 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 burn-up history, refuelling frequency, and inspection and storage requirements including that for 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 demineralised water. In each case, the information provided should include the design bases, a system description, flow and instrumentation diagrams, a safety evaluation, 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, process sampling, or equipment and floor drainage systems, should be discussed in this section. The discussions should include the design bases, a system description, a safety evaluation, 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) should be discussed here. 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 could 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 system, sanitary provisions, sewage systems, and gas service systems.

CHAPTER 11: RESEARCH REACTOR UTILIZATION

A.11.1. This chapter should describe the expected experimental use of the research reactor and 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 general public. Additional guidance may be found in Ref. [2], and in the Safety Guide on Safety in the Utilization and Modification of Research Reactors, Safety Series [5].

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, experimental loops, etc. Ageing effects which 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, which 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, which will not be allowed to be used in experiments in or near the reactor core should be specified, together with materials that may only be utilized under additional safety conditions.

CHAPTER 12: OPERATIONAL RADIOLOGICAL SAFETY

A.12.1. This chapter should describe, for normal operational conditions:

(a) The radiation protection programme, including the radiation protection policies, objectives of the operating organization;
(b) Sources of radiation at the research reactor;
(c) Research reactor design for radiological safety;
(d) Waste management systems[25];
(e) Dose assessment for normal operation;

[25] In some cases, the waste management systems and operational radiological safety are discussed separately.

(f) Conclusions.

A.12.2. The estimated radiation exposure of the staff and the general public for accident conditions should be analysed in Chapter 16 (Safety Analysis). 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. Radiological emergency planning is described in Chapter 20 (Emergency Planning and Preparedness), and irradiated fuel management should be treated in Chapter 10 (Auxiliary Systems).

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], which refer to the IAEA Basic Safety Standards for Radiation Protection and the recommendations of the International Commission on Radiological Protection. In particular, this section should summarize the legal dose limits to both occupationally exposed personnel and the general public, as well as the operational emission limits based on these dose limits. The regulatory requirements for maintaining exposures and releases of radioactive wastes and effluents below these legal dose limits should be described, as well as the reference level of the doses and releases established by the operating organization to assist the research reactor management in applying the optimisation principle[26] to ensure that the radiation doses and operational emissions are as low as reasonably achievable and below the above mentioned limits. The records which should be kept to prove that exposure to radiation is adequately 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, as well as the emission control philosophy at the research reactor, including the organizational policy concerning control and monitoring of releases and the evaluation of trends.

Organization, staffing and responsibilities

[26] Guidance on the radiation optimization principle can be found in Ref. [18].

A.12.5. This section should describe the administrative organization of the radiation protection management and staff, 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 radiation protection advice, 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 applicable to the radiation protection activities.

Facilities, equipment and instrumentation

A.12.6. The health physics facilities and equipment, such as laboratories for radioactive analysis, contamination control equipment and decontamination facilities, should be described, including the location of these facilities, as well as the arrangements for maintenance and calibration of health physics instruments and for personnel monitoring (e.g. film badge, thermo luminescent dosimetry service).

A.12.7. This section should describe the radiation and contamination monitoring stations, including fixed hand and foot monitors, portal monitors (if used) and portable activity monitors located at these stations. The portable and laboratory equipment and instrumentation for performing radiation and contamination surveys, for contamination control between different access zones, for airborne radioactivity monitoring/sampling, 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, are 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, and only a brief summary may be given in this section.

Procedures and training

A.12.11. An overview of the written procedures for the radiological 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 resident times in radiation 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 internal radiation exposure assessment, 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 of handling and storage of sources, radioisotopes or other radioactive material; and
- 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 for normal operation and maintenance, in-service inspection and refuelling. References should also be made to the operating procedures, which include provisions for monitoring of systems that collect, contain, store or transport radioactive liquids, gases and solids. Any procedures relating to experimental facilities, isotope production and laboratory activities should be referenced.

A.12.13. This section should describe the methods and procedures for controlling and evaluating exposures of experimenters and other personnel (e.g. contractors and students) 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 (Emergency Planning and Preparedness) for emergency situations at the research reactor when dose levels may be high.

A.12.15. This section should give a brief description of the radiation protection training programme for the radiation protection management and staff, 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 the effluent 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 and airborne) 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, design of ventilation systems, dose assessment, waste management and determination of effluent releases.

A.12.19. For typical sources that are shielded or contained, information should be provided on the form, location, geometry, isotopic content and activity. For typical liquid and airborne sources, 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 should provide drawings of the research reactor, showing the location of all typical sources.

Research reactor design for radiological safety

A.12.22. In the description of the design considerations for the research reactor and equipment it should be demonstrated that external and internal radiation exposures to personnel and the general public are based on the radiation protection policy described in para A.12.3. It should be described how the design philosophy reduces exposure of personnel, minimizes the undesirable production of radioactive material, reduces the need and the time spent for maintenance and operational activities with the possibility of internal or external exposure, and keeps the 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 general public and how it prevents other hazards. This layout may include zones which are classified according to their potential for radioactive contamination and/or radiation 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, which guard against approach by personnel to areas of high radiation fields and potentially contaminated areas, or which prevent the placement of a radiation source (e.g. spent fuel or activated/irradiated material) in an area where personnel is 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, including the radiation levels external to the shielding at locations where occupancy may be required, as well as the materials used, the criteria for penetrations of the shielding, and the calculational methods. The section should also describe other protective features, such as geometric arrangements (e.g. distance) or remote handling to ensure that the exposure of research reactor personnel and of the general public are within the specified requirements and based on the optimization principle, as well as the methods ensuring that beam tubes and other experimental facilities are adequately shielded against radiation streaming during experimental use.

Ventilation for radiological protection

A.12.25. This section should discuss the radiological protection aspects of the ventilation system on the basis of the description of the system in Chapter 4 (Buildings and Structures) or Chapter 7 (Engineered Safety Features).

Radiation monitoring systems

A.12.26. This section describes the permanent monitoring systems for radiation area, effluents and airborne radiation, including information on:

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

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

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

Waste management systems[27]

Solid waste

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

(a) The types and class of waste, the sources and quantities of solid wastes, including the physical form, volume and isotopic compositions, and the measured or estimated activity;
(b) For wet waste, the methods of dehydration; and
(c) The methods of collection, processing, packaging, storage and shipment.

Liquid waste

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

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

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

Gaseous waste

A.12.32. This section should describe the treatment of gaseous sources that are considered to be waste including:

[27] For additional guidance see IAEA Radioactive Waste Safety Standards (RADWASS) programme publications.

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

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 does not result in off-site doses to the general public that exceed regulatory 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 have demonstrated that radioactive releases are a small fraction of the operational emission limits and are acceptable, and that direct and indirect radiation is also within acceptable limits, this section should provide only a summary of all pathways of radiation: airborne, liquid, direct and indirect radiation exposure.

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 boundary and at off-site locations due to the effect of all releases. A description of the calculational assumptions, methods and tools should be presented as well. It should be shown that the combined effect of all releases meets regulatory requirements for doses to the public.

A.12.37. This section should state the criteria to be used for determining that the releases of gaseous and liquid wastes are at an acceptable rate. The effluent concentrations by radionuclide and total annual release 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 to the environment should be identified, providing for each quantity:

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

Occupational

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 research reactor radiation areas 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 handling, refuelling and in-service inspection. An estimate of the annual dose at the boundaries of the restricted 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 materials is acceptable. If data are available, a summary of the annual doses to research reactor staff should be provided.

Conclusion

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

CHAPTER 13: CONDUCT OF OPERATIONS

A.13.1. This chapter should describe the organizational structure and the way in which the operating organization will conduct 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, and records and reports. General requirements and additional guidance on the above topics can be found 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 operational 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 for the personnel required during 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 kind 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, as well as the composition and qualifications of the review and audit group, the rules for group meetings, and the items to be reviewed by the group, such as changes to the licence, the operational limits and conditions, the procedures and the research reactor; 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 interval between audits, and how audit findings are addressed by the research reactor management within the management system programme for operation (see Chapter 19, Management system).

Operating instructions and procedures

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

A.13.10. These written instructions and procedures, see also Ref. [11] should include information on the following items:

- Reactor start-up, operation and shutdown;
- Loading, unloading and movement of fuel and irradiated material;
- Inspection and tests of items important to safety, in particular the safety systems;
- Setting up, testing and conducting of performance and test of experiments with safety significance;
- Maintenance, in particular concerning major components or systems important to safety;
- Radiation protection;
- Response to anticipated abnormal occurrences, system or component failures and accident conditions;
- Effluent monitoring and environmental surveillance;
- Emergency situations;
- Physical protection (see A.13.12 and A.13.13); and
- Fire protection.

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

Maintenance, testing and inspection

A.13.11. This section should describe the conduct of the research reactor maintenance, periodic testing and the inspection programme for reactor equipment and components. An overview is sufficient if the detailed programme is given in supplementary documents. The maintenance, 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 inspection, testing or maintenance;

(e) Approval of maintenance work; and

(f) Resumption of normal operation after maintenance.

Physical protection[28]

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

A.13.13. The physical protection of the research reactor may be described in a separate document, which would be confidential.

Records and reports

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

(a) Reactor operation (log book, strip charts, check-lists, automatic data readout);

(b) Operational status (type and number of operational components, and of components out of service);

(c) Maintenance, inspection and testing 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 on medical examinations;

(i) Effluent and environmental monitoring;

(j) Safety related component failures and occurrences;

(k) Documents on training and retraining.

[28] Guidance on nuclear security is presented in the Nuclear Security Series [31], [32], [33] and [34], as well as in Refs [29] and [30]

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 research reactor operation (see Chapter 18, Management Systems).

Feedback of operational experience

A.13.16. This section should describe the process for the evaluation and feedback of the operational experience, including the evaluation of trends of the operational disturbances, trends in malfunctions, near misses and incidents occurred at the research reactor as well as in other research reactors.

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 such as: construction, operation, modification, decommissioning of the research reactor.

A.14.2. This section should briefly discuss the following points, in connection with the related information included in Chapter 3 (Site Characteristics):

(a) The environment 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; and
(e) An analysis providing a balance of the environmental effects of the licensing action and the alternatives available for reducing or avoiding environmental effects, as well as a summary of the environmental, economic, social, technical and other benefits resulting from the research reactor.

A.14.3. Some licensing actions may have little or no environmental effect. In these cases, the decision to take these 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 the 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 verified adequately. For existing research reactors this chapter should describe the commissioning programme which have 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 verified adequately. 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 sequences:

– 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, Management system);
(d) A summary schedule for the major phases of the programme; and
(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 the commissioning information of similar operational facilities will be utilized. The method for reporting the commissioning results to the regulatory body should be described, including resolution regarding non-conforming 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 decommissioning

A.15.6. After the 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;

(c) A summary of the accepted non conformances; and when appropriate, their corrective actions; and,

(d) Overview of possible modifications of systems, structures, components, safety analysis and safety analysis report, procedures, etc..

Existing research reactors

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

(e) A summary of the programme and objectives;

(f) A summary of the results;

(g) A summary of the accepted non conformances; and when appropriate, their corrective actions; and,

(h) The method for updating the safety analysis report, if required, to include the results of modifications and related commissioning tests.

Commissioning of modifications

A.15.8. The information outlined in the foregoing should also be included in a safety analysis report involving modifications to existing 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, the research reactor design and especially the structures, systems and components important to safety should be evaluated for their susceptibility 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 capability of the research reactor to control or 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 events initiating accidents 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.16.45.);
(5) *Summary:* A summary of significant results and conclusions regarding acceptability (paras A.16.46.–A.16.47.).

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 may be of some assistance in completing this section, but the level of detail of this annex 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 can be found in Ref. [19].

Research reactor characteristics

A.16.5. This section should summarize the reactor parameters and initial conditions used in transient analysis (paras A.16.18.–A.16.23). These parameters and permitted operating bands will form the basis of the operational limits and conditions in Chapter 17 (Operational Limits and Conditions).

Core parameters

A.16.6. A summary of the research reactor parameters and ranges for specified operating conditions considered in the safety analysis should be given. 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 may not be 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 and moderator temperature reactivity coefficients;
(k) Void reactivity coefficient;
(l) Available shutdown reactivity worth; and
(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 burn-up, refuelling or other factors.

A.16.8. The permitted operating band on research reactor system parameters should be specified, including permitted fluctuations in a given parameter and associated uncertainties. The most adverse conditions within the operating band 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 backup cooling.

Selection of 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 provides some information on methodologies. The selection should consider the points mentioned in paras A.16.11.–A.16.12.

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 event; and
(h) Human errors.

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. Such events would 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; and
(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 depends on the research reactor type. For those situations where a particular initiating event is not bounding, only the qualitative reasoning which led to that conclusion need to be given, together with a reference to the section presenting an evaluation of the more bounding initiating event. Further, for those initiating events which 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 need not be performed for each such initiating event.

Identification of causes

A.16.15. For each event evaluated, a description of the occurrences that led to the initiating event under consideration should be included both due to equipment failure and 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 time-scale, e.g. flux monitor trip, or start of insertion of control rods;

(b) Indication of proper functioning of normally operating reactor instrumentation and controls and their failure to function;
(c) Indication of proper functioning of reactor protection and safety systems and 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); and
(g) Justification for exclusion of sequences that are outside the design basis.

A.16.17. Not every postulated initiating event need 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 component 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 which 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 reactor core and system performance under accident conditions should be discussed and the important results of the analysis presented. The discussion should include, where appropriate, an evaluation of the parameters that may affect the performance of barriers that restricting the transport of radioactive material from the fuel to the environment (e.g. fuel cladding interaction and fuel failure modes, primary cooling system and building/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, yield conservative predictions,

represent all important physical phenomena and have been validated properly. This section should provide only a summary of mathematical models and computer codes or lists, 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; and
   (vi) The method combining these codes (if a set of codes is used).

(b) A brief description of input data to 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; and
   (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 experiment or operation, or with other models that have also been compared with the experiment or operation;
   (ii) Demonstration of adequate numerical accuracy or of the degree of conservatism;
   (iii) Confirmation that the modelling represents all important physical phenomena; and
   (iv) Confirmation that the empirical correlations are conservative, based on experiment (where practicable) and appropriate for the range of operational parameters.

A.16.21. *Input parameters and initial conditions.* The input parameters and initial conditions used in the analysis should be clearly identified. Annex II provides an example for a list 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.

A.16.22. *Results.* 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, core average temperature and hot channel exit temperature);
- Core temperature (maximum fuel centre line temperature, maximum clad temperature) and maximum fuel enthalpy;
- Reactor coolant inventory (total inventory and coolant level in various locations in the reactor coolant system); and
- Parameters of the secondary heat exchanger system (inventory and level, enthalpy, temperature and 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 given.

Classification of damage states

A.16.25. The transient analysis may show that the fuel design limits have been exceeded, resulting in some fuel cladding damage. An estimate of the type of damage, the quantity of fuel

affected and other factors (such as fuel and cladding temperatures, coolant characteristics, chemical interactions) should be provided.

A.16.26. Some event sequences may result in different radiological hazards, including failure of experiments or of irradiation/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 which 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 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 to 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 meet regulatory requirements).

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

A.16.30. *Assessment of releases to the reactor building.* The quantity of radionuclides released inside the building, the isotopic content and other physical factors characterizing the

[30] Additional information can be found in Ref. [20]

releases should be described for each relevant sequence. The parameters and assumptions used in the analysis should be presented, including:

(1) The fission product inventory (or radionuclide inventory for accidents not involving fuel);
(2) The nature of the fuel element damage, and fraction of the damaged fuel cladding;
(3) The fractions of the fission product release from the fuel; and
(4) The retention factors and plate out of radionuclides in water and on surfaces.

A.16.31. *Assessment of releases from the reactor building.* The quantity of radionuclides released to the environment, the isotopic content and other physical factors characterizing the release should be given for each of the event sequences that results in releases to the reactor building. Both airborne and aqueous releases should be considered. The parameters and assumptions used in the analysis should be presented, including:

(1) Removal of radionuclides by liquid and gaseous hold-up systems, recirculation and ventilation systems, including filter efficiencies;
(2) Surface deposition and resuspension;
(3) Radionuclide hold-up time, decay and precursor production;
(4) Reactor building leak rate or liquid effluent release rate;
(5) Release mode (single puff, intermittent, continuous); and
(6) Release point (stack, ground level, etc.)

A.16.32. *Assessment of other hazards*. Descriptions should also be given of accidents which might result in significant exposure to staff members or the general public to direct 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 which are contained but which present a radiation hazard;
- Aqueous spills or releases of other radioactive material that are contained locally; and
- Loss of shielding.

Evaluation of the radiological consequences

A.16.33. This section should discuss the calculational methods used to determine 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 radiological consequences are associated with a given event sequence, this section should simply contain a statement to that effect.

A.16.35. *Methods for analysis of the radiological consequences.* The methods used to analyse radiological consequences that might result from reactor incidents or accidents should be presented in this section. The assumptions and methods used in determining the 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 radiological consequences should include the following:

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

A.16.37. *Dose results.* This section should present the results of the dose calculations giving the effective dose equivalent at the site or the exclusion boundary[31] and, if necessary, the effective dose equivalent for the general 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

[31] The deliberate *exclusion* of a particular area of *exposure* from the scope of the research reactor of *regulatory control* on the grounds that it is not considered amenable to *control* through the regulatory instruments in question.

during the accident to the control room personnel and personnel at other places on site, where appropriate.

A.16.38. Consideration should be given to direct radiation fields, both from aqueous and atmospheric releases, and to the possibility of ground contamination.

A.16.39. *Direct radiation fields.* Direct radiation fields associated with releases occurring within the research reactor and which could result in radiation doses 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 time-scale of the release;
- Radionuclide decay and precursor production;
- Shielding parameters, build-up factors and scattering (e.g. sky shine); and
- Distance to critical groups and the time-scale over which doses are calculated.

A.16.40. *Aqueous releases.* This section should summarize the assessment of aqueous releases and, where appropriate, dispersion in surface waters and groundwater, contamination of biological chains and food-chains, and the consequent doses to individuals and 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:

- Direct radiation from released fluids;
- Evaporation or airborne resuspension of radionuclides from the released fluids;
- Ground contamination; and
- 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 or recirculation systems;
- Potential discharge points, the inventory of radionuclides released, their concentration in the fluid, the release rate and mode of release (continuous, intermittent);
- 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 which may be needed to determine radionuclide movement and exposure pathways;
- Direct and indirect pathways for radioactive contamination of the food-chain; and
- Radionuclide uptake in humans and 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 aqueous releases to surface or groundwater aquifers are 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 discussed.

A.16.44. Atmospheric releases. This section should present the doses to research reactor staff and to the general public after an airborne release of radioactive material from the research reactor, taking into account 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 form, and any other factors necessary to completely specify the dispersion of radioactive material to the environment, including buoyancy;
- Mode and characteristic of release (single release (puff), intermittent, continuous and related release durations);
- Location of release and characteristics, including stack height and diameter ;
- Distance to receptors and intervening terrain;
- Meteorology data, including wind speed and direction, and data on inversions and other atmospheric stability;
- Wake effects of the building;
- Diffusion parameters;

- The physical and chemical form of radionuclides at the receptor location, and whether they are airborne or deposited; and
- Results of dose calculation (immersion, ingestion and/or ground shine).

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

CHAPTER 17: OPERATIONAL LIMITS AND CONDITIONS[32]

A.17.1. This chapter of the safety analysis report should contain the operational limits and conditions important to safe reactor operation which have been derived from the safety analysis. The operational limits and conditions represent an envelope of parameters, developed by the operating organization, which will protect the research reactor, the staff, the public and the environment from undue exposure if they are not exceeded. Therefore, it is essential that the operational limits and conditions are 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. Additional information is contained in paras 7.29.-7.41. of Ref. [2] and in paras 3.27 – 3.43 of Ref. [11].

[32] Additional guidance to be followed can be found in Ref. [11]

A.17.2. The operational limits and conditions are based on an agreement between the operating organization and the regulatory body and form an important part of the requirements for authorization of the operation of the research reactor by the regulatory body. 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 safe operation, each OLC must 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 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 OLC should cover the following points:

(a) The objectives to be met by the establishment 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 fuel plate temperature or pool water level, or to conditions of these barriers. Sometimes the applicability refers to the equipment set-up, such as the minimum number of measuring channels being operable.

(c) The specification(s) of the OLC, 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 or safety calculations included in the safety analysis, which allow for 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 and 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, (Reactor), and 16, (Safety Analysis).

Safety systems settings

A.17.5. Safety system settings should be provided for those variables and parameters which, if not controlled, could result in a safety limit being exceeded. This section should identify the safety system settings and 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, measurement accuracy and system response time. Safety system settings are derived primarily from Chapters 5 and 16.

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. Although in many cases the limiting conditions are established by the administration, they are constraints on equipment and operational characteristics, which are identified in the safety analysis report as being important to safety and which should be adhered to during operation of the research reactor. In some cases, when the process variables or parameters reach a limiting condition for safe operation, they may initiate alarms to enable the operating personnel to take appropriate action in order to prevent safety system settings from being exceeded. Some examples of limiting conditions for safe operation are as follows:

- Core configurations and design limitations (reactivity coefficients, burn-up limits, minimum and maximum number of the fuel and reflector elements, and their geometrical arrangements, inspection, etc.)
- Minimum number, design and performance of reactivity control mechanisms;
- Fuel design parameters (enrichment, fuel type, cladding type, etc.);
- Maximum reactivity insertion rate;
- Minimum operational reactor measurement and control systems and safety set points;
- Equipment required to achieve 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 and effluent monitoring systems and their safety set points for the different operational stages (e.g., shutdown, operation, fuel handling, etc.);
- Limits on effluent releases;
- Limitations on experiments (reactivity, materials, etc.);
- 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 or the acceptability of the substitution of equipment for failed equipment should be stated. Additional guidance is presented in paras 3.27 – 3.32 of NS-G-4.4 [11].

Administrative requirements

A.17.8. This section should contain the administrative and organizational requirements, as well as the organizational structure and responsibility, the staffing requirements, the review and audit of research reactor operation procedures, the review of operational events, reports and records and the radiation protection area classification. These limiting conditions and administrative requirements are derived primarily from Chapter 13 (Conduct of Operations).

CHAPTER 18: MANAGEMENT SYSTEMS

A.18.1. The IAEA requirements and guidance uses the term “management system”, rather than ‘quality assurance’. The term management system reflects and includes the initial concept of ‘quality assurance and quality control’ (controlling the quality of products) and its evolution through quality assurance (the system to ensure the quality of products) and ‘quality management’ (the system to manage quality). The management system is a set of interrelated

or interacting elements that establishes policies and objectives and which enables those objectives to be achieved in a safe, efficient and effective manner.[33]

A.18.2. The operating organization is responsible for the preparation and implementation of an integrated management system that will ensure conformance to every aspect of safety. The principles and scope of the management system should be established in accordance with the general requirements of Safety Requirements No. Ref. [2], and with other national standards.

A.18.3. This section should describe the management system or 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 or implemented. The level of control and verification of quality should also be defined, and the means available for achieving it should be described.

A.18.4. This section should describe or 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, its objectives, status and characteristics, and it 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 related 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 achieved. In particular, the responsibility and authority of the personnel concerned the management system should be defined.

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

A.18.7. This section should describe the procedures covering the operating activities performed under the management system. Examples are activities related to reactivity and criticality management, thermal safety of the core, safety of experimental devices, reactor modifications, component and material manipulations and human surveillance.

[33] Additional guidance om Management Systems could be found in Refs [6] and [7]

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 and the operational procedures to facilitate the decommissioning process. The design basis related to decommissioning should be described.

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

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

CHAPTER 20: EMERGENCY PLANNING AND PREPAREDNESS

Emergency plan

A.20.1. This section should contain or refer to a plan, which will provide reasonable assurance that actions can and will be taken to mitigate emergencies that might occur at the research reactor. However, safety precautions taken in the design and operation of the reactor will greatly reduce the risk of an accident. Additional information on this plan may be obtained from 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. The information should cover the following items:

(a) The emergency organization, giving clear instructions regarding authority and responsibility;
(b) The process for identifying and classifying the emergency;
(c) The agreements made with off-site agencies which will help in an emergency;
(d) Notification of on-site personnel and, if necessary, off-site personnel;
(e) Notification of the government and local authorities;
(f) Reliability of communications between the emergency control centre if available and outside locations;
(g) Protective measures;
(h) Equipment 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; and
(l) Adequacy of resources to implement the emergency plan.

Emergency procedures[34]

A.20.4. This section should demonstrate that the emergency plan will be implemented by emergency procedures. These procedures should contain the specific actions which will be taken to mitigate the consequences of emergencies.

A.20.5. This section should contain information on the arrangements for periodic review of the emergency plan and the implementing procedures ensuring that the requirements of new experiments or research reactor modifications are included.

A.20.6. The emergency procedures should contain guidance on limits to exposure of personnel performing rescue missions or missions to reduce the consequences of an emergency.

[34] Additional guidance can be found in NS-G-4.4 ([11], paras 5.53-5.56)

REFERENCES

[1] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles: Safety Fundamentals, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[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, Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. GS-G-4.1, Vienna (2004).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verification for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No NS-G-1.2, Vienna (2002).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in the Utilization and Modification of Research Reactors, IAEA Safety Standard Series DS 397, Vienna (In preparation).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, The Operating Organization and Recruitment, Training and Qualification of Personnel for Research Reactors, IAEA Safety Standards Series NS-G-4.5, IAEA, Vienna (2008).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No NS-R-1, Vienna (2000).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Review and Assessment of Nuclear Facilities by the Regulatory Body, IAEA Safety Standards Series No. GS-G-1.2, IAEA, Vienna (2002).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Research Reactors, IAEA Safety Standards Series No. NS-G-4.4, IAEA, Vienna (2008).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Commissioning of Research Reactors, IAEA Safety Standards Series No. NS-G-4.1, IAEA, Vienna (2006).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Periodic Testing and Inspection of Research Reactors, IAEA Safety Standards Series No. NS-G-4.2, IAEA, Vienna (2006).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Software for Computer Based Systems Important to Safety in Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.1, IAEA, Vienna (2000).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

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

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

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Radioactive Waste Management in the Operation of Research Reactors, IAEA Safety Standards Series NS-G-4.6, IAEA, Vienna (2009)

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

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Derivation of the Source Term and Analysis of the Radiological Consequences of Research Reactors Accidents, IAEA Safety Reports Series No.53, IAEA, Vienna (2008).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, IAEA, Vienna (2007).

[22] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA , Vienna (1996).

[23] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, Publication 103, Pergamon Press, Oxford and New York (2007).

[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Governmental, Legal and Regulatory Framework for Safety, IAEA Safety Standards Series, IAEA, Vienna (in preparation)[DS415].

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3, IAEA, Vienna (2003).

[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Licensing Process for Nuclear Installations, IAEA Safety Standards Series, IAEA, Vienna (in preparation)[DS416].

[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4, IAEA, Vienna (2009).

[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series WS-G-2.1, IAEA, Vienna (1999).

[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on the Physical Protection of Nuclear Material, INFCIRC/274, and the Amendment of 2005 thereto, IAEA, Vienna (2005).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, The Physical Protection of Nuclear Material and Nuclear Facilities, INFCIRC/225/Rev.4 (corrected), IAEA, Vienna (1999).

[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Engineering Safety Aspects of the Protection of Nuclear Power Plants Against Sabotage, IAEA Nuclear Security Series No. 4, IAEA, Vienna (2007).

[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Safety Culture, IAEA Nuclear Security Series No. 7, IAEA, Vienna (2008).

[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Preventive and Protective Measures Against Insider Threats, IAEA Nuclear Security Series No. 8, IAEA, Vienna (2008).

[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Development, Use and Maintenance of the Design Base Threat, IAEA Nuclear Security Series No. 10, IAEA, Vienna (2009).

[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management for Research Reactors, IAEA Safety Standards Series, IAEA, Vienna (in preparation).[DS412]

ANNEX I:

SAFETY ANALYSIS APPROACH AND METHODS

I-1.1. Annex I outlines some considerations which may be useful in developing a safety analysis for a research reactor. The well accepted basic approach to this is to consider initiating events for credible accidents, using a deterministic method to estimate the maximum possible releases to the environment. Probabilistic methods may be used to evaluate which accident sequences are of higher likelihood; they will be useful also for evaluating relative rankings of risks and hence for providing countermeasures. They may also be used for identifying hidden weaknesses of the design and for quantifying the value of possible improvements or modifications. However, probabilistic safety assessment (PSA) is not treated in this Safety Guide since the deterministic methods are used. For further information on applications of PSA to research reactors, see - [16] and [17][35].

I-1.2. These considerations cover a wide spectrum of research reactors and thus may contain information, which is not applicable to all research reactors. Consequently, these considerations are not intended to be requirements but are provided for additional guidance.

Methods for identification and selection of initiating events

I-1.3. Postulated initiating events are occurrences that may lead to reactor fault sequences or accident scenarios. They originate from component failures, system malfunctions, human error or external events and special internal events.

I-1.4. The method used to identify postulated initiating events and to select sets of particular events for further analysis should be established. This method should ensure that the list of initiating events is as complete as possible, that initiating events are grouped in some 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:

(a) *Lists of initiating events in research reactors.* A list of possible research reactor initiating events in research reactors is given in Table I.

(b) *Engineering evaluation.* Potential sources and types of radiological hazards 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.

[35] More detailed information on the development of deterministic safety analysis can be found in Ref. [19]

(c) *Operational experience.* Past experience from the research reactor or from similar facilities, including examination of safety reports, and IAEA IRSRR database can be used to develop or 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-1.5. Methods used to reject particular initiating events and to exclude them from further analysis should 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 probabilistic bases (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 under this category.

I-1.6. 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 similar influence on reactor behaviour or on structures 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; and

(d) External initiating events that have the potential for a common cause impact on the whole research reactor.

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

I-1.7. The method developed to select limiting initiating events for further analysis should include those having potential consequences that are limiting for all other initiating events in the group.

Methods for event sequence analysis

I-1.8. A method should be developed to evaluate the step by step sequence of events, from the initiation of the event to the final stabilized condition. This method should include any rules or conventions regarding the extent to which reactor systems, including the reactor protection system are assumed to function. If there is a possibility of fuel cladding failure, then other barriers to prevent to spread of activity should be considered, not only if all systems function correctly but also if some of them fail. Consideration should be given to the types of events that will be evaluated using this method, and the types of events that will be evaluated by other methods (see paras I-116 to I-120).

I-1.9. Methods should be established to investigate event sequences. The sequences should include the response of the reactor and the reactor systems, as well as human interactions; possible sequences for the case where a system fails should be described. The following points should be considered:

(a) Use of structured techniques, such as event trees or event sequence diagrams;

(b) Identification of significant occurrences on a time-scale, e.g. 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 research reactor, cooling the fuel and maintaining confinement of radionuclides, including an indication of both the correct functioning of reactor protection and safety systems and their failure;

(e) Required operator actions;

(f) Frequency or probability evaluations to be carried out in assessing the sequence of events; and

(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 release of radioactive material.

I-1.10. Rules or conventions should be established to determine the response of reactor systems; these rules or conventions should 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); and
(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-1.11. Rules or conventions should 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 exclusion of events the occurrence of which are 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-1.12 The effects of dependent failures (e.g. common cause or cross linked effects) and human error should be considered; this includes:

(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; and

(c) Assessments of the validity of any assumptions or rules concerning the response of research reactor systems during accident sequences.

I-1.13. The frequency or probability of event sequences may be evaluated; this would help to determine, which sequences should be excluded from the design basis or to assess the relative risk presented by various sequences. This evaluation should include:

(a) The known or estimated frequency of the initiating event, e.g. loss of electrical 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, and specific meteorology, etc.), which can lead to many similar event sequences and which may have a low cumulative probability; and

(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 independent.

I-1.14. Limiting or bounding event sequences in each class should be selected for further analysis in order to reduce the number of events to be analysed using analysis methods of core transient. Consideration should 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 of 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; and

(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-1.15. General methods used to evaluate particular external and internal events, such as earthquakes, tornadoes or sudden catastrophic rupture of reactor pressure retaining components or reactor internals, should be 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 regarding protection against these events is given in Chapters A.2 and A.3 of the Appendix.

I-1.16. 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} $year^{-1}$) for which protection is provided to preserve essential safety related structures, systems and equipment.

(e) The design basis parameters for the phenomenon are evaluated, corresponding to the design basis return recurrence frequency.

I-1.17. Design qualification may prevent failure of pressure retaining components. In this case the appropriate chapter of the safety analysis report should describe the design and construction standards used (e.g. acceptable engineering codes and practices) to prevent structural failures and to preserve the required safety functions. Reference may be made to the appropriate chapters of the safety analysis report (see Chapter 2 and 3 of the Appendix).

Qualitative evaluations

I-1.18. Consideration should 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; and

(c) Justification of administrative measures to reduce the probability of occurrence of faults.

I-1.19. Such qualitative arguments should be used with caution and the regulatory body should be consulted concerning acceptability.

Acceptance criteria

I-1.20. The significant results of the safety analysis must be compared with the acceptance criteria (see paras 2.14.-2.18.).

I-1.21. The safety analysis report should present not only the acceptance criteria appropriate to the safety analysis but also the results of the comparisons referred to in para I-1.20.

ANNEX II:

EXAMPLES OF INPUT PARAMETERS AND INITIAL CONDITIONS

II-1.1. Examples of input parameters and initial conditions, which should 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/exit temperature
- Core coolant inlet/exit pressure
- 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's (differential and total, shutdown margin)
- Reactivity insertion rate during an emergency.

ANNEX III:
EXAMPLE OF ITEMS TO BE CONSIDERED IN THE RESEARCH REACTOR DESCRIPTION

III-1.1. *Summary description:* A brief description of the following aspects of the research reactor should 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/outlet temperatures;
- Power density.

III-1.2. *Reactor Structures:* A detailed description of the following items is required:

(a) Reactor pool/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) Natural circulation provisions.

The description should include materials and dimensions, and should be supported by drawings. The effects of corrosion, fatigue, neutron doses, etc., on the lifetime of mechanical components that are related to safety should be discussed.

III-1.3. *Reactivity control system, reactor shutdown system:* The function of the mechanical and electrical design should be described. The description should include the materials and dimensions and should be supported by drawings. The reactivity control mechanisms and their instrumentation, such as their position or status (coupled/decoupled), should be presented, together with the insertion time and interlocks. The effects of corrosion, fatigue, neutron doses, etc., on the lifetime of the mechanical and electrical components should also be discussed. The safety related design parameters should be presented, such as:

- Speed of control rod;
- Insertion time of shutdown rods;
- Maximum withdrawal of rods.

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

III-1.4. *Fuel elements:* The fuel used should be specified, including the uranium enrichment and the type of fuel. The description of the fuel element should be supported by drawings, and the main characteristics of the fuel elements should be presented such as:

(a) Thickness of cladding;

(b) Length of active zone;

(c) Width of coolant channel;

(d) Number of fuel plates/pins;

(e) Cladding material;

(f) Uranium loading.

Experience with the fuel used should be described.

If control fuel elements are used which contain channels for the motion of neutron absorbing blades or rods, they should be described.

III-1.5. *Reactivity control systems:* The reactivity control systems should be described, giving the main dimension, the neutron absorber material used, and information on the experience with these or similar reactivity control systems. The description should be supported by drawings.

ANNEX IV:
TYPICAL SOURCES OF RADIOACTIVE MATERIAL OR RADIATION FIELDS IN A RESEARCH REACTOR

IV-1.1. 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 and activation and corrosion products in the pool or coolant system and related systems such as purification system;
- Equipment, systems and piping containing activation sources;
- Solid and liquid waste and waste management facilities, and leakage or spills from these facilities;
- Gaseous radioactive materials from the pool, coolant systems, cover gas systems, reflector systems and experimental facilities connected to ventilation systems or any leakage from these systems;
- Filters of the ventilation systems;
- Airborne radioactive material in areas normally occupied by personnel;
- Experimental facilities with the potential to generate activated or other radioactive material, or facilities for storage and handling of such material, including sample activation/irradiation facilities, in-core experiments and hot cells;
- Material irradiated by the research reactor;
- Neutron start-up sources; and
- Sources for test and calibration of radiation monitoring equipment.

CONTRIBUTORS TO DRAFTING AND REVIEW

Abou Yehia, H.	International Atomic Energy Agency
Boogaard, J.P.	International Atomic Energy Agency
Couturier, J.	Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France
D'Arcy, A.	Nuclear Energy Corporation of South Africa (NECSA), South Africa
Garea, V.	INVAP, Argentina
Hargitai, T.	KFKI Research Institute, Hungary
Shokr, A.	International Atomic Energy Agency