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

DS510A

DRAFT SAFETY GUIDE

(Revision of SSG-20)

Contents

1. INTRODUCTION

BACKGROUND

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

- IAEA Safety Standards Series No. DS510B, Safety in the Utilization and Modification of Research Reactors [2];
- IAEA Safety Standards Series No. DS509A, Commissioning of Research Reactors [3];
- IAEA Safety Standards Series No. DS509B, Maintenance, Periodic Testing and Inspection of Research Reactors [4];
- IAEA Safety Standards Series No. DS509C, Core Management and Fuel Handling for Research Reactors [5];
- IAEA Safety Standards Series No. DS509D, Operational Limits and Conditions and Operating Procedures for Research Reactors [6];
- IAEA Safety Standards Series No. DS509E, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [7];
- IAEA Safety Standards Series No. DS509F, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors [8];
- IAEA Safety Standards Series No. DS509G, Ageing Management for Research Reactors [9];
- IAEA Safety Standards Series No. DS509H, Instrumentation and Control Systems and Software Important to Safety for Research Reactors [10].
- IAEA Safety Standards Series No. DS511, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [11].

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

1.3. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report[1].

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

OBJECTIVE

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

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

SCOPE

1.6. This Safety Guide is primarily intended for use for heterogeneous, thermal spectrum research reactors having a power rating of up to several tens of megawatts. Research reactors of higher power, specialized reactors (e.g. homogeneous reactors, fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops) may need additional guidance. For such research reactors, the recommendations provided in IAEA Safety Standards Series Nos SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [14] and ~~SSG-41~~GS-G-4.1, Format and Content of the Safety Analysis Report for Nuclear Power Plants [15] might be more suitable.

1.7. Research reactors with a low hazard potential having a power rating of up to several tens of kilowatts, critical assemblies and subcritical assemblies might need a less comprehensive safety assessment and safety analysis report than that outlined here. While all recommendations in this Safety Guide are to be considered, some might not be applicable to those research reactors with low hazard potential (see paras 2.15 – 2.17 and Requirement 12 of SSR-3 [1], and IAEA Safety Standards Series No. DS511, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [11]).

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

1.9. This publication covers aspects of research reactor operation that are normally included in the safety analysis report, such as operational limits and conditions[2], commissioning, operating procedures,

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

and utilization and modification; more detailed recommendations on these ~~other~~ aspects of research reactor operation are provided in other Safety Guides [2-11].

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

1.11. This Safety Guide provides recommendations relating to utilization (i.e. for experiments and experimental facilities[3]) only with regard to safety analyses for the safety analysis report for the reactor. Detailed recommendations on safety analyses for experiments at research reactors and experimental facilities[4] are provided in DS510B [2].

1.12. Detailed Recommendations on nuclear security are not provided in this Safety Guide. However, recommendations are provided on the interfaces between nuclear safety and nuclear security and on handling of confidential information. In general, documentation and electronic records relating to safety analysis processes and outputs provide limited information regarding equipment location and facility layout. However, such information needs to be reviewed to identify any sensitive information that could be used to support malicious acts, and such information needs to be protected appropriately. Guidance on sensitive information and information security is provided in IAEA Nuclear Security Series No. 23-G, Security of Nuclear Information [17] and Computer Security at Nuclear Facilities, NSS-17 [17A].

STRUCTURE

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

conditions for safe operation, surveillance requirements and administrative requirements.

[3] An experimental facility includes any device installed in or around a reactor to utilize the neutron flux and ionizing radiation from the reactor for research, development, isotope production or any other purpose.

[4] ~~An experimental facility includes any device installed in or around a reactor to utilize the neutron flux and ionizing radiation from the reactor for research, development, isotope production or any other purpose.~~

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

2. SAFETY ASSESSMENT IN THE AUTHORIZATION PROCESS

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

RESPONSIBILITIES

2.2. The government is required to establish an appropriate governmental, legal and regulatory framework for safety, which will provide the legal and regulatory basis for assessing the safety implications of a research reactor (GSR Part 1 (Rev.1) [18], Requirements 1 and 2). The establishment of an independent regulatory body is an important requirement for an adequate legal and regulatory framework. Requirements on the establishment of a regulatory body are established in GSR Part 1 (Rev. 1) [18]. SSR-3 [1] also establishes requirements for the framework of the system for ensuring safety specific to research reactors.

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

2.4. The content of the application for a licence will depend on the legal and regulatory framework of the State. Relevant requirements for the authorization process are established in GSR Part 1 (Rev. 1) [18]. The information provided in support of a licence application should be commensurate with the magnitude of the potential hazard associated with the research reactor and its utilization, and should be consistent with the particular stage of the authorization process.

2.5. Authorization is an ongoing process, starting at the stages of siting and site evaluation and continuing up to and including decommissioning and the release of the research reactor from regulatory control. The authorization process should be understandable by interested parties and should be

predictable (i.e. well defined, clear, transparent and traceable). The different stages of the authorization process should be established in a coherent yet flexible way in order to achieve the most efficiency. These stages should be discrete and should follow a logical order. Detailed recommendations on the authorization process are provided in IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [19].

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

(1) Siting and site evaluation;

(2) Design;

(3) Construction;

(4) Commissioning;

(5) Operation, including utilization and modification;

(6) Decommissioning and release from regulatory control.

2.7. The operating organization is required to submit a demonstration of nuclear safety, including an adequate safety analysis, at each stage of the authorization process, which should be reviewed and assessed by the regulatory body before the next stage is authorized. Operating Organization, before each stage of the authorization process, should revise the safety analysis report based on the feedbacks from pervious stages. In some States, consideration has been given to the adoption of a 'pre-licensing' process. The pre-licensing process contributes to fostering the mutual understanding of licensees, vendors and the regulatory body on the design concept, safety concepts as well as safety expectations and requirements to be fulfilled. Such an approach may help to minimize the duplication of effort at different stages of the authorization process and it may allow for some stages to be conducted in parallel. It also provides for a clear division of responsibilities, at different stages, between the regulatory body, the vendor and the operating organization; gives the public opportunities for early participation; and ensures that the most important safety issues are dealt with properly at the pre-licensing stage (see para. 2.6 SSG-12, ~~para. 2.6~~).

2.8. At all stages, the operating organization should be able to demonstrate that it has control over the research reactor and that it has an adequate organizational structure, a management system, and adequate resources to discharge its obligations and, as appropriate, its liabilities. Further requirements and recommendations on the management system are provided in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [20], IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [21], and IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [22].

2.9. To meet the requirements for the review and assessment of information relevant to safety by the regulatory body (see GSR Part 1 (Rev. 1) [18]), the operating organization is required to submit to the

regulatory body, in a timely manner, any information that the regulatory body has requested. The operating organization is required to make arrangements with vendors to ensure the availability of information that has been requested by the regulatory body. The operating organization is also required to keep the regulatory body informed of relevant new information and of any changes to information submitted previously; see Requirement 83 of SSR-3 [1].

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

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

2.12. The operating organization should revise all documentation associated with any modification or activity that might affect the safety of a research reactor (and all documentation having an indirect but significant influence on the safety of the research reactor), as appropriate. The revised documentation is required to be submitted to the regulatory body for its review and assessment (see GSR Part 1 (Rev. 1) [18], para. 4.45), with the potential magnitude and nature of the associated hazards being taken into account (see GSR Part 1 (Rev. 1) [18], Requirement 26).

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

(a) Understand the reactor design, the safety concepts on which it is based, and the management system and the approach to operational safety proposed by the operating organization.

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

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

2.14. The primary basis for the review and assessment of the safety aspects of the proposed research reactor is the information contained in the safety analysis report submitted by the operating organization

to the regulatory body. The safety analysis report should provide sufficient information for the regulatory body to decide on the following points:

— Whether the operating organization has provided the information that is both necessary and adequate for the purpose and scope of the review and assessment (see para. 4.2).

— Whether this information is in compliance with all applicable regulatory requirements.

— Whether this information is accurate. This might be determined by means of independent checks of the design, including calculations, and by inspections of ~~the programmes and facilities~~ the operating organization's facilities and management system.

ACCEPTANCE CRITERIA

2.15. In addition to the acceptance criteria established within the regulatory framework, the operating organization should develop additional acceptance criteria to demonstrate adequate application of the principles and objectives of safe design and operation established in the IAEA safety standards, including application of requirements for radiation protection (see IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [23] and IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [24]).

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

— A set of numerical limits on the values of predicted parameters;

— A set of conditions for facility states during and after an accident;

— A set of performance requirements on systems, including safety features for design extension conditions (see Requirement 22 of SSR-3 [1]);

— A set of requirements on the need for, and the ability to credit, actions by the operating organization, including, for design extension conditions, protective measures that are limited in terms of times and areas of application.

2.17. Acceptance criteria may be specified as 'basic acceptance criteria' or 'specific acceptance criteria'. Basic acceptance criteria are aimed at achieving an adequate level of defence in depth. Examples include maximum allowable doses to the public ~~or~~ and the prevention of fuel failure [~~24~~25]. Specific acceptance criteria should include additional margins beyond the basic acceptance criteria as established within the regulatory framework, to allow for uncertainties. Specific acceptance criteria may be proposed by the operating organization and should be satisfactory to the regulatory body.

2.18. Acceptance criteria for design extension conditions without significant core degradation should be defined to ensure, with an adequate level of confidence, that core melting can be prevented, that there

are adequate margins to avoid cliff edge effects[5] and there is no, or only minor, off-site radiological impact. In accordance with para 6.68 of SSR-3 [1], conditions that could lead to an early radioactive release or large radioactive releases[6] are required to be practically eliminated, ~~and so~~ acceptance criteria for design extension conditions with core melting should be defined in a way that ensures mitigation of radiological consequences, as far as reasonably practicable. The analysis of design extension conditions might lead to the implementation of additional safety features, or the extension of the capability of safety systems to fulfill the main safety functions and to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material confined in the facility, including radioactive material resulting from degradation of the reactor core.

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

(a) Radiological criteria such as:

— Dose limits and design target doses (see NS-G-4.6 [~~1~~8]) for public exposure;

— Dose limits and design target doses (see NS-G-4.6 [~~1~~8]) for occupational exposure;

— Radiation levels for accident conditions and for lifesaving actions in an emergency, consistent with Requirements 11 and 24 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [25]);

— Authorized limits on radioactive discharges to the environment during normal operation and acceptable limits on radioactive releases to the environment in accident conditions;

— Risk criteria (where applicable).

(b) Nuclear fuel performance criteria:

— Maximum cladding temperature below blistering temperature;

— Maximum heat flux not exceeding the critical heat flux during a transient;

— Maximum heat flux not exceeding the onset of significant voiding during a transient;

— Flow conditions not exceeding the onset of flow instability;

— Maximum fuel and fuel cladding temperature below failure;

— Other limits to prevent significant damage to fuel and fuel cladding failures;

(c) Performance criteria, including the following:

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

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

— Limits on parameters to prevent damage of the primary coolant system boundary;

— Limits on parameters to prevent damage to systems important to safety caused by in-core or out of core experimental facilities;

— Limits on parameters to prevent damage to the containment systems;

— Maintenance of core cooling;

— Limits on parameters to prevent reactivity accidents;

— Frequency limits for anticipated operational occurrences and for accident conditions.

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

2.21. The acceptance criteria should include the following:

— An event should not generate a more serious condition of the research reactor without the occurrence of a further independent failure. Thus, an anticipated operational occurrence by itself should not generate a design basis accident; a design basis accident by itself should not generate design extension conditions.

— There should be no consequential loss of function of the safety systems necessary to mitigate the consequences of an accident.

— Systems used for mitigation of the consequences of accidents should be designed and constructed in accordance with their importance to safety, to withstand the maximum loads and stresses and the most extreme environmental conditions for the accident analysed.

INFORMATION REQUIREMENTS AT THE VARIOUS STAGES OF THE AUTHORIZATION PROCESS

2.22. The operating organization should provide the regulatory body with all relevant information on the safety of the research reactor. This information is normally presented in a safety analysis report, which is described comprehensively in the Appendix to this Safety Guide. Recommendations on the preparation and presentation of the safety analysis report are provided in Section 3, and recommendations on its review and assessment are provided in Section 4. The following paragraphs provide a summary of the information that is normally required for each stage of the authorization process. Sequential requests for information might lead to successive updating, with each version of the safety analysis report corresponding to a particular stage of the authorization process (see para. 3.4 of SSR-3 [1]).

2.23. The preparation of the safety analysis report should start as early as possible in the project, to allow the designers to derive the maximum benefit from the safety analysis, as well as to allow the

regulatory body to become familiar with the design and the safety features of the reactor. The amount of information provided in the safety analysis report, at each stage, should be sufficient to allow both the operating organization and the regulatory body to make a decision on the acceptability of the reactor for that stage.

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

2.25. For research reactors with low hazard potential, particularly critical assemblies and subcritical assemblies, the amount of information and analysis to be provided according to paras 2.26 to 2.49 could be reduced in accordance with a graded approach.

SUBMISSION OF INFORMATION TO THE REGULATORY BODY

Schedule for the submission of information

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

Siting and site evaluation

2.27. The operating organization should provide sufficient information commensurate with the type, complexity and hazards associated with the research reactor to demonstrate to the regulatory body that the proposed site is suitable for the type and design of the proposed research reactor. Difficulties that will need to be resolved during the subsequent stages of the authorization process should be identified. Information on the site itself, and preliminary information on the research reactor and its interaction with the site and the surrounding environment, should be provided. In addition, a preliminary statement on the potential radiological impacts on site personnel, on the population in the surrounding area and on the environment should be provided. ~~If required in the State, a~~ A radiological environmental impact assessment should be performed as a part of the authorization process; see GSG-10 [26].

2.28. The characteristics of the site that might affect the safety of the research reactor should be investigated and assessed by the operating organization (See requirement 8 of GSR Part 4 (Rev.1) [13]). The objective of the assessment should be to assess how the site characteristics would influence the design and operation of the research reactor and to demonstrate the adequacy of the characteristics of the site from the point of view of safety. The requirements for the initial site evaluation and site selection, the general criteria for site evaluation and the external events that should be considered for site

evaluation are provided in section 5 of SSR-3 [1]. Additional recommendations on siting and site evaluation are provided in the Appendix of this Safety Guide (see Chapter 3: Site characteristics) and in SSG-12 [~~10~~19] and requirements on site evaluation are established in IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [27]. Nuclear security issues and their interface with safety should also be considered at the siting and site evaluation stage; see Ref. [28]. The details on siting that should be addressed in the safety analysis report are presented under Chapter 3 of the Appendix.

Design and construction

2.29. Before authorization of the construction of the research reactor, features such as the physical layout and the type of construction of the research reactor, as well as the key elements of the construction process, should be carefully considered, and their effects on the safety of the research reactor throughout its lifetime should be assessed. At this stage, due consideration should be given to the selection of materials, to ageing mechanisms for materials and structures, systems and components, and to the effects of these ageing mechanisms on safety; additional recommendations on ageing management are provided in IAEA Safety Standards Series No. SSG-10, Ageing Management for Research Reactors [17]. Consideration should also be given to nuclear security, including physical protection [29,~~–~~30]~~–~~, information security [17] and the interface of nuclear security with safety. The operating organization should describe the arrangements for the control of activities in construction, manufacture and installation. The initial decommissioning plan prepared at the design stage should cover issues such as strategies to be applied, radiation doses to be expected and amounts of waste expected to be generated; see SSG-47 [16]. Information on the matters addressed in this paragraph should be submitted to the regulatory body for review and assessment.

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

2.31. Those aspects of the design that should be submitted to the regulatory body for review and assessment before the design is finalized should be identified in agreement with the regulatory body ~~so that activities can proceed while the reactor is under construction~~. The information should be updated and resubmitted to the regulatory body as the detailed design ~~and the construction~~ of the reactor proceed. In some cases, revised versions of documents will be sufficient; in other cases, technical supplements may be appropriate. Additional recommendations on the authorization process for this stage are provided in SSG-12 [~~10~~19].

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

Commissioning

2.33. When construction is at a sufficiently advanced stage, the information contained in the safety analysis report should be reviewed and updated, where necessary. The updated safety analysis report should be resubmitted to the regulatory body for review and assessment in order to obtain the required authorization for commissioning.

2.34. Paragraph 7.52 of SSR-3 [1] states "Commissioning tests shall be arranged in functional groups and in a logical sequence. This sequence includes pre-operational tests, initial criticality tests, low power tests, and power ascension and power tests". Every test sequence should be completed successfully, and the results should be submitted to the regulatory body for review and assessment. The test results should be made subject to approval by the operating organization at the appropriate level of management and, as necessary depending on the direct effect on safety, by the regulatory body before the subsequent test sequence is started. Paragraph 7.52 of SSR- 3 [1] further states:

> "The commissioning programme shall … be divided into stages, which are usually arranged in the following sequence:

— Stage A: tests prior to fuel loading;

— Stage B: fuel loading tests, initial criticality tests, low power tests;

— Stage C: power ascension tests and power tests".

Commissioning should be carried out in accordance with the commissioning programme that has been reviewed and assessed by the regulatory body. Detailed recommendations on the commissioning of research reactors are provided in SSG-12 [19] and NS-G-4.1 [3].

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

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

2.37. The commissioning procedures for a particular commissioning stage should be reviewed before the start of that commissioning stage and should be updated, as necessary. The 'as-built' design of the research reactor and the results of the previous commissioning stages should be taken into account. The

updated commissioning procedures should be submitted to the regulatory body for review and assessment to obtain the required authorizations for commissioning.

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

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

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

2.41. The information referred to in paras 2.34–2.40 should be updated after each commissioning stage, and submission to the regulatory body should form the basis of the start of the next commissioning stage as a part of the authorization process.

Operation

2.42. In its application for an operating licence, the operating organization should submit all the information referred to in the preceding sections. Additional information to prove the capability for safe operation should be submitted to the regulatory body. Some of this information will be required at the different stages of the authorization process, and some information should be submitted after the formal licence has been obtained. Additional recommendations on the authorization stages are provided in SSG-12 [19].

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

~~2.43.~~2.44. Systematic periodic safety reviews of the research reactor are required to be performed throughout its lifetime (see para. 4.25 of SSR-3 [1]). Such periodic reviews of the safety of the research reactor include periodic safety reviews required by the regulatory body (see paras 7.121 and 7.122 of SSR-3 [1]) and self-assessments performed by the operating organization. Such reviews should address important issues such as the cumulative effects of ageing of the research reactor. The nature of such reviews and the interval between reviews should reflect the risks that the research reactor presents. For such reviews, a comparison of the existing safety analysis report with operating experience should be made, including operating experience from accidents and information on radiation protection, modifications, experiments and other aspects of operation. If required as a result of a periodic safety review, the operating organization should submit to the regulatory body a request for an amendment of the licence, which should include a revised safety analysis report, as appropriate.

Utilization and modification

~~2.44.~~2.45. The operating organization should submit to the regulatory body for review and assessment information on experiments and modifications that might affect the safety of the research reactor. The specific requirements for submission will depend on the safety significance of the experiments and modifications. These requirements are set out in Requirement 83 of SSR-3 [1]. Specific recommendations on the development of appropriate procedures for the control of experiments and modifications, including review by the safety committee, are provided in SSG-24 [2] and NS-G-4.4 [6].

~~2.45.~~2.46. Experiments and modifications having major safety significance should be categorized and subjected to procedures for design, construction, commissioning and safety analysis that are equivalent to those for the research reactor itself (see Requirement 83 of SSR-3 [1]). This safety analysis may need to be performed in stages according to the various phases of the modification project or utilization activity. These stages could be: pre-implementation (project initiation, project definition, design); implementation (fabrication, installation, commissioning); and post implementation (operation, surveillance). Further recommendations on utilization and modification, including categorization of experiments and modifications, are provided in SSG-24 [2]. The safety aspects of each phase of the modification project should be analysed and presented in a separate safety analysis report for the modification or utilization activity, or the appropriate chapters of the existing safety analysis report for the research reactor should be revised and submitted to the regulatory body for review and assessment. In addition, the safety analysis report provides boundaries for operational limits and conditions that have been demonstrated to be safe, and any experiments and modifications should fall within these boundaries.

~~2.46.~~2.47. If applicable, the operating organization should revise the relevant acceptance criteria and should submit them to the regulatory body for review and assessment, and for approval for them to be used in the safety analysis of the proposed experiment or modification.

~~2.47.~~2.48. Commissioning of the experiment or the modified research reactor should be conducted to demonstrate compliance with the design requirements in the safety analysis report. In addition, if changes to the safety analysis report or to some analyses are made, it should be ensured that all other safety analyses are still valid.

Decommissioning and release from regulatory control

~~2.48.~~2.49. The operating organization should submit the decommissioning plan and supporting documents to the regulatory body for review and assessment. The type of information and level of detail in the decommissioning plan, including safety assessment, depends upon the hazards associated with the decommissioning of the research reactor. Requirements on decommissioning of research reactors are established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [31].

~~2.49.~~2.50. At some point in the decommissioning process (e.g. after the removal of all fuel from the site), the safety analysis report ceases to be a major working document and a safety assessment for decommissioning, commensurate with the remaining hazards, should be prepared by the operating organization. Further recommendations on decommissioning are provided in SSG-47~~s~~ [16] and recommendations on release from regulatory control are provided in IAEA Safety Standards Series No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices [32].

3. PREPARATION OF THE SAFETY ANALYSIS REPORT

PURPOSE AND SCOPE OF THE SAFETY ANALYSIS REPORT

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

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

— To aid the designer in confirming that individual systems are integrated correctly with one another, since the design of the research reactor and the development of the safety analysis report are complementary and interactive processes;

— To ensure that the safety analysis has properly identified the safety issues relevant to the design and that the safety analysis and the design are consistent;

— To aid in the understanding of the relevant design criteria, limitations and requirements, and to aid in the evaluation of the hazards posed by the research reactor in operational states and accident conditions;

— To aid in the training and retraining of operating personnel and in their familiarization with the research reactor;

— To aid in the establishment of operational limits and conditions on certain parameters that have to be met at all stages of the lifetime of the reactor in order to ensure adequate margins of safety for the reactor;

— To aid in the establishment of the programme for maintenance, periodic testing and inspection and operating procedures;

— To identify ageing mechanisms and their effects on safety for the development of an ageing management programme;

— To aid in the understanding of the interface between safety and nuclear security;

— To aid the development and establishment of the decommissioning plan;

— To aid in the development and establishment of the emergency preparedness and response arrangements.

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

— The evolution of the design, operation and utilization of the research reactor and the associated experimental facilities and devices, and any safety significant modifications to the research reactor;

— Changes made in the research reactor and its operation as a result of events that might have occurred during the lifetime of the research reactor or operating experience feedback, including from other nuclear installations and that might influence the actions that will need to be taken in decommissioning the research reactor.

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

3.5. In accordance with para. 3.7 of SSR-3 [1], the safety analysis report is required to form the basis for the operational limits and conditions of the reactor. The operational limits and conditions should be incorporated into the authorization for operation. As a minimum, the safety analysis report should describe the content of the operational limits and conditions if they will be described in detail in a separate document. The safety analysis report is also required to provide details of the conduct of operations intended by the operating organization, including its organization and the management system procedures established for the design and operation of the research reactor. The safety analysis report is also required to provide information on the emergency arrangements for the research reactor, and it should set out the design provisions and operating procedures relating to decommissioning.

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

SPECIFIC RECOMMENDATIONS ON THE SAFETY ANALYSIS REPORT

3.7. In accordance with Requirement 5 of SSR-3 [1], the operating organization is required to ensure that an independent verification of the safety assessment is performed by individuals or groups separate from those carrying out the design. The independent verification should be carried out under the responsibility of the operating organization by a team of experts who are independent of the designers

and of those performing the safety assessment and who have not participated in any part of the design or the safety assessment. This verification should be conducted either by the operating organization or by another qualified organization on its behalf (see paras 4.64, 4.66 and 4.67 of GSR Part 4 (Rev. 1) [13]). Irrespective of the process followed for the development and verification of the safety analysis, the operating organization remains responsible for the content, comprehensiveness and quality of the safety analysis. ~~(~~see Requirement 2 in SSR-3 [1]).This independent verification should be performed before the safety analysis report is submitted to the regulatory body for review and assessment as a part of the authorization process.

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

3.9. The proposal for a research reactor and the application for an authorization should be subject to open public participation by means of regular meetings, formal hearings or other appropriate means of communication. For these purposes, the operating organization may have to develop a non-technical version of the safety analysis report that can be understood by the public, considering confidentiality aspects (see para. 3.12). Recommendations on public participation are provided in SSG-12 [19].

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

> "The safety analysis report shall cite references that may be necessary for its thorough review and assessment. This reference material shall be readily available to the regulatory body and shall not be subject to any classification or limitation that would prevent its adequate review and assessment".

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

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

3.12. Certain information provided by the operating organization or its contractors should be considered confidential, because of its proprietary nature, for security reasons or because of the right of individuals to privacy, in accordance with national laws and regulations. Such confidential information should be made available, as necessary, without restriction to the regulatory body; that is, it should be made available to its staff, technical support organizations, consultants and advisory committees as well

as to any governmental bodies involved in the review and assessment process. The regulatory body should formally inform the operating organization which consultants and advisers will be involved on behalf of the regulatory body. Those persons to whom such information is to be entrusted should be advised of its confidential nature and should be obliged, consistent with national laws and regulations, to protect its confidentiality. If consultants, technical support organizations and external advisory committees need to have confidential documents at their disposal, a process to ensure confidentiality should be put in place.

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

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

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

DEVELOPMENT OF THE SAFETY ANALYSIS

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

General considerations

3.17. ~~3.26.~~ The general requirements in the development of the safety analysis are established in SSR-3 [1]. To ensure that the safety analysis meets the intended objective, the detailed recommendations on the preparation of the safety analysis as presented in the Appendix to this Safety Guide (Chapter 16: Safety analysis) should be taken into account.

~~3.16.~~3.18. ~~3.27.~~ The safety analysis should identify design basis accidents and design extension conditions without significant fuel degradation and design extension conditions with melting of the reactor core. In addition, accidents that have more severe consequences should also be analysed for purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident.

~~3.17.~~3.19. The safety analysis should be used mainly to enable the operating personnel to understand the basis for the safe operation of the reactor, and to demonstrate to the regulatory body the way in which the design of the research reactor and the related operational and emergency procedures will contribute to the prevention of accidents or mitigation of the consequences of accidents. The safety analysis should include analyses of the response of the reactor to the postulated initiating events listed in para. 3.2~~3~~1. The safety analysis should also serve as a basis for the determination of the operational limits and conditions, the safety classification of the structures, systems and components, and the development of the accident management procedures and the emergency plan.

~~3.18.~~3.20. The consideration of accident conditions should determine the design of the research reactor and the design limits for the safety systems and for most structures, systems and components necessary for the operation of the research reactor. The accident conditions should also be considered in the operating instructions and procedures for operating personnel. In addition, the potential radiological consequences of accident conditions for workers, the public and the environment is typically more severe than the radiological consequences of normal operation. For this reason, an important part of the effort in the peer review and verification by the operating organization should be directed to the safety analysis of accident conditions. The scope and extent of this analysis should be in accordance with the magnitude and nature of the hazards associated with the particular research reactor. Safety analysis may be considered to consist of the following major steps:

— Identification and selection of the postulated initiating events;

— Categorization of the postulated initiating events;

— Determination of enveloping postulated initiating events;

— Evaluation of the sequence of events, development of the postulated initiating events in relation to system responses and their consequences;

— Comparison against acceptance criteria.

~~3.19.~~3.21. The following aspects should be verified in the safety analysis:

— That sufficient defence in depth has been provided, that the levels of defence are independent and preserved to the extent possible and that the potential accident sequences are arrested as early as possible.

— That the research reactor can withstand the physical and environmental conditions that it would experience. These would include extreme environmental conditions and other extreme conditions.

— That human factors and human performance issues have been adequately addressed (see IAEA Safety Standard Series No, SSG-51, Human Factor Engineering in the design of Nuclear Power Plants [41]).

— That long term ageing mechanisms that could detract from the reliability of structures, systems and components over their design life are identified, monitored and managed (i.e. by upgrading, refurbishment or replacement), so that safety is not affected and risks do not unacceptably increase.

Identification, categorization and grouping of postulated initiating events

~~3.20.~~3.22. The identification and selection of the postulated initiating events should be the first step of the safety analysis. The selection method used should be systematic and should be auditable. Moreover, as complete as possible a listing of postulated initiating events should be established. An important feature of the review and assessment process should be to consider whether the method of identification of postulated initiating events meets these recommendations and whether the list of postulated initiating events is acceptable as the basis for the safety analysis. The use of systematic techniques such as hazard and operability (HAZOP) studies or failure modes and effects analysis (FMEA), among others could facilitate the selection process.

~~3.21.~~3.23. Typical examples of postulated initiating events leading to event sequences categorized as anticipated operational occurrences, design basis accidents and design extension conditions include those given below, sorted by types of sequence. This list is broadly indicative. The actual list will depend on the type of reactor, actual design and potential hazards associated with the research reactor[7]. The list of selected postulated initiating events is taken from appendix I of SSR-3 [1]:

(1) Loss of electrical power supplies:

— Loss of normal electrical power.[8]

(2) Insertion of excess reactivity:

— Criticality during fuel handling and loading (e.g. due to an error in fuel insertion, dropping of fuel assembly on core);

— Startup accident;

— Control rod failure or control rod follower failure;

— Control drive failure or control drive system failure;

— Failure of other reactivity control devices (such as a moderator or reflector);

— Unbalanced rod positions;

[7] Depending on national regulation and interface arrangements with nuclear security, the list may be complemented by relevant nuclear security events.

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

— Failure or collapse of structural components;

— Insertion of cold or hot water;

— Changes in the moderator material (e.g. voids or leakage of D_2O into H_2O systems or leakage of H_2O into D_2O system);

— Effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material, error in loading or unloading);

— Insufficient shutdown reactivity margin;

— Inadvertent ejection of control rods;

— Maintenance errors with reactivity devices;

— Spurious control system signals;

— Removal of poisons from the coolant or moderator.

(3) Loss of flow:

— Primary pump failure;

— Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger);

— Effect of the failure or mishandling of an experiment;

— Rupture of the primary coolant boundary leading to a loss of flow;

— Fuel channel blockage or flow reduction (e.g. due to foreign material);

— Improper power distribution due to, for example, unbalanced rod positions, in core experiments or in fuel loading (power-flow mismatch);

— Reduction in coolant flow due to bypassing of the core;

— Deviation of system pressure from specified limits;

— Loss of heat sink (e.g. due to the failure of a valve or a pump, or a system rupture).

(4) Loss of coolant:

— Rupture of the primary coolant boundary;

— Damaged pool;

— Pump down of the pool;

— Failure of beam tubes or other penetrations.

(5) Erroneous handling or failure of equipment or components:

— Failure of the cladding of a fuel element;

— Mechanical damage to core or fuel (e.g. mishandling of fuel or dropping of a transfer flask onto the fuel);

— Failure of the engineered safety features (e.g. emergency core cooling system);

— Malfunction of the reactor power control;

— Criticality in fuel in storage;

— Failure of the means of confinement, including the ventilation system;

— Loss of coolant to fuel in transfer or storage;

— Loss or reduction of proper shielding;

— Failure of experimental apparatus or material (e.g. loop rupture);

— Exceeding of fuel ratings.

(6) Special internal events:

— Internal fires or explosions, including internally generated missiles;

— Internal flooding;

— Loss of support systems;

— Security related incidents;

— Malfunction in reactor experiments;

— Improper access by persons to restricted areas;

— Fluid jets and pipe whip;

— Exothermic chemical reactions;

— Drop of heavy loads;

(7) External events[9]:

— Earthquakes (including seismically induced faulting and landslides);

— Flooding (including failure of an upstream or downstream dam and blockage of a river and damage due to a tsunami or high waves);

— Tornadoes and tornado missiles;

— Sandstorms;

— Hurricanes, and ~~snow~~ storms and lightening;

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

— Tropical cyclones;

— Electromagnetic interference (e.g. from solar events);

— Explosions;

— Aircraft crashes;

— Fires;

— Toxic spills;

— Accidents on transport routes (including collisions into the research reactor building);

— Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);

— Biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects;

— Extreme meteorological phenomena;

— Lightning strikes;

— Power or voltage surges on the external supply line;

(8) Human errors.

~~3.22.~~3.24. The list of postulated initiating events specified in para 3.2~~3~~1 should be reviewed for applicability for subcritical assemblies. The resulting list of postulated initiating events should be justified and documented for the specific configuration of the research reactor. For example, the following postulated initiating events might not be applicable to some subcritical assemblies, depending on their specific design features:

(1) Loss of electrical power supplies,

(2) Insertion of excess reactivity:

— Control rod failure or control rod follower failure;

— Control drive failure or control drive system failure;

— Failure of other reactivity control devices (such as a moderator or reflector);

— Unbalanced rod positions;

— Insertion of cold or hot water;

— Maintenance errors with reactivity devices;

— Spurious control system signals.

— Removal of poisons from the coolant or moderator

(3) Loss of flow.

(4) Loss of coolant:

— Pump-down of the pool.

(5) Erroneous handling or failure of equipment or components:

— Failure of the emergency core cooling system;

— Malfunction of the reactor power control;

— Loss of coolant to fuel in transfer or storage;

— Exceeding of fuel ratings.

(6) Special internal events:

— Fluid jets and pipe whip.

3.25. ~~3.30.~~ The categorization of postulated initiating events should be developed on the basis of initiating frequency, likelihood of system recovery and potential consequences of a postulated initiating event, to determine the following:

(a) Postulated initiating events that are likely to occur during the lifetime of a research reactor but that do not lead to accident conditions (i.e. they could lead to anticipated operational occurrences) should be analysed to show that the research reactor has a sufficient safety margin to comply with the acceptance criteria for such events. Such a safety margin may be present due to the provision of specific safety systems and engineered safety features in the design and the establishment of operating procedures to (i) restore the safe state, and (ii) prevent or minimize damage.

(b) Postulated initiating events of low likelihood that reflect the specific characteristics of the design and could lead to an accident (design basis accidents) should be analysed to show that the research reactor has an adequate safety margin to comply the acceptance criteria for such events. Such a safety margin may be present due to the provision of safety systems or engineered safety features or because of an inherent behaviour of the reactor tending to (i) restore the safe state, (ii) prevent the release of radioactive material and (iii) limit any release to an acceptably low level.

~~3.23.~~3.26. Postulated initiating events should be grouped in accordance with the expected frequencies of the initiating events and should be clearly assigned to different plant states. The purpose of this categorization is:

— To justify the basis for the range of events under consideration;

— To reduce the number of initiating events necessitating detailed analysis to a set that includes the enveloping cases in each of the various event groups credited in the safety analysis but that does not contain different events that are associated with identical system performance (such as events that are identical in terms of timing, response of plant systems and radiological release fractions);

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

~~3.24.~~3.27. Both internal and external postulated initiating events of all types, for all operational states, including shutdown and fuel loading, should be considered in this process of event grouping. The process of event grouping should lead to a list of enveloping postulated initiating events to be analysed. Failures in other systems such as experimental facilities, failures in the availability of off-site power or the total loss of off-site power, and failures in spent fuel storage and in storage tanks for radioactive liquids should also be considered as postulated initiating events.

~~3.25.~~3.28. In the selection and grouping of postulated initiating events for the analysis, the list given in para. 3.23~~1~~ should form the basis of the postulated initiating events to be considered. Considerations on the methodology that can be used are given in Annex I to this Safety Guide. Annex I also sets out considerations for analyses of the event sequences triggered by the postulated initiating events and for analyses of external events and internal events. In particular, the analyses should clearly identify the assumed input parameters and initial conditions. These assumed input parameters and initial conditions should be presented in the safety analysis report and will provide the basis for the determination of the operational limits and conditions. Annex II to this Safety Guide gives examples of these parameters.

~~3.26.~~3.29. ~~The general requirements in the development of the safety analysis are established in SSR-3 [1]. To ensure that the safety analysis meets the intended objective, the detailed recommendations on the preparation of the safety analysis as presented in the Appendix to this Safety Guide (Chapter 16: Safety analysis) should be taken into account.~~

~~3.30. The safety analysis should identify design basis accidents and design extension conditions without significant fuel degradation and design extension conditions with melting of the reactor core. In addition, accidents that have more severe consequences should also be analysed for purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident.~~

Deterministic and Probabilistic Methods

~~3.27.~~3.31. Annex I deals mainly with deterministic methods, which are normally used for safety assessments of research reactors. Deterministic techniques for anticipated operational occurrences and design basis accidents are characterized by conservatism and are based on defined sets of rules for event selection, analytical methods, and parameter specification and acceptance criteria. For design extension conditions, best estimate methods with realistic boundary conditions can be applied. Through the use of these methods, reasonable assurance is provided that the ultimate objective of preventing or limiting the release of radioactive material can be achieved without the need to perform complex calculations, because these methods tend to overestimate the radioactive releases. The most severe of these releases are taken into account in the selection of a site or in setting design requirements for engineered safety

features for the research reactor. The choice of these accidents is based on experience and engineering judgement, without the need for determining the probabilities of the event sequences.

~~3.28.~~3.32. Probabilistic techniques could be used to supplement the above-mentioned safety assessments (See requirement 15 of GSR Part 4 (Rev.1) [13]). Probabilistic methodologies use the assumption that all accidents are possible and that any number of simultaneous failures might occur, although the probabilities might be very low. Some postulated accidents or combinations of accidents might have less severe consequences than the postulated accidents used in the deterministic methodology. However, when they are weighted by their likelihood, they might represent a significant risk and might impose different demands on the design. In addition, the deterministic approach has difficulties in effectively treating system interdependences (e.g. common cause failure), which probabilistic methods can address analytically and quantitatively. Application of probabilistic techniques also leads to significant improvements in the understanding of system behaviour and interactions, and of the role of operating personnel in accident conditions. Probabilistic techniques might be appropriate for some specific cases, which could be discussed between the operating organization and the regulatory body.

~~3.29.~~3.33. ~~The categorization of postulated initiating events should be developed on the basis of initiating frequency, likelihood of system recovery and potential consequences of a postulated initiating event, to determine the following:~~

~~(a) Postulated initiating events that are likely to occur during the lifetime of a research reactor but that do not lead to accident conditions (i.e. they could lead to anticipated operational occurrences) should be analysed to show that the research reactor has a sufficient safety margin to comply with the acceptance criteria for such events. Such a safety margin may be present due to the provision of specific safety systems and engineered safety features in the design and the establishment of operating procedures to (i) restore the safe state, and (ii) prevent or minimize damage.~~

~~(b) Postulated initiating events of low likelihood that reflect the specific characteristics of the design and could lead to an accident (design basis accidents) should be analysed to show that the research reactor has an adequate safety margin to comply the acceptance criteria for such events. Such a safety margin may be present due to the provision of safety systems or engineered safety features or because of an inherent behaviour of the reactor tending to (i) restore the safe state, (ii) prevent the release of radioactive material and (iii) limit any release to an acceptably low level.~~

~~3.30.~~3.34. ~~The results of the safety analysis of the research reactor should be reflected in the safety analysis report by taking into account the recommendations provided in the Appendix to this Safety Guide (Chapter 16: Safety analysis). The recommendations provided regarding Chapter 16 of the safety analysis report also include the comparison of the results with the acceptance criteria to determine the acceptability of the research reactor.~~

Consideration of design extension conditions in the safety analysis

~~3.31.~~3.35. Design extension conditions include events more severe than design basis accidents that originate from extreme events or combination of them that could cause damage to structures, systems and components important to safety or challenges the fulfillment of main safety functions, as well as progressions of events that could lead to reactor core damage or a radioactive release. Examples of design extension conditions that are applicable to research reactors can be found in Ref. [33]. The analysis of design extension conditions should be performed with best estimate codes, models and initial and boundary conditions to demonstrate that core melting can be prevented or mitigated with an adequate level of confidence and there are adequate margins to avoid any cliff edge effects.

~~3.32.~~3.36. The analysis of design extension conditions, including assessment of the response of the research reactor to those conditions, should demonstrate that the design of the research reactor is adequate to prevent accident conditions or to mitigate their consequences as far as reasonably practicable. The results of the analysis may indicate the need for additional safety features for design extension conditions, or extension of the capability of safety systems, to fulfill the main safety functions and to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material confined within the research reactor facility, including radioactive material resulting from degradation of the reactor core.

~~3.33.~~3.37. The analysis of design extension conditions should also demonstrate the following:

- The reactor can be brought into the state in which the confinement function can be maintained in the long term;
- The structures, systems and components are capable of avoiding an early radioactive release or a large radioactive release;
- Control locations remain habitable to allow the performance of necessary actions.

In addition, it should be demonstrated that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated. Additional accidents that are postulated for the purposes of emergency preparedness and response should also be analysed.

~~3.34.~~3.38. The analysis should address the impact of the most challenging conditions and should demonstrate that the compliance with acceptance criteria is achieved by safety features for design extension conditions implemented in the design, combined with the implementation of procedures or guidelines for accident management.

3.39. Paragraph 6.66 of SSR-3 [1] states “For subcritical assemblies, the likelihood of criticality shall be sufficiently remote to be considered as a design extension condition”. The potential for accidental criticality should be analysed to demonstrate compliance with acceptance criteria, to ensure adequate margins to avoid any cliff edge and to identify additional safety features for design extension conditions, or extension of the capabilities of safety systems, to prevent or mitigate the consequences of such an event.

Summary of Results

3.40. ~~3.31.~~ The results of the safety analysis of the research reactor should be reflected in the safety analysis report by taking into account the recommendations provided in the Appendix to this Safety Guide (Chapter 16: Safety analysis). The recommendations provided regarding Chapter 16 of the safety analysis report also include the comparison of the results with the acceptance criteria to determine the acceptability of the research reactor.

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

PURPOSE AND SCOPE OF THE REVIEW AND ASSESSMENT

4.1. IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [18] establishes requirements for the review and assessment process, by the regulatory body, of information submitted by the operating organization to demonstrate the safety of the research reactor. Paragraphs 3.149 to 3.209 of IAEA Safety Standards Series No. GSG-13, Functions and Processes of the Regulatory Body for Safety [34] provide~~s~~ recommendations for the regulatory body on meeting these requirements. Review and assessment are undertaken to enable the regulatory body to make a decision or a series of decisions on the acceptability of the research reactor in terms of safety. The review and assessment process consists of examining the submissions of the operating organization on all aspects relating to the safety of the research reactor. It includes consideration of both normal operation and failures, and of events, including malfunction related to organizational or human errors, that have the potential to cause exposure of site personnel or the public, or radiological hazards to the environment. This safety analysis should be complete and should cover all the postulated 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 also be coordinated with inspections on the site and elsewhere to verify the claims made in the submissions. The operating organization might have had external peer reviews conducted at the research reactor by national bodies or international organizations. The results of such reviews could provide the regulatory body with additional insights into the safety of the research reactor.

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

(a) To determine whether the site is suitable for the type, power and use of the proposed research reactor;

(b) To determine, before construction, whether the proposed research reactor design meets the regulatory body's requirements, and to impose any further requirements or conditions that may be deemed necessary by the regulatory body;

(c) To determine whether the operating organization has the necessary ability, reliability, resources, organizational structure and competent personnel to meet the regulatory requirements;

(d) To determine whether the construction remains consistent with the requirements of the regulatory body;

(e) To determine whether the commissioning programme is adequate and whether its results demonstrate that the design requirements have been met;

(f) To determine whether the operational limits and conditions are established in accordance with the regulatory requirements and whether an adequate level of operational safety can be ensured, including the provisions made for accident conditions;

(g) To determine whether the experiment or the modification of the research reactor meet the requirements of the regulatory body;

(h) To determine whether the decommissioning programme meets the requirements of the regulatory body.

PROGRAMME FOR REVIEW AND ASSESSMENT

4.3. The operating organization should discuss with the regulatory body at an early stage the programme for review and assessment (including a schedule for submission of documents), which should be established by the regulatory body. The programme for review and assessment should take into account the stages of the authorization process as described in para. 2.5 and paras 2.26–2.49.

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

4.5. A major feature of the submission by the operating organization will be its analysis of normal operation as well as its analyses of deviations from normal operation. The safety of the research reactor is required to be based on proven engineering practices and adequate management for safety, and the safety analysis is a confirmation of the adequacy of the engineering and management and not a substitute for them. The value of safety analysis is in extending knowledge about and understanding of the research reactor and its behaviour, and in identifying shortcomings and areas in which safety can be improved.

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

(a) Documentation establishing 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 design of the proposed research reactor, to confirm that it will meet the safety requirements, including requirements for occupational health and requirements for fire safety;

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

(e) The design features of the nuclear security system (including physical protection and information security) that are important to safety;

(f) Information necessary for verification of the design.

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

(a) The as built design of the reactor;

(b) The commissioning programme for Stage A;

(c) The operational limits and conditions for Stage A commissioning;

(d) The system for control of records and reports;

(e) The management system, the organizational structure and programme for operation.

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

(a) The records of the results of the previous commissioning stage, including non-conformances and, where appropriate, their associated corrective actions;

(b) The revisions to the commissioning programme for Stage B, if any;

(c) The operational limits and conditions for Stage B commissioning;

(d) The radiation protection programme;

(e) The operating instructions, operating procedures, emergency procedures and administrative rules;

(f) The system for control of records and reports;

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

(h) Documentation on occupational health and safety and fire safety;

(i) The management system, the organizational structure and programme for operation;

(j) The emergency plan;

(k) The system of accounting for and control of nuclear material and radioactive material;

(l) The security plan.

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

(a) The records and results of the commissioning tests of Stage B, including non-conformances and, where appropriate, their associated corrective actions;

(b) The revisions to the commissioning programme, if any;

(c) The operational limits and conditions for Stage C commissioning;

(d) Any revised arrangements.

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

(a) The records and results of commissioning tests of Stage C, including non-conformances and, where appropriate, their associated corrective actions;

(b) Verification that the radiation dose rates in the reactor are as expected and verification of the adequacy of the shielding;

(c) The operational limits and conditions for normal operation;

(d) Any revised arrangements;

(e) The arrangements for emergency preparedness and response (e.g. the emergency plan, the training and exercise programme);

(f) The arrangements for maintenance, periodic testing, inspection, control of modifications and changes to specifications and surveillance.

4.11. Before starting the implementation of proposals for experiments and modifications that are of major safety significance or that might have a significant effect on safety, the operating organization should submit the appropriate documentation to the regulatory body for review and assessment. Detailed recommendations on utilization and modification projects including commissioning of experiments and modifications are provided in SSG-24 [2].

4.12. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for decommissioning should include the following:

(a) The records and results of operating experience;

(b) The decommissioning plan.

Before the operating organization can be allowed to relinquish an authorization (i.e. before release from regulatory control), the results of decommissioning should be submitted to the regulatory body. Recommendations on decommissioning are provided in SSG-47 [16] and recommendations on release from regulatory control are provided in WS-G-5.1 [32].

Appendix

CONTENT OF A SAFETY ANALYSIS REPORT

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

The chapters to be included in the safety analysis report for subcritical assemblies should be the same as for research reactors. However, in accordance with the application of a graded approach, the amount of information and the level of detail should be consistent with the lesser complexity and lower hazards associated with subcritical assemblies. In addition, some of the technical content in this Appendix might not be applicable to some types of subcritical assembly. These matters are highlighted in the Appendix by an asterisk (*), or are specifically indicated.

CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF THE RESEARCH REACTOR

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

General description of the research reactor

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

Historical review

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

Comparison with other facilities

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

Identification of the owner, the operating organization and representatives

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

Safety features

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

Experimental programme

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

Material incorporated by reference

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

Requirements for further technical information

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

CHAPTER 2: SAFETY OBJECTIVES AND ENGINEERING DESIGN REQUIREMENTS

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

Safety objectives and general design requirements

A.2.2. This section should describe the safety objectives and the general design requirements followed in the design of the research reactor, in consideration of the requirements for normal operation,

anticipated operational occurrences, design basis accidents and design extension conditions. Safety objectives and design requirements for prevention of accidents and mitigation of the consequences of accidents should also be included. Other measures that can be used to mitigate accident conditions should be described in the appropriate chapters of the safety analysis report.

A.2.3. A statement of the overall safety objectives should be included. This should be followed by a brief description of the underlying safety objectives and general design requirements that are important to the ~~design~~safety. Safety objectives are set out in section 2 of SSR-3 [1], and general design requirements are established in section 6 (see Requirements 16–41) of SSR-3 [1]. These objectives and requirements may address the following:

(a) The management system;

(b) A high standard of engineering design and, in particular, conservative design margins, engineered safety features, barriers to prevent radionuclide transfer and protection of these barriers;

(c) Inherent safety features (those relying only on physical properties);

(d) Passive safety features (i.e. which do not depend on an external input);

(e) The extent to which unique or unusual features that might affect the consequences or the probability of releases are incorporated in the design;

(f) The extent to which redundancy, diversity, physical separation and functional independence are applied in the design of safety systems and engineered safety features, so as to achieve the necessary reliability of these systems and features and to protect against common cause failures;

(g) Fail-safe features;

(h) Defence in depth applied in the design, including the independent effectiveness of the different levels of defence;

(i) Accident prevention;

(j) Accident management;

(k) Proven engineering practices and use of generally accepted standards;

(l) Assessment of organizational and human factors and dependent failures;

(m) Radiation protection;

(n) Provisions for utilization and modification;

(o) Provisions for ageing management;

(p) Safety features for design extension conditions;

(q) Provisions for emergency preparedness and response;

(r) Provisions for fire protection

~~(q)~~(s) Provisions for decommissioning in the design of research reactors and their experimental facilities

~~(r)~~(t) Provisions for interfaces between nuclear safety and nuclear security.

Emphasis should be placed on the principles used in the design and not on a description of the research reactor. The summary description of the research 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 established in section 6 of SSR-3 [1] (see Requirements 42–66) and address the following:

(1) The management system 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) The integrity of the reactor core.

(4) Protection against flow instabilities and suppression of power oscillations*.

(5) Criteria for sharing of common structures, systems and components important to safety between facilities at the same site (e.g. emergency power supply, on-site fire brigade).

(6) Consideration of human factors and ergonomic principles to reduce the potential for human error and to relieve stress for the operating personnel.

(7) The design analysis with validated techniques, models or codes.

(8) Provisions for reactivity control, including the following:

(a) Redundant reactivity control*;

(b) Reactivity limits;

(c) Availability of sufficient negative reactivity to maintain the reactor subcritical under all operational states and accident conditions.

(9) Design of the reactor coolant system and related systems, including the following:

(a) Adequate core cooling for all operational states and accident conditions*;

(b) Integrity of the reactor coolant system and protection of the boundary from leakage*;

(c) Preventing the uncovering of the core*.

(10) Design of the reactor core and fuel, including the following:

(a) Fuel design bases for neutronic, thermohydraulic, mechanical, material and chemical design;

(b) Safety margins for fuel design parameters*;

(c) Verification of fuel integrity;

(d) Prevention of inadvertent fuel movement;

(e) Design bases for mechanical, thermal and chemical design of reactor materials important to safety;

(f) Shutdown margins*;

(g) Prevention of criticality for subcritical assemblies.

(11) Provisions for safe utilization and modification, including the following:

(a) Radiation protection for all operating conditions;

(b) Design to ensure that safety system settings are not adversely affected;

(c) Provisions to preserve the means of confinement and shielding of the reactor;

(d) Recognition of the interdependence between the reactor and any installed experimental equipment.

(12) Reactor safety systems, including the following:

(a) Provision of systems for shutdown, fuel cooling and control of radionuclide releases;

(b) Operation of reactors safety systems;

(c) Provision of separation between safety system and control functions;

(d) Application of the single failure criterion;

(e) Fail safe characteristics.

(13) Reliability and testability of instrumentation and control systems, including the following:

(a) Provision of means to achieve required levels of reliability;

(b) Periodic testability;

(c) Fail safe characteristics;

(d) Functional diversity.

(14) Capability for surveillance and maintenance of equipment important to safety.

(15) Radiation protection systems, including the following:

(a) Control of radioactive releases;

(b) Stationary dose rate meters for monitoring at places routinely accessible and at suitable locations for anticipated operational occurrences and accident conditions;

(c) Monitors and laboratories for determining the concentration of selected radionuclides;

(d) Monitoring and control of effluents;

(e) Facilities and Equipment for measuring radioactive surface contamination, and doses to and contamination of personnel;

(f) Monitoring at gates and other entrances;

(g) Arrangements to assess the impact on the region surrounding the research reactor.

(16) Buildings and structures, including the following:

(a) Buildings and structures designed for design basis accidents and, as far as practicable, for design extension conditions*;

(b) Provisions for leaktightness of the reactor building and the ventilation system.

(17) Means of confinement, including the following:

(a) Provision of confinement of radioactive substances in operational states and in accident conditions;

(b) Protection of the reactor against natural external events and human induced events;

(c) Radiation shielding in operational states and in accident conditions.

(18) Emergency cooling of the reactor core, including*:

(a) Preventing damage to the fuel in the event of a loss of coolant accident;

(b) Provisions to perform periodic inspection of components for functional testing and verification of performance of emergency core cooling system.

(19) Use of computer based equipment in systems important to safety, including the following:

(a) Implementation of appropriate standards and best practices for the development and testing of computer hardware and software;

(b) Independent assessment for high reliability of computer based equipment;

(c) Consideration of common cause failures deriving from software*;

(d) Protection against accidental or deliberate interference and cyber-attacks, including design basis threats;

(e) Verification, validation and testing of the software systems;

(f) Independence and performance of data communication;

(g) Suitability of predeveloped software for use in systems important to safety.

(20) Emergency response facilities on the site, including*:

(a) Communication of necessary information about reactor parameters, monitoring systems and information to be used for continuous assessment to the relevant emergency response facilities on the site;

(b) Provisions for means of communication.

(21) Electrical power supply systems, including the following:

(a) Reliable normal electrical power supplies for essential safety functions;

(b) Provision of uninterrupted power supplies;

(c) Provision of an emergency power supply.

(22) Handling and storage systems for fuel and core components, including the following:

(a) Provisions for safely storing a sufficient number of spent fuel elements* and irradiated core components;

(b) Provisions to unload all fuel from the core safely at any time;

(c) Provisions for prevention of criticality by an adequate margin, and for performing inspections and testing;

(d) Provisions to prevent the inadvertent dropping of heavy objects on the fuel and appropriate storage of suspect or damaged fuel elements;

(e) Provisions to permit adequate heat removal and shielding for irradiated fuel for all operational states and accident conditions*.

(23) Systems for management of radioactive waste, including the following:

(a) Provisions to ensure~~nhance~~ safety in waste management and to minimize the generation of radioactive waste;

(b) Provisions for treating solid, liquid and gaseous radioactive waste to keep the amounts and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges.

Classification of structures, systems and components

A.2.5. The approach to the classification of structures, systems and components for purposes of analysis or design, such as for seismic safety or nuclear safety, the basis for the safety classifications and the list of classes should be presented in this section of the safety analysis report. Additional recommendations are provided in IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [35].

Protection against external events

A.2.6. In this section, the design criteria for the resistance of structures, systems and components to the external events listed in para. 3.23~~1~~(7) should be presented. Extreme weather conditions including effects due to climate change should be taken into account for the determination of the external events

as well as combinations of external events. Requirements on site evaluation for research reactors are established in SSR-1 [27].

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 the use of such codes and standards should be provided, particularly if the codes will be used in the design of structures, systems and components important to 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 the following:

— Mechanical design, including stress analysis and fracture mechanics;

— Structural design;

— Earthquake resistant design;

— Selection of materials;

— Fabrication of equipment and components;

— Inspection of fabricated and installed structures, systems and components;

— Thermohydraulic and neutronic design;

— Electrical design;

— Design of instrumentation and control systems;

— Shielding and radiation protection;

— Fire protection;

— Maintenance, periodic testing and inspection as related to design;

— Design, qualification and production of fuel.

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

Technical design methods

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

Design for internal fire protection

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

Qualification of equipment and components

A.2.12. This section should describe the design bases for qualification of equipment and components to resist such environmental factors as vibration, thermal expansion, radiation, corrosion, dynamic effects, mechanical loadings and high pressure, high temperature, humidity, water, steam, chemicals, low temperature or a vacuum. Qualification tests and analyses that have been (or will be) performed should be described.

A.2.13. This section should describe the scope of the qualification programme and the qualification procedures adopted to confirm that the items important to safety, including safety features for design extension conditions, are capable of meeting the design requirements and of remaining fit for purpose in the range of individual or combined environmental challenges identified for the situations under which they are supposed to perform. The identified challenges should take into account all the stages and their duration in the lifetime of the research reactor.

Compliance with national and international standards

A.2.14. This section should provide a statement of the conformance of the research reactor design with the design principles and criteria established in national and international standards, which themselves will allow compliance with the safety objectives adopted for the reactor.

Conclusions

A.2.15. This section should provide the conclusion that the research reactor is designed to meet the overall safety objective and underlying safety objectives, and that appropriate external events, codes, standards and design methods have been considered in the design of the research reactor, including for the qualification of components.

CHAPTER 3: SITE CHARACTERISTICS

A.3.1. This chapter of the safety analysis report should provide information on the geological, seismological, hydrological and meteorological characteristics of the site and the region surrounding the site, in conjunction with present and projected population distributions, land use, site activities and planning controls. The purpose is to indicate how these site characteristics have influenced the design of the research reactor and the operating criteria, and to show the adequacy of the site characteristics

from the safety point of view. Requirements on site evaluation for research reactors are established in SSR-1 [27].

A.3.2. Information should be provided in sufficient detail to permit an independent evaluation and to support the analysis and conclusions of Chapter 16 of the safety analysis report, to demonstrate that the research reactor can be safely operated at the proposed site. For some research reactors with low hazard potential, critical assemblies and subcritical assemblies, the amount of detail provided in this chapter can be substantially reduced. In addition, most of the details described below relating to geology and seismology, meteorology, hydrology and oceanography, radiological impact, adequacy of the site for emergency response actions might not be required for some subcritical assemblies.

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

General site description

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

(a) The location of the research reactor, the site area and the boundaries of the site area;

(b) Location and orientation of principal buildings and equipment;

(c) Location of any nearby industrial, commercial or military facilities, and any institutional, recreational or residential structures;

(d) Nearby highways, roadways, airports, waterways, pipelines and railway lines;

~~(e) Boundaries of the site area, i.e. the area controlled by the operating organization;~~

~~(f)~~(e) Boundaries for establishing release limits for effluents.

A.3.5. This section should describe the legal rights of the operating organization with respect to all areas that lie within the designated site area[10], as well as any activities unrelated to the operation of the research reactor that will be permitted in the site area.

Evaluation of site specific hazards

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

A.3.7. This section should describe the appropriate methods adopted for establishing the external effects that will constitute the postulated initiating events for important natural phenomena and human induced effects. Attention should be paid to external hazards that could potentially lead to common

[10] The site area is the geographical area that contains an authorized research reactor, authorized activity or source, and within which the management of the authorized research reactor or authorized activity may directly initiate emergency response actions [12].

cause failures of the safety systems and additional safety features for design extension conditions. Further information on design criteria for protection against these effects should be given in Chapter 2 of the safety analysis report (see para. A.2.6).

Geology and seismology

A.3.8. The geological, tectonic, seismological and volcanic characteristics of the site and the region surrounding the site should be described in this section in sufficient detail to identify effects that could present a hazard to the research reactor. The evaluation of seismic hazards should be based on a suitable geotectonic model substantiated by appropriate evidence and data. The results of this analysis, to be used further in other sections of the safety analysis report in which structural design, seismic qualification of components and safety analysis are considered, should be described in detail. A historical overview of reported earthquakes that could reasonably be expected to have affected the region surrounding the site should be presented.

A.3.9. Information that is used to establish the seismic design, such as earthquake recurrence intervals and ground motion (including the static and dynamic stability of all soil or rock slopes, both natural slopes and artificial slopes) 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, including the potential for sink holes;

— Assessing the potential for volcanic activity;

— Assessing the potential for liquefaction and ground motion.

Meteorology

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

Hydrology and oceanography

A.3.11. The surface and underground hydrology of the site and the region surrounding the site 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 artificial structures should be indicated, including dams, diversion channels and any flood control measures. Foreseeable changes in land use that might influence hydrology should be described, for example, changes in runoff characteristics resulting from urbanization, or realignment of drainage channels.

A.3.12. A description of the groundwater hydrology in the region surrounding 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 research reactor is to be built by the coast, oceanographic and hydrographic information, including a bathymetric map of the near-shore area in front of the location of the reactor, should be provided.

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

— Flooding;

— Surges, seiches and wave action, including effects of ice ridges;

— Seismically induced phenomena such as tsunamis and dam failures.

Nearby industrial, transport and other facilities

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

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

Radiological impact

A.3.17. This section should describe radiological aspects and, in particular, the biological aspects of transfers of radioactive material to people. Most of these details might not be necessary for some low hazard, low power reactors, critical assemblies and subcritical assemblies. In this case, only a brief summary should be given under each heading. If no radiological impact section is provided, justification should be provided for omitting this section of the safety analysis report. This section should also cover all aspects of site activity that have the potential to affect the radiological impacts of the reactor throughout its lifetime, including construction, operation under normal conditions and decommissioning

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

Population distribution

A.3.19. The population distribution around the research reactor and in the region, including seasonal and daily variations, and the land use that is relevant to the safe design and operation of the research reactor should be presented in this section. In particular, information on existing or projected population distributions around the research reactor should be collected and kept up to date during the lifetime of the research reactor.

Natural environment and land and water usage

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

(a) Land and bodies of water supporting wildlife;

(b) Land devoted to agricultural use;

(c) Land devoted to livestock or dairy farming;

(d) Land devoted to commercial, residential or recreational purposes;

(e) Bodies of water used for commercial or sport fishing;

(f) Bodies of water used for commercial purposes or recreation;

(g) Direct and indirect pathways for radioactive contamination of food chains.

Baseline radioactivity levels

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

Atmospheric dispersion of radioactive material

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

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

concentrations and dose calculations, which should be presented in Chapters 12 and 16 of the safety analysis report.

Dispersion of radioactive materials through surface waters and groundwater

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

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

Adequacy of the site for emergency response actions

A.3.26. This section should consider, but not be limited to:

— Population distributions and projected population changes in the region surrounding the research reactor;

— Present and projected land use and water use in the region;

— Potential radioactive source terms, and doses to the population from different exposure pathways (e.g. airborne radioactive material and aqueous pathways);

— Potential contamination of food chains;

— Potential exposures of on-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 response actions if required;

— The feasibility of emergency plans (if they are required), with account taken of the population distribution, national and international borders, special groups (e.g. in hospitals), special geographical features (e.g. islands), the availability of evacuation routes and reception centres for evacuees, and communication and transport provisions.

Monitoring of site related parameters

A.3.27. This section should define site related parameters that could be affected by the external events that have been taken into account for the analyses (e.g. parameters that could be affected by seismic, atmospheric, water and groundwater related events, and demographic, industrial and transport related

factors). The strategy for monitoring, the provisions for monitoring and the use of the results in preventing, mitigating and predicting the effects of site related hazards should be described.

Conclusion

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

CHAPTER 4: BUILDINGS AND STRUCTURES

Reactor building

A.4.1. This section should contain a description of the reactor building and internal structures (e.g. reactor pools and internals, supporting structures, cranes, ventilation systems), emphasizing those characteristics of the building that assist in maintaining acceptable radiation levels on and off the site for all operational states and accident conditions, as appropriate. Requirements for the reactor building are established in SSR-3 [1] (Requirements 42 and 43).

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 (e.g. air locks, doors windows, mechanical and electrical penetrations) in relation to their resistance to internal and external events (see paras A.2.11 and A.3.7).

A.4.3. The design and operation of the ventilation systems should be described, including requirements for containment or means of confinement, and including the ventilation exchange rates for the different modes of operation. If applicable, distinction should be made between the ventilation system used for normal operation and the ventilation system used for emergencies. The specific efficiencies of the air filters and iodine traps should be given.

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

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

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

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

Auxiliary structures

A.4.8. This section should include a description of reactor auxiliary buildings and structures important to safety. If applicable, it should include a description of emergency response facilities and the supplementary control room*.

CHAPTER 5: THE REACTOR

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

— Control of reactivity;

— Removal of heat from the reactor and from the fuel storage;

— Confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

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

Summary description

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

Fuel elements

A.5.4. Basic information to be provided on the fuel design and fuel properties should comprise:

(a) Fuel material, enrichment, composition and metallurgical state (e.g. oxide, alloy);

(b) Material (i.e. type, composition) of all other fuel parts, such as cladding, spacers and fittings, and burnable neutron absorbers;

(c) Fuel geometry, dimensions and tolerances (together with drawings);

(d) The material properties required for the analyses mentioned in paras A.5.5–A.5.8;

(e) The maximum temperatures to which the fuel elements can be subjected without deformation (due to blister formation or mechanical weakening);

(f) Fuel qualification;

(g) Operating experience relating to the fuel, if any;

(h) Fuel element instrumentation, if any.

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

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

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

A.5.8. An analysis should be provided that shows that the intended irradiation conditions and limits (e.g. fission, density, total fissions at the end of core lifetime) are acceptable and will not lead to undue deformation or swelling of components that might contain fissile material. The anticipated upper limit of the eventual deformation (e.g. expressed as minimum cooling channel width) should be provided for the thermal safety analysis.

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

Reactivity control system

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

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

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

Nuclear design

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

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

(a) Core configuration and composition, such as the type and anticipated loading pattern of fuel elements, control elements and other components that affect the nuclear properties of the core. Since the core configuration for the research reactor might change with the changing experimental applications and requirements, the analysis may use a standard core configuration that has conservative properties with respect to all other configurations. An explanation of the intended fuel replacement strategy should complement this information. The information should be supported by drawings.

(b) Horizontal and vertical distributions of the neutron flux in the core at thermal neutron and fast neutron energy levels.

(c) Basic reactivity characteristics of the core such as the infinite and the effective neutron multiplication factors; the anticipated effectiveness and the position of control elements during the core lifetime; the minimum shutdown capacity; reactivity feedback properties with regard to temperature and void; and reactivity worth of individual core components (e.g. fuel elements, irradiation devices).

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

A.5.16. An analysis should be provided that shows that the effectiveness, speed of action and shutdown margin of the reactor shutdown system[11] are acceptable, and that a single failure in the shutdown system will not prevent the system from fulfilling its safety functions when required. A sufficient shutdown margin should be provided so that the reactor can be brought to and maintained in a subcritical state in all operational states and accident conditions.

Thermohydraulic design

A.5.17. Information should be provided to prove that, in all operational states, adequate capacity for core cooling will be available to keep the reactor fuel thermal parameters within acceptable levels, and that adequate safety margin will be maintained to prevent or to minimize fuel damage under accident conditions. For subcritical assemblies, the details described below should be addressed, as applicable, commensurate with the design configuration of the specific facility.

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

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

(a) All safety related hydraulic characteristics of individual core components and of the core as a whole (such as average and local coolant velocities, and coolant pressures, as appropriate) for operational states during forced and natural convection cooling;

(b) The power distribution, including power peaking factors, in all core components that might contain fissile materials, as derived from the nuclear design characteristics provided in para. A.5.14(b).

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

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

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

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

Reactor materials

A.5.23. Information should be provided that shows that all materials that have been selected for the construction of safety relevant structures and components can withstand the nuclear, thermal and chemical environments to which they will be subjected, without unacceptable worsening of the performance of the safety functions of such structures and components. Ageing effects due to the deterioration of properties as well as irradiation damage should be included. Materials with low activation properties should be considered in the process of selection of materials.

A.5.24. Items that should be considered include the following:

(a) Core support and hold down structures;

(b) Safety relevant reactor internals such as guides of the reactivity control mechanism;

(c) The reactor pool or tank and related components constituting the primary coolant boundary;

(d) Support structures for the reactor tank, safety instrumentation, irradiation facilities, beam tubes.

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

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

CHAPTER 6: RESEARCH REACTOR COOLING SYSTEMS AND CONNECTED SYSTEMS

A.6.1. This chapter of the safety analysis report should provide a description of the reactor cooling systems that remove the heat from the reactor. The description should contain the main design characteristics and performance characteristics in operational states and accident conditions. It should be supported by schematic flow diagrams and an elevation drawing of the cooling systems. For research reactors with low hazard potential, critical assemblies and subcritical assemblies, this chapter of the safety analysis report should be commensurate with the safety significance of the cooling systems and connected systems. A brief statement should be provided to justify the level of detail in this chapter.

Primary cooling system

A.6.2. The design and operation of the primary cooling system should be described in detail. The design and performance characteristics of the main components (pumps, valves, heat exchangers, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials ~~of~~ that the components are made of and the effects of irradiation on these materials should be specified. The reactor vessel, together with in-service environmental factors, such as corrosion, fatigue, thermal stress cycling and ageing effects, should be described.

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

A.6.4. The chemistry data for the primary coolant should be presented, including the effects of irradiation of the primary coolant. The system for monitoring of radionuclides in the primary coolant should also be described.

Secondary cooling system

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

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

Moderator system

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

Emergency core cooling system

A.6.8. The design and operation of the emergency core cooling system should be described in detail. The accident conditions for which this system is designed should be mentioned, and analyses should be provided to demonstrate that the system fulfils the requirements. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials that the components are made of should be specified, the effects of irradiation, if any, should be addressed, and any environmental effects and ageing effects should also be addressed. The procedures for inspection and testing of the emergency core cooling system should be ~~mentioned~~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 accident conditions for which this system is designed should be presented and analyses should be provided to demonstrate that the system fulfils the requirements. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of should be specified; the effects of irradiation, if any, and any corrosion and ageing effects should be addressed, as well as unfavourable environmental conditions for the ultimate heat sink.

Primary purification system

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

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

Primary coolant make-up system

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

CHAPTER 7: ENGINEERED SAFETY FEATURES

A.7.1. This chapter of the safety analysis report should identify and provide a summary of the types, locations and functions of the engineered safety features provided in the research reactor for anticipated operational occurrences and accident conditions. Examples of engineered safety features are the emergency core cooling system and the containment or other means of confinement. The requirements for these systems and supplementary features are stated in Requirement 43 and Requirement 48 of SSR-3 [1]. Examples of safety features for design extension conditions are an additional cooling water supply and non-permanent equipment, e.g. portable diesel generators. For research reactors with low hazard potential, critical assemblies and subcritical assemblies, this chapter should be commensurate with the safety significance of the engineered safety features. A brief statement indicating these features should be provided to justify the level of detail in this chapter.

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

A.7.3. Information should be provided on the following:

(a) Component reliability, system interdependence, redundancy, diversity, ~~of~~ fail-safe characteristics and physical separation of redundant systems;

(b) Evidence that the material used will withstand the postulated accident conditions (e.g. radiation levels, radiolytic decomposition, temperature, pressure);

(c) Provisions for tests, inspections and surveillance (including those performed under simulated accident conditions) to ensure that the feature will be dependable and effective upon demand;

(d) Effects of ageing on the operability of the engineered safety feature.

A.7.4. The design specifications of safety features for design extension conditions, where provided, should be described, along with a description of the capability of these features for preventing or mitigating the radiological consequences, including their reliability regarding the functions that they are

required to fulfill, independence from those features used in design basis accidents, and capability of performing in the environmental conditions pertaining to design extension conditions.

A.7.5. Reference should be made to the relevant chapters of the safety analysis report, where provided, or to other documents where the engineered safety features and additional safety features for design extension conditions are further described.

CHAPTER 8: INSTRUMENTATION AND CONTROL SYSTEMS

A.8.1. This chapter of the safety analysis report should provide information regarding the instrumentation and control systems of all safety systems and safety related items and systems. The information provided should emphasize those instruments and associated equipment that affect reactor safety. The requirements for instrumentation and control systems are established in Requirement 49 of SSR-3 [1].

A.8.2. All instrumentation and control systems and supporting systems (with emphasis on safety systems and safety related systems), including alarm, communication and display instrumentation, should be listed, and considerations of instrumentation errors should be included. Information on the human factors considered in the design of instrumentation and control systems important to safety should also be included. Recommendations on this topic are provided in SSG-37 [10]. Adequate schematic diagrams should also be provided.

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

Reactor protection system

A.8.4. Requirement 50 of SSR-3 [1] establishes requirements for the reactor protection system. The reactor protection system, including all its components, should be described in detail in this section of the safety analysis report. 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 diversity) and to bring the research 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 failures and common mode failures, as well as with single failures. It should also be shown that protection system instrumentation is fail safe in nature.

A.8.6. For computer based digital protection systems, evidence of software verification and validation should be included. Additional recommendations on verification and validation of software, which can also be useful for research reactors, are provided in IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [36].

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

A.8.8. This section should describe the methods used to prevent adverse environmental conditions (e.g. conditions of temperature, humidity, high voltage, electromagnetic fields) from influencing the reactor protection system, as well as methods to protect against tampering.

Reactor power control system

A.8.9. All elements of the reactor power control system should be described (including the design criteria and functionality). Any interfaces between the power regulating system and the reactor protection system should be identified and analysed to confirm that they do not lead to a degradation of safety. For subcritical assemblies, this chapter should be commensurate with the safety significance of the reactor power control system. A brief statement should be provided to justify the level of detail in this chapter.

Other instrumentation and control systems

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

— The fire protection system;

— The experiment control system;

— The ventilation control system;

— The secondary cooling system*;

— The coolant chemistry control system;

— The radiation monitoring system;

— The seismic monitoring system;

— The monitoring system for external meteorological and hydrological conditions.

Alarm system

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

Interlocks

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

Control room

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

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

A.8.15. The information required in emergencies, including information available in the supplementary control room and on-site emergency response facilities, where provided, should be addressed.

CHAPTER 9: ELECTRIC POWER

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

Off-site power supply

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

Emergency power supply

A.9.3. This section should describe the design and operation of the emergency power supply, including provisions for non-permanent equipment necessary to restore the electrical power supply in design extension conditions, as needed, and should emphasize the connection to the off-site power supply.

A.9.4. The description should include the following:

(a) The dependability of the system;

(b) The starting load demands of the equipment powered by the system;

(c) The starting time of the system and the time sequence for connecting loads;

(d) The starting method (automatic or manual);

(e) The duration of operation with and without diesel backup.

Uninterruptible power supplies

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

Cables and routeing

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

Grounding and lightening protection

~~A.9.6.~~A.9.7. This section should provide description of the grounding and lightning protection (both internal and external protection) system, including the components associated with the various grounding subsystems.

CHAPTER 10: AUXILIARY SYSTEMS

A.10.1. This chapter of the safety analysis report should provide information concerning the auxiliary systems included in the research reactor. A description of each system, the design bases for the system and for essential components, a safety assessment demonstrating how the system meets the requirements of the design basis, information on the testing and inspection to be performed to verify the capability and dependability of the system, and information on the instrumentation and control system required should be provided. The storage system for non-permanent equipment used in design extension conditions, where applicable, should be described. 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 research reactor and its safety features or that might contribute to the control of radioactive material inside the research reactor. For those systems, foreseeable ageing effects that could affect safety should also be addressed.

Fuel storage and handling

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

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

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

Water systems

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

Auxiliary process systems

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

Heating, ventilation and air conditioning systems

A.10.7. The systems for heating, air conditioning and ventilation provided for all areas of the reactor building should be addressed in this section. This information should include the design basis, a system description, testing and inspection requirements and foreseeable ageing effects. Consideration should be given to the result of the safety analysis of design extension conditions, maintaining the habitability and good condition of control room in accordance with Requirement 75 of SSR-3[1] (see paras A.16.47-A.16.52). Additional functions of ventilation systems, for example, ventilation systems used in the confinement function, may be addressed in other relevant chapters of the safety analysis report.

Fire protection system

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

Lifting equipment

A.10.9. A description of the lifting equipment should be provided in this section. The related rules and assumption for design should also be described and justified. Special attention should be given to critical load handling operations that could potentially have an effect on the fulfilment of safety functions. The information provided should demonstrate that Requirement 63 of SSR-3 [1] is fulfilled and should include the parameters defining the load that, if dropped, would cause the greatest damage;; the area of the research reactor where the load would be handled; the design of the lifting equipment; and the applicable procedures for operation, maintenance and inspection.

Other auxiliary systems

A.10.10. In this section, the design bases, system descriptions and safety analysis should be provided for the other auxiliary systems, such as general communication systems, lighting and emergency lighting systems, sanitary provisions, sewerage systems and gas service systems.

CHAPTER 11: UTILIZATION OF THE RESEARCH REACTOR

A.11.1. This chapter of the safety analysis report should describe the expected experimental use of the research reactor and should provide information demonstrating that provisions have been made to ensure

that the experimental facilities and experiments are within the safety criteria established for the research reactor, the site personnel, experimenters and the public. Requirements are established in SSR-3 [1], and recommendations are provided in SSG-24 [2].

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, pneumatic rabbit systems and experimental loops. The postulated initiating events such as failure of experimental apparatus or material (e.g. loop rupture, exothermic chemical reactions, see para. 3.23~~1~~), should be evaluated. The analysis results and the safety design features of experimental facilities with respect to these events should be provided. Ageing effects that could affect safety should also be addressed.

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

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

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

A.11.6. The maximum allowable positive as well as negative reactivity of materials used in the experiments inserted in or near the reactor should be specified. This should include the maximum speed of insertion and withdrawal of experiments.

CHAPTER 12: OPERATIONAL RADIATION SAFETY

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

(a) The radiation protection programme (see requirement 84 of SSR-3[1], including the radiation protection policies and objectives of the operating organization;

(b) Sources of radiation at the research reactor;

(c) Design of the research reactor for radiation protection;

(d) The waste management programme and waste management systems;

(e) Dose assessment for normal operation;

(f) Conclusions.

A.12.2. The estimated radiation exposure of the site personnel and the public for anticipated operational occurrences and accident conditions should be analysed in Chapter 16 of the safety analysis report. Planning for a nuclear or radiological emergency is described in Chapter 20, and management of irradiated fuel should be treated in Chapter 10 of the safety analysis report.

Radiation protection programme

Radiation protection policy and objectives of the operating organization

A.12.3. This policy statement should endorse the radiation protection objective as stated in paras 2.2 and 2.3 of SSR-3 [1]. In particular, this section should summarize the authorized dose limits for both personnel and the public, as well as the discharge limits based on these dose limits. The regulatory requirements for maintaining exposures and discharges of radioactive material, including radioactive waste and effluents, below the authorized limits should be described. The dose constraints established by the operating organization to assist the research reactor management in applying the optimization principle to ensure that radiation doses and discharges are as low as reasonably achievable and are below the authorized limits should also be described. Recommendations on the application of the optimization principle are provided in NS-G-4.6 [8]. The records that should be kept to prove that use of the research reactor leading to possible exposure to radiation is justified should also be specified.

A.12.4. The radiation protection programme established and implemented by the operating organization of the research reactor, including the application of the optimization principle, should be described. The policy and arrangements for control of radioactive releases at the research reactor, including the organizational policy concerning monitoring of releases and the evaluation of trends, should also be described.

Organization, staffing and responsibilities

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

Facilities, equipment and instrumentation

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

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

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

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

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

Procedures and training

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

— The policy, methods and frequencies for conducting radiation surveys and air sampling;

— Effluent monitoring;

— Administrative measures for controlling access to or occupancy times in controlled areas;

— Control of contamination of personnel and equipment;

— Control of compliance with applicable regulations for the transport of radioactive material;

— The methods and procedures for personnel monitoring, including methods for recording, reporting and analysing results;

— The programme for assessment of internal radiation exposure, such as bioassay or whole body counting, and other related medical surveillance of personnel, in particular for cases of overexposure;

— The issue, selection, use and maintenance of protective equipment such as respirators;

— The handling and storage of sources, radioisotopes or other radioactive material;

— The handling and disposal of radioactive waste;

— The training of experimenters and site personnel.

A.12.12. Reference should be made to the operating procedures, which include provisions for controlling the doses to operating personnel in normal operation and during work for maintenance, in-

service inspection and refuelling. Reference should also be made to the parts of the operating procedures that address provisions for the monitoring of systems that collect, contain, store or transport radioactive liquids, gases or solids. Any procedures relating to experimental facilities, isotope production or laboratory activities should be referenced.

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

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

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

Effluent monitoring programme

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

Audit and review programmes

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

Radiation sources at the research reactor

A.12.18. All radiation sources (contained sources liquid and airborne radioactive material) relating to reactor operation and all other radiation sources located throughout the research reactor that can be identified should be catalogued in this section. Such sources might include those used as bases for shielding calculations, the design of ventilation systems, dose assessment, waste management and the determination of effluent releases.

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

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

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

Design of the research reactor for radiation protection

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

Access control and zoning

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

Shielding and protective features

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

Ventilation for purposes of radiation protection

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

Radiation monitoring systems

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

— Locations of monitors, detectors and samplers;

— Types of monitor and instrumentation (stationary or mobile, sensitivity, type of measurement, range, accuracy and precision);

— Types and locations of local and remote alarms, annunciators, readouts and recorders;

— Alarm or controller set points;

— Provision of emergency power supplies;

— Requirements for calibration, maintenance and testing;

— Automatic actions to be initiated or taken.

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

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

Radioactive waste management programme and waste management systems

Solid radioactive waste

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

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

(b) For wet radioactive waste, the methods of dehydration;

(c) The methods of collection, segregation, processing, packaging, storage and transport of radioactive waste;

(d) The type and size of waste container.

Liquid waste

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

(a) The types and quantities of liquid radioactive waste, and the origins, locations, forms and estimated activities of liquid radioactive waste;

(b) Diagrams of flow paths and flow rates, process equipment, storage tanks and release points for releases to the environment;

(c) Measures to separate radioactive effluents and non-radioactive effluents and to ensure that the effluents released to the environment are soluble;

(d) Administrative control levels for releases;

(e) Requirements for the system capacity, redundancy and flexibility, and for the capability of the system to facilitate maintenance, reduce leakage and prevent uncontrolled releases such as overflow from tanks to the environment.

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

Gaseous waste

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

(a) The types and quantities of gaseous waste, and the sources, locations, forms and calculated quantities of radionuclides;

(b) Diagrams of flow paths and flow rates, process equipment and release points for releases to the environment;

(c) Measures to separate radioactive effluents and non-radioactive effluents;

(d) Administrative control levels for releases;

(e) Requirements for the system capacity, redundancy and flexibility, and for the capability of the system to facilitate maintenance, reduce leakage and prevent uncontrolled releases to the environment.

A.12.33. The expected effluent concentrations should be tabulated by radionuclide released, including total annual radioactive release to the environment. The dilution factors upon release should be given.

A.12.34. If applicable, design provisions to handle hazardous gaseous material with potential for explosion should be described.

A.12.35. Detailed requirements on the subject are established in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [37].

Dose assessment for normal operation

Doses to the public

A.12.36. This section should demonstrate that the combined effects of direct radiation and of releases of radioactive material from the research reactor do not result in off-site doses to the public that exceed authorized limits. In addition, measures to reduce the exposures on the basis of the optimization principle should be described.

A.12.37. If previous sections of this chapter of the safety analysis report have demonstrated that radioactive releases are a small fraction of the discharge limits and are acceptable, and that both direct and indirect exposure to radiation are also within authorized limits, this section should provide only a

summary of all pathways of radiation exposure: airborne radioactive material, liquid radioactive material, and direct and indirect exposure to radiation.

A.12.38. If radioactive releases have not been treated in terms of discharge limits, then this section should include a calculation of the individual dose to the representative person (see GSR Part 3 [23]), at the research reactor site boundary and at off-site locations, due to the effects of all releases. A description of the assumptions, methods and tools used in the calculation should also be presented. It should be demonstrated that the combined effects of all releases meet regulatory requirements for doses to the public.

A.12.39. This section should state the criteria to be used for determining that gaseous and liquid radioactive releases are generated at an acceptable rate. The effluent concentrations tabulated by radionuclide released and the total annual radioactive releases to the environment should be included, together with the methods, parameters and assumptions used in calculating these quantities.

A.12.40. In addition, for gaseous effluents, all points of release of radioactive material to the environment should be identified, providing the following for each quantity:

(a) The height of the release;

(b) The effluent temperature and the exit velocity;

(c) Assumptions made concerning the transport and dilution of the gases in the environment.

Occupational exposure

A.12.41. This section should present a diagram showing the radiation fields in normally occupied areas of the research reactor and in areas where maintenance activities will be performed. Estimated annual occupancy data for the controlled areas of the research reactor should be used to show that the expected doses are acceptable for the major functions, such as research reactor operation, conduct of experiments, maintenance, radioactive waste management, refuelling and in-service inspection. An estimate of the annual dose at the boundaries of controlled areas should be provided. Further guidance on occupation exposure is provide in NS-G-4.6 [8] and GSG-7 [24].

A.12.42. This section should demonstrate that the estimated radiation exposure of personnel due to inhalation in areas with airborne radioactive material is acceptable. If data are available, a summary of the annual doses to research reactor personnel should be provided.

Conclusion

A.12.43. This section should give a conclusion regarding the acceptability of the operational radiation protection programme and the design features at the research reactor.

CHAPTER 13: CONDUCT OF OPERATIONS

A.13.1. This chapter of the safety analysis report should describe the organizational structure and the way in which the operating organization will conduct the operations of the research reactor. This should include the staffing, review and audit of operations of the research reactor; operating procedures; maintenance; testing and inspection; interfaces with nuclear security; and records and reports. Consideration of organizational and human factors should also be addressed along with the information provided on staffing, training and qualification of personnel, operating procedures, and maintenance, periodic testing and inspection programme. Requirements on these topics are established in SSR-3 [1], and recommendations on these topics are provided in NS-G-4.2 [4], NS-G-4.4 [6] and NS-G-4.5 [7].

Organizational structure

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

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

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

Staff qualification and training

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

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

Review and audit

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

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

Operating instructions and procedures

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

A.13.10. These written instructions and procedures (see also NS-G-4.4 [6]) should include information on the following items, as appropriate:

— Reactor startup, operation and shutdown;

— Loading, unloading and movement of fuel and irradiated material;

— Inspection and testing of items important to safety, in particular the safety systems;

— Setting up, testing and performance of experiments with safety significance;

— Use of radioactive material produced and shipment of radioactive materials

— Maintenance, in particular concerning major components or systems important to safety;

— Radiation protection;

— Response to anticipated abnormal occurrences, failures of systems or components, and accident conditions;

— Effluent monitoring and environmental monitoring;

— Emergencies;

— Nuclear security, including physical protection and information security (see paras A.13.13 and A.13.14);

— Fire protection.

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

Maintenance, periodic testing and inspection

A.13.11. This section should describe the conduct of the maintenance, periodic testing and inspection programme for equipment and components of the research reactor, which should be based on the recommendations provided in NS-G-4.2 [4]. An overview is sufficient if the detailed programme is provided in supplementary documents. This section should provide information on the following aspects of the maintenance, periodic testing and inspection programme:

(a) The system or equipment to be inspected or tested;

(b) The inspection or testing criteria;

(c) The inspection or testing intervals;

(d) The persons responsible for the maintenance, testing or inspection;

(e) Approval of maintenance work;

(f) Resumption of normal operation after maintenance.

Ageing management

A.13.12. This section should describe the ageing management programme including inspection and periodic testing of materials for research reactor structures, systems and components which should be effective and developed systematically based on the recommendations provided in SSG-10 [9]. An overview is sufficient if the detailed programme is given in supplementary documents. This section should provide information on the following aspects of the systematic ageing management programme:

(a) Screening of structures, systems and components for ageing management review;

(b) Identification and understanding of degradation mechanisms;

(c) Minimization of ageing effects;

(d) Detection monitoring and trending of ageing effects;

(e) Mitigation of aging effects;

(f) Acceptance criteria;

(g) Corrective actions.

Nuclear safety and nuclear security interfaces

A.13.13. The measures taken to protect the research reactor against unauthorized access and sabotage, and to protect against unauthorized removal of fissile and radioactive material, should be kept confidential and therefore be described in a separate plan for physical protection (see Refs [17] and [29]), including procedures for access to the site and to the research reactor, and the physical protection systems.

A.13.14. It should be indicated how the operating organization ensures that safety measures and nuclear security measures are implemented in accordance with Requirement 90 of SSR-3 [1]. Safety measures and nuclear security measures are required to be designed and applied in an integrated manner, and as far as possible in a complementary manner, so that nuclear security measures do not compromise safety and safety measures do not compromise security. This section should describe the system to address safety and nuclear security aspects in a coordinated manner and involving all interested parties, together with the identification of specific provisions important for integration of safety and nuclear security.

Documents and Records

A.13.15. This section should provide information on the system for controlling records, data and reports that are important to safety. The records should address the following:

(a) Reactor operation (e.g. logbooks, strip charts, checklists, automatic data readout);

(b) Operational status (e.g. type and number of operational components and of components out of service);

(c) Maintenance, testing and inspection protocols;

(d) Modifications;

(e) Irradiation of samples and radionuclides produced;

(f) Movement of fissile material;

(g) Radiation levels;

(h) Radiation exposure (external and internal), radiation doses to personnel and records of medical examinations;

(i) Results of effluent monitoring and environmental monitoring;

(j) Failures of and other events involving components important to safety;

(k) Training and retraining.

A.13.16. This section should give the minimum time interval for which records are to be stored in accordance with the management system for the operation of the research reactor (see Chapter 18 of the safety analysis report).

Programme for the feedback of operating experience

A.13.17. This section should describe the programme for the evaluation and feedback of operating experience, including the evaluation of trends in operational issues, trends in malfunctions, near misses and other events that have occurred at the research reactor and, as far as applicable, at other nuclear installations. The programme should include consideration of technical, organizational and human factors. Recommendations on this area are provided in the IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [38].

CHAPTER 14: ENVIRONMENTAL ASSESSMENT

A.14.1. This chapter of the safety analysis report should provide a summary of the report of the environmental impact assessment of the authorized facility and activities including construction, operation, modification and decommissioning of the research reactor.

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

(a) The environmental impact of the authorized facility and activities;

(b) Unavoidable adverse environmental effects;

(c) Alternatives to the authorized facility and activities that were considered;

(d) Irreversible and irretrievable commitments of resources;

(e) An analysis providing a balance of the environmental effects of the authorized facility and activities and the alternatives available for preventing or mitigating environmental effects, as well as a

summary of the environmental, economic, societal, technical and other benefits deriving from the research reactor.

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

CHAPTER 15: COMMISSIONING

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

A.15.2. The safety analysis report should provide a description of the major stages of the commissioning programme and specific objectives to be achieved for each stage (see para 2.33).

Research reactors under construction

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

(a) A summary of the commissioning programme and its objectives;

(b) Details of the commissioning organization, including training requirements;

(c) An outline of the management system for commissioning (see Chapter 18 of the safety analysis report);

(d) A summary schedule of the major phases of the commissioning programme;

(e) A summary of the operational limits and conditions for commissioning and of the commissioning procedures.

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

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

Research reactors after commissioning

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

(a) A summary of the results of commissioning;

(b) A summary of the major technical and organizational changes during the commissioning process;

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

(d) An overview of modifications of structures, systems and components, of procedures and of the safety analysis and the safety analysis report.

Existing research reactors

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

(a) A summary of the commissioning programme and its objectives;

(b) A summary of the results of commissioning;

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

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

Commissioning of modifications

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

CHAPTER 16: SAFETY ANALYSIS

A.16.1. The safety analysis presented in this chapter forms the focal point of the safety analysis report. In previous chapters, it is stated that the research reactor design, and especially the design of structures, systems and components important to safety, should be evaluated for the susceptibility of structures, systems and components to malfunctions and failure. In this chapter, the effects of anticipated deviations in operational processes from normal operation and postulated component failures and malfunction due to organizational or human errors (i.e. postulated initiating events) should be described, including their consequences, to evaluate the ability of the research reactor to control or to accommodate such situations and failures. These analyses include deterministic safety analysis of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation and of design extension conditions with core melt, and analyses performed in support of 'practical elimination' of conditions arising that could lead to early radioactive releases or large

radioactive releases, as well as any probabilistic safety assessment performed to complement deterministic safety analyses.

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) Characteristics of the research reactor— the research reactor parameters and initial conditions used in the safety analysis (paras A.16.5–A.16.9);

(3) Selection of postulated initiating events — the range of postulated initiating events considered in the safety analysis (paras A.16.10–A.16.12);

(4) Evaluation of individual event sequences — the results of the safety analysis (paras A.16.13–A.16.46);

(5) Analysis of design extension conditions (paras A16.47 –A.16.52)

~~(5)~~(6) Summary — a summary of significant results and conclusions regarding acceptability (paras A.16.~~47~~53–A.16.~~48~~55).

Introduction

A.16.3. This section should provide an overview of the methods and approaches used in the safety analysis. The information provided should be sufficient for a reviewer to obtain a basic understanding of the methods used and of the general nature of the criteria used to assess the acceptability of the results. Annex I of this Safety Guide may be of some assistance in completing this section, but the level of detail of Annex I is not necessary ~~here~~in this section. Additional guidance is provided in Ref. [25~~4~~].

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

(1) Methods of identification, selection and justification of postulated initiating events.

(2) Categorization of the postulated initiating events in anticipated operational occurrences, design basis accidents and design extension conditions.

(3) 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.

(4) Acceptance criteria.

Characteristics of the research reactor

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

Core parameters

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

(a) Core power;

(b) Core inlet temperature;

(c) Fuel element cladding temperature;

(d) Reactor system pressure*;

(e) Core coolant flow rate*;

(f) Axial and radial power distribution and hot channel factor*;

(g) Power peaking factor*;

(h) Excess reactivity;

(i) Reactor kinetics parameters;

(j) Fuel temperature reactivity coefficient and moderator temperature reactivity coefficient;

(k) Void reactivity coefficient;

(l) Available shutdown reactivity margin*;

(m) Margin to criticality for subcritical assemblies

(n) Insertion characteristics of reactivity control and safety devices.

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

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

Functions of the research reactor protection system

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

Identification, categorization and grouping of postulated initiating events

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

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

(a) Loss of electric power supplies;

(b) Insertion of excess reactivity;

(c) Loss of flow;

(d) Loss of coolant;

(e) Erroneous handling or failure of equipment;

(f) Special internal events, including failure of experiments;

(g) External events;

(h) Human error.

For some subcritical assemblies, the categorization will be dependent on facility specific design features and their importance to safety.

A.16.12. The basis for the categorization and grouping of postulated initiating events should be described and justified. The list of scenarios to be addressed in this chapter of the safety analysis report should cover anticipated operational occurrences, design basis accidents and design extension conditions. The postulated initiating events in each group should be evaluated to identify the bounding events, and the events selected for further analysis should be indicated and justified. The events selected for further analysis should include those having potential consequences that are bounding for all other postulated initiating events in the group.

Evaluation of individual events

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

(a) Identification of causes;

(b) Sequence of events and systems operation;

(c) Transient analysis;

(d) Classification of damage states;

(e) Derivation of source terms;

(f) Evaluation of radiological consequences.

A.16.14. The extent of the quantitative information that should be given for these topics will differ for the various postulated initiating events and will depend on the type of research reactor. For those situations in which a particular postulated initiating event is not a bounding event, only the qualitative reasoning that led to that conclusion should be given, together with a reference to the section presenting an evaluation of the bounding event for the relevant category. Furthermore, for those postulated initiating events that require a quantitative analysis, it might not be necessary to provide such an analysis for each topic. For example, there are a number of events initiating a research 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 does not need to be performed for each such initiating event.

Identification of causes

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

Sequence of events and systems operation

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

(a) Identification of significant occurrences on a timescale, for example, neutron flux monitor trip or start of insertion of control rods;

(b) Indication of the proper functioning of normally operating reactor instrumentation and controls, and of their failure to function;

(c) Indication of proper functioning of the reactor protection system, the safety systems and other engineered safety features, and of their failure to function;

(d) For design extension conditions, additional failures that are assumed in the event sequence;

(e) Indication of the required operator actions;

(f) Evaluation of dependent failures and malfunction due to organizational or human errors;

(g) Qualitative evaluation of sequence probabilities (if employed);

(h) Justification for event sequences that are considered 'practically eliminated' and justification, ~~with a high level of confidence~~, that they are physically impossible or that they are with a high level of confidence extremely unlikely to arise. It should be noticed that this part is not an evaluation of the consequences of the event

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

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

Transient analysis

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

Computational models

A.16.20. This section should describe the computational models employed, including computer codes or analogue simulations used in the analyses. The description should demonstrate that the models are applicable for the expected range of operational parameters, that they represent all important physical phenomena and that they have been properly verified. The description should also demonstrate that the computational models use conservative approaches in the case of anticipated operational occurrences and design basis accidents, and best estimate approaches in the case of design extension conditions. This section should provide only a summary of mathematical models and computer codes ~~or lists~~ used, referring to detailed descriptions in documents available to the regulatory body. The following should also be provided:

(a) A general description of the model, including the following:

(i) The purpose of the model and its range of application, including the extent or range of variables investigated;

(ii) A summary description of the analytical models and empirical correlations used;

(iii) Any simplifications or approximations introduced in the analysis;

(iv) The degree of conservatism of the methods and correlations;

(v) The numerical accuracy of the model, including the estimated accuracy of results and factors contributing to the uncertainties;

(vi) The method used to combine codes (if a set of codes is used).

(b) A brief description of input data for each model, including the following:

(i) The method of selection of input parameters, including their applicability and their degree of conservatism;

(ii) A listing of input data for each model;

(iii) The sensitivity of the model to particular input parameters.

(c) A summary of results of verification studies, including the following:

(i) Comparisons of model predictions with results of experiments or operation, or with other models that have also been compared with results of experiments or operation;

(ii) Uncertainty in the predictions and the experiments;

(iii) A description of the validation models used;

(iv) A demonstration of adequate numerical accuracy or of the degree of conservatism (for anticipated operational occurrences and design basis accidents);

(v) Confirmation that the modelling represents all important physical phenomena;

(vi) Confirmation that the empirical correlations are conservative, are based on experiment (where practicable) and are appropriate for the range of operational parameters.

Input parameters and initial conditions

A.16.21. The input parameters and initial conditions used in the analysis should be clearly identified. Annex II to this publication provides a list of examples of these. 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.

Results

A.16.22. The results of the analysis should be presented and described in the safety analysis report. Key parameters should be graphically presented as functions of time for 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 (e.g. inlet temperature, average core temperature, hot channel exit temperature);

— Core temperature (e.g. maximum fuel centre line temperature, maximum cladding temperature) and maximum fuel enthalpy;

— Reactor coolant inventory (e.g. total inventory and coolant level in various locations in the reactor coolant system);

— Parameters of the secondary heat exchanger system (e.g. inventory and level, enthalpy, temperature, mass flow rate).

For research reactors with low hazard potential, critical assemblies and subcritical assemblies, parameters should be identified depending upon the design features of the facility and their importance to safety (e.g. measures to address reactivity accidents).

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

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

Classification of damage states

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

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

Derivation of source terms

A.16.27. The source terms, if any, for each bounding sequence mentioned in the previous section of the safety analysis report should be described. These descriptions should include the quantity of radioactive material that might be released from the research reactor, its physical and chemical form, and any other factors necessary to completely specify its potential dispersion in the environment. Factors that 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. Additional information on the derivation of source terms is provided in Ref. [39].

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 the representative person would meet regulatory requirements).

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

Assessment of releases within the reactor building

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

(a) The fission product inventory (or radionuclide inventory for accidents not involving fuel damage);

(b) The nature of the fuel element damage, and the fraction of the fuel cladding damaged;

(c) The fractions of the fission products released from the fuel;

(d) The retention factor and plate-out (deposition of daughter products of a radioisotope onto the surface of another material) factor of radionuclides in water and on surfaces.

Assessment of releases from the reactor building

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

(a) Removal of radionuclides by liquid and gaseous hold-up systems, recirculation systems and ventilation systems, including filter efficiencies;

(b) Surface deposition and resuspension;

(c) Radionuclide hold-up time, decay time and precursor production rate;

(d) Reactor building leak rate or liquid effluent release rate;

(e) Release mode (i.e. single puff, intermittent, continuous) and estimated release duration;

(f) Release point (e.g. stack, ground level).

Assessment of other hazards

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

— Inadvertent criticality;

— Releases from an experiment or the research reactor that are contained but that present a radiation hazard;

— Aqueous spills or other releases of radioactive material that are contained locally;

— Loss of shielding (e.g. a loss of coolant accident that uncovers the reactor core but does not lead to cladding damage).

Evaluation of the radiological consequences

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

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

Methods for analysis of the possible radiological consequences

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

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

— A description of the mathematical or physical models employed, including any simplifications or approximations introduced into the analysis;

— A description of the meteorological data used to perform the calculations;

— A summary of the computer codes or analogue simulations used in the analyses, with reference to detailed descriptions;

— Information on the validation of the calculational methods used, including the restrictions and limitations on their utilization;

— Consideration of uncertainties in the calculational methods used, the performance of equipment, instrumentation response characteristics or other intermediate effects that were taken into account in the evaluation of the results.

Results of dose calculations

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

External exposure

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

Radiation fields

A.16.39. Radiation fields associated with accident conditions, including releases that occur within the research reactor and direct radiation from sources (including the reactor core), that could result in radiation doses due to external exposure should be described, together with estimates of doses to the representative person. The parameters and assumptions used in the analysis should be justified, including the following:

— The quantity of radionuclides released and the timescale of the release;

— Radionuclide decay time and precursor production rate;

— Shielding parameters, buildup factors and scattering (e.g. for gamma radiation from radionuclides in an airborne plume (‘cloud shine’));

— Velocity of propagation, the distance to representative person and the timescale over which doses are calculated.

Aqueous releases

A.16.40. This section should summarize the assessment of aqueous releases and, where appropriate, dispersion in surface waters and groundwater, contamination of the flora and fauna and food chains, and

the consequent doses to individuals and to the population. Reference should be made to paras A.3.11–A.3.14 for data on hydrological and hydrogeological characteristics of surface water and groundwater. The information on potential hazards should include the following:

— Radiation from released fluids;

— Evaporation or airborne radioactive material caused by resuspension of radionuclides from the released fluids;

— Ground contamination;

— Contamination of aquifers and reservoirs on and off the site.

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

— Radionuclide removal by liquid hold-up systems or recirculation systems;

— Potential discharge points, the inventory of radionuclides released, their concentrations in the fluid, the release rate and the mode of release (i.e. single, continuous or intermittent release);

— Radionuclide decay time and precursor production rate;

— Dilution and dispersion characteristics, including migration and retention characteristics of soils, radionuclide movement in hydrogeological formations, the reconcentration ability of sediments and biota, and other effects that might be needed to determine radionuclide movement and exposure pathways;

— Direct and indirect pathways for contamination of the food chain;

— Radionuclide uptake by humans and the consequent doses.

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

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

Atmospheric releases

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

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

— The source term, characterizing it in terms of the radionuclide inventory, the physical and chemical forms, and any other factors necessary to completely specify the dispersion of radioactive material to the environment, including buoyancy;

— Mode and characteristics of the release (i.e. single, intermittent or continuous release, release duration);

— Location of the release and characteristics, including height and diameter of the stack;

— Distance to receptors and intervening terrain;

— Meteorological data, including wind speed and wind direction, precipitation, and data on inversions and other atmospheric stability factors;

— Wake effects of the building;

— Diffusion parameters;

— The physical and chemical forms of radionuclides at the receptor location, and whether they are airborne or deposited;

— Results of dose calculations (for doses due to inhalation, ingestion and ground shine).

Ground contamination

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

Analysis of design extension conditions

A.16.47. Paragraph 6.68 of SSR-3 [1] states:

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

If the results of the analysis do not demonstrate that these criteria are met, additional safety features for design extension conditions should be implemented.

A.16.48. This section of the safety analysis report should present the assumptions used and the results obtained from the analysis of design extension conditions without significant fuel degradation. The analysis should demonstrate with an adequate level of confidence that core melting can be prevented and that there are adequate margins to avoid any cliff edge effects.

A.16.49. This section should also present the assumption used and the results obtained from the analysis of design extension conditions with core melting with subsequent releases of radioactive material to the containment (or to the research reactor building).

A.16.50. This section should also provide identification of the most severe parameters resulting from core melt sequences, and should demonstrate the following:

- That the research reactor can be brought into a state where the confinement function can be maintained in the long term;
- That the structures, systems and components of the research reactor are capable of preventing any early radioactive release or large radioactive release;
- That compliance with the acceptance criteria is achieved by safety features implemented in the design, combined with the implementation procedures or guidelines for severe accident management;
- That the possibility of conditions arising that could lead to an early radioactive release or large radioactive release is practically eliminated. Nevertheless, a good practice would be to implement a dedicated section for practically eliminated event sequences.

A.16.51. This section should also describe the analysis of additional accidents, e.g. a large release of tritiated heavy water, damage of targets, that are postulated for the purposes of emergency preparedness and response.

A.16.52. The scope and content of the information provided for design extension conditions should be similar to that for design basis accidents (see para 16.13-16.46).

Summary

A.16.53. This section should summarize the important results of the safety analysis, including a brief description of the dominant accident sequences. Significant conclusions arising from the analyses should be presented. The effect of uncertainties in the results should be considered and evaluated.

A.16.54. For anticipated operational occurrences and accident conditions, the results of the analyses should be compared with the appropriate acceptance criteria. It should be shown that the criteria set out in paras 2.15–2.21 have been met. An evaluation of the results should demonstrate for anticipated operational occurrences and design basis accidents that the design is acceptable and should confirm the validity of the operational limits and conditions described in Chapter 17 of the safety analysis report.

A.16.55. For design extension conditions, the results of the analysis should demonstrate that the criteria set out in paras 2.18-2.21 have been met.

CHAPTER 17: OPERATIONAL LIMITS AND CONDITIONS

A.17.1. This chapter of the safety analysis report should contain the operational limits and conditions important to safe reactor operation that have been derived from the safety analysis. The operational limits and conditions represent an envelope of parameters, developed by the operating organization, that,

if they are not exceeded, will protect the research reactor and that will protect personnel and the public from exposure and the environment from contamination. The operational limits and conditions should be understood by the responsible operating personnel. The operational limits and conditions include safety limits, safety system settings, limiting conditions for safe operation, and surveillance and administrative requirements. Requirements are established in Requirement 71 of SSR-3 [1], and recommendations are provided in NS-G-4.4 [6].

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

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

(a) The objectives to be met by the establishment of the operational limit or condition (e.g. prevention of situations that might lead to accident conditions).

(b) The applicability of the operational limit or condition, for example, to physical variables related to physical barriers, such as the fuel cladding temperature or pool water level, or to the conditions of these barriers. Sometimes the applicability refers to the equipment set-up, such as the minimum number of measuring channels that are operable.

(c) The specification(s) of the operational limit or condition; for example, the value that is not permitted to be exceeded, or a specific condition on equipment.

(d) The bases for these topics, in particular for the adopted specifications. These bases are normally the design calculations or safety calculations included in the safety analysis, which allow for margins in engineering and measuring uncertainties. However, these bases are sometimes simple conservative assumptions from previous operating experience, or the results of proposed experiments.

Safety limits

A.17.4. The safety limits for important process variables or parameters should be stated and justified by the analyses provided in the safety analysis report. Safety limits normally involve operational parameters, such as fuel temperatures, fuel cladding temperatures, the reactor coolant temperature, the reactor pressure, the reactor power, coolant flow rates and, for pool reactors, the water level above the

core. These safety limits are derived primarily from the results set out in Chapters 5 and 16 of the safety analysis report.

Safety system settings

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

Limiting conditions for safe operation

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

— Core configurations and design limitations (e.g. reactivity coefficients, power peaking factor, burnup limits, minimum and maximum number of the fuel elements and reflector elements, their geometrical arrangements, inspection);

— Minimum number, design and performance of reactivity control mechanisms*;

— Fuel design parameters (e.g. enrichment, fuel type, cladding type);

— Maximum positive reactivity insertion rate*;

— Minimum number of operational measurement systems and control systems for the research reactor and safety set points;

— Structures, components and systems required to provide confinement or containment;

— Operations that necessitate means of confinement or containment;

— Minimum operational equipment for ventilation systems;

— Equipment and performance of the emergency power supply systems*;

— Minimum operational equipment for radiation monitoring systems and effluent monitoring systems, and their safety set points for the different operational stages (e.g. shutdown, operation, fuel handling);

— Limits on effluent releases;

— Limitations on experiments (e.g. reactivity worth, materials);

— Other design limitations important to safety.

Surveillance requirements

A.17.7. This section should describe the surveillance requirements regarding the frequency and scope of tests, showing that the performance levels set by the safety limits and the limiting conditions for safe operation are being met. The requirements for monitoring, inspection, operability checks and calibrations should be included, and the actions to be taken if a system fails should be described. The conditions for continuing operation during repair work or the acceptability of the substitution of replacement equipment for failed equipment should be stated. Recommendations are provided in paras 3.27–3.32 of NS-G-4.4 [6].

Administrative requirements

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

CHAPTER 18: THE MANAGEMENT SYSTEM

A.18.1. The management system is an integrated set of interrelated or interacting elements that establishes policies and objectives, and that enables those objectives to be achieved in a safe, efficient and effective manner. Requirements on the management system are established in GSR Part 2 [20] and recommendations are provided in GS-G-3.1 [21] and GS-G-3.5 [22].

A.18.2. The operating organization is responsible for the development and use of a management system that will ensure conformance with the requirements for every aspect of safety. The objectives and scope of the management system should be established in accordance with the requirements of SSR-3 [1] and with national standards.

~~A.18.2.~~A.18.3. The management system should establish a safety committee (or advisory group) to advise the operating organization on the safety assessment of design, commissioning and operational issues, as well as all relevant aspects of the safety of the reactor and the safety of its utilization.

~~A.18.3.~~A.18.4. This section should describe the management system or should refer to a description of it. A summary should be provided of the items, services and processes to which the management system applies, and of the organizational structure within which the activities are to be planned and implemented. The level of control and verification of quality should also be defined, and the means available for achieving this level should be described.

~~A.18.4.~~A.18.5. This section should describe or should refer to the particular parts of the management system that have been established for the stages of design, procurement, construction, commissioning or operation and decommissioning, as appropriate. The management system procedures should be consistent with the requirements of the research reactor project and its objectives, status and characteristics, and the management system should be acceptable to the regulatory body.

Management system procedures

~~A.18.5.~~A.18.6. This section should describe or refer to the planning, implementation and control of essential activities relating to the management system procedures to ensure that the specific requirements — such as regulatory requirements, design and construction criteria, and acceptance criteria — are correctly applied and fulfilled. In particular, the responsibilities and authorities of the personnel concerned under the management system should be specified.

~~A.18.6.~~A.18.7. This section should describe the procedures covering specific activities under the management system, such as resolution of non-conformances, design changes, design deviations and concessions, and the analysis of their impacts on safety requirements. This section should describe the procedures covering the operating activities performed under the management system. Examples are activities relating to core management and fuel handling, cooling of the core, safety of experimental devices, reactor modifications, procurement and storage of components and materials, and human surveillance.

~~A.18.7.~~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 provisions and the operational procedures to facilitate the decommissioning process. The design basis relating to decommissioning should be described.

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

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

CHAPTER 20: EMERGENCY PREPAREDNESS AND RESPONSE

Emergency plan

A.20.1. This section of the safety analysis report should contain or refer to an emergency plan, which will provide reasonable assurance that response actions can and will be taken in a nuclear or radiological emergency that might occur at the research reactor. Requirements for the plans and procedures for emergency response are established in IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [25].

A.20.2. This section should demonstrate that the emergency plan and procedures are based on accident conditions analysed in the safety analysis report as well as those postulated for the purpose of emergency preparedness and response on the basis of the hazard assessment. This section should also demonstrate that the emergency plan is prepared in coordination with all other response organizations.

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

(a) The emergency response arrangements, giving clear instructions regarding authorities and responsibilities;

(b) The process for identifying and classifying an emergency;

(c) The agreements made with off-site emergency services;

(d) Notification of on-site personnel and, if necessary, off-site personnel;

(e) Notification of government authorities and local authorities;

(f) Reliability of communications between the control room and emergency response facilities and with on-site and off-site emergency response organizations;

(g) Protective actions and other response actions;

(h) Equipment items available to deal with an emergency and their location;

(i) Arrangements with medical facilities to treat individuals contaminated or overexposed to radiation;

(j) Training of personnel;

(k) Frequency and scope of training, drills and exercises;

(l) Adequacy of resources to implement the emergency plan.

A.20.4. For research reactors with low hazard potential as well as critical assemblies and subcritical assemblies the type and nature of details will depend on the results of the hazard assessment, as required in GSR Part 7 [25] and further described in Ref. [40].

Emergency procedures

A.20.5. This section should demonstrate that the emergency plan will be implemented by means of emergency procedures. The emergency procedures should include the specific mitigatory and response actions that will be taken in a nuclear or radiological emergency.

A.20.6. This section should contain information on the arrangements for periodic review of the emergency plan, the emergency procedures and their implementation, to ensure that the requirements of new experiments or research reactor modifications are included.

A.20.7. The emergency procedures should contain guidance on limits to values for restricting exposure of emergency workers, as well as generic and operational criteria for use in emergency preparedness and response as described in GSR Part 7 [25].

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Annex I

APPROACH TO AND METHODS OF SAFETY ANALYSIS

I–1. This annex presents examples of methods for developing a safety analysis for a research reactor. The approach to developing a safety analysis is to consider postulated initiating events for credible accidents, using a deterministic method to estimate the maximum possible releases to the environment. More detailed information on the development of deterministic safety analysis can be found in Ref. [I–1]. ~~Deterministic and p~~Probabilistic methods may be used ~~in a~~ complementary to deterministic methods~~way~~ to evaluate which accident sequences are of a higher likelihood; they will also be useful for evaluating relative rankings of risks, and hence for determining countermeasures. They may also be used for identifying any latent weaknesses in the design and for quantifying the value of possible improvements or modifications. However, probabilistic safety assessment is not treated in this Safety Guide, and consequently only deterministic methods are addressed here. For further information on application of probabilistic safety assessment to research reactors, see Refs [I-2, I-3]. Recommendations on the development of probabilistic safety analysis are provided in in Refs [I-4, I-5].

I–2. These considerations cover a wide range of research reactors and thus might contain information that is not applicable to all research reactors.

METHODS FOR IDENTIFICATION AND SELECTION OF POSTULATED INITIATING EVENTS

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

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

(a) Lists of postulated initiating events in research reactors. A list of possible postulated initiating events in research reactors is provided in para. 3.2~~1~~3 of this Safety Guide.

(b) Engineering evaluation. Potential sources of radiation and types of radiological hazard within the research reactor are identified, and a systematic review of the research reactor design, operations and site factors is made to identify occurrences that could lead to radiological hazards.

(c) Operating experience. Past experience from the research reactor or from similar facilities, including experience derived from the examination of safety reports and the IAEA's Incident Reporting System for Research Reactors (IRSRR) database, can be used to develop or to supplement the list of postulated initiating events.

(d) Logical analysis. An example is a top-down logical model known as a master logic diagram, which is similar to a fault tree.

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

(a) Incredible postulated initiating events, i.e. postulated initiating events that are not possible for the research reactor under study.

(b) Very rare postulated initiating events, i.e. postulated initiating events whose frequency of occurrence is so low that they could be candidates for rejection on a probabilistic basis using statistical data or conservative estimates. Combinations of mutually independent initiating events, each having a low frequency of occurrence, would also fall into this category.

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

(a) Postulated initiating events that involve similar safety functions and which determine the design parameters of the safety systems;

(b) Postulated initiating events that involve similar safety functions and which determine the parameters of the additional safety features for design extension conditions;

(c) Postulated initiating events that have a similar influence on reactor behaviour or on structures, systems or components and for which similar calculational models are used;

(d) Postulated initiating events that can assist in the selection of limiting cases for analysis in each group;

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

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

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

METHODS FOR EVENT SEQUENCE ANALYSIS

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

I–9. The sequences include the response of the reactor core, the research reactor systems, engineered safety features and safety features for design extension conditions, as well as human interactions. Possible sequences for the case in which a system fails are described in detail for accident conditions. The following points are considered:

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

(b) Identification of significant occurrences on a timescale, for example, neutron flux monitor trip and start of insertion of control rods;

(c) Indications of correct and incorrect functioning of normally operating reactor instrumentation and controls;

(d) Additional failures assumed for safety features for design extension conditions;

(e) Evaluation of the three main safety functions (shutting down the reactor, cooling the fuel and maintaining confinement of radioactive material), including an indication of the correct functioning of the reactor protection system and safety systems as well as of safety features for design extension conditions;

(f) Credited operator actions for functioning of manually operated safety systems for design basis accidents and safety features for design extension conditions;

(g) Credited protective measures for design extension conditions;

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

(i) Conditions for termination of the analysis, including, for example, situations in which stable conditions are reached (no exposures or releases), or if the likelihood of the sequence becomes so low that further analysis is not warranted, or if all levels of defence against the initiating event are exceeded and the sequence leads to significant exposure of personnel or to the release of radioactive material.

I–10. Rules or conventions are established in order to determine the response of reactor systems. These rules or conventions refer to the following:

(a) The effect of single, random failures;

(b) System qualification (or lack of qualification) under accident conditions;

(c) Safety systems, the reactor protection system, and engineered safety features as well as safety features for design extension conditions, including their reliability in quantitative terms, if applicable;

(d) Support systems, such as normal and emergency electric power and for cooling;

(e) Redundant trip parameters;

(f) Actions of systems that are independent;

(g) Operator actions (e.g. in terms of response time, display of information on a console);

(h) The effect of failures assumed for safety systems for design basis accidents or safety features for design extension conditions;

(i) For carrying out frequency or probability evaluations to assess the system response, the extent to which such evaluations will be used and the methods to be employed (including validation).

I–11. Rules or conventions are developed in order to determine those event sequences that are excluded from further analysis. Such rules could be based on:

(a) Qualitative and quantitative frequency or probability arguments justifying the exclusion of or practical elimination event sequences ~~that are practically eliminated~~;

(b) Justification, including design and qualification, for crediting structures, systems and components in the event sequences.

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

(a) Investigations carried out to identify the specific causes of dependent failures or human error;

(b) Evaluation of the effect of human error on either initiating an accident or worsening the development of accident sequences;

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

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

(a) The known or estimated frequency of the initiating event, for example, loss of electrical power supply and failure of a pump or rupture of pipe work.

(b) Methods for estimating the probability of failure of each of various safety systems or safety related systems.

(c) Rules regarding the subdivision of event sequences to avoid (or to accommodate) an arbitrary subdivision at the systems level, as well as an arbitrary subdivision of initiating events (e.g. a set of similar pipe breaks rather than the generic event, specific meteorology) that can lead to many similar event sequences and that might have a low cumulative frequency.

(d) Conventions for determining the likelihood of event sequences, with due regard to the effects of a dependent failure. For example, the probability of loss of a particular safety function might be determined as the product of the failure probability of the associated systems and the cumulative frequency of similar initiating events if these systems and events are independent.

I–14. Limiting or bounding event sequences in each category are selected for further analysis, to reduce the number of events to be analysed. Consideration is given to the following:

(a) Conservative assumptions made in the categorization of events to provide a safety margin (e.g. uncertainty allowances and not taking full credit for mitigating actions of systems or of operator response) or to ensure that all sequences in a category have been covered, starting from all permitted states in the operating envelope;

(b) The methods used to choose bounding sequences in a category of events that represent the entire category and not just specific sequences, including those sequences that have the most severe consequences.

METHODS FOR EVALUATION OF EXTERNAL EVENTS AND SPECIAL INTERNAL EVENTS

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

I–16. In general, design qualification is an accepted practice for protection against external events in combination with site evaluation, see SSR-1 [I-6] (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 hazards can be summarized as follows:

(a) The potential of an event to occur at the research reactor site for each hazard 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 hazard is identified.

(c) A relationship between the severity of the hazard and the frequency of occurrence is determined, or a model appropriate to the hazard in the site region is constructed.

(d) A particular design basis frequency of occurrence is established (the defined recurrence frequency) for which protection is provided to preserve structures, systems and components important to safety.

(e) The design parameters for the hazard are evaluated, corresponding to the design basis frequency of occurrence.

I–17. Design extension conditions are specified for a range of frequency of occurrence for which additional safety features for design extension conditions have to be provided to fulfill the main safety functions, especially the confinement function.

I–18. Design qualification can be undertaken to prevent failure of pressure retaining components. In this case, the safety analysis report describes the design and construction standards used (e.g. acceptable engineering codes and practices) to prevent structural failures and to maintain the required safety functions. Reference may be made to the appropriate chapters of the safety analysis report (see Chapters 2 and 3 of the Appendix to this Safety Guide).

Qualitative evaluations

I–~~18~~19. Consideration is given to the conditions under which qualitative evaluations are used in the safety analysis to treat particular event sequences; for example:

(a) Treatment of fault sequences that are not limiting (e.g. they are bounded by other initiating events);

(b) Justification of design measures to prevent certain fault sequences or to demonstrate that the events would not be considered credible;

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

I–~~19~~20. Such qualitative arguments are used with caution and after consultation with the regulatory body concerning their acceptability.

ACCEPTANCE CRITERIA

I–~~20~~21. The significant results of the safety analysis for anticipated operational occurrences, design basis accidents and design extension conditions and the comparison with the acceptance criteria (see paras 2.14–2.20 of this Safety Guide) are presented in the safety analysis report.

REFERENCES TO ANNEX I

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Annex II

EXAMPLES OF INPUT PARAMETERS AND INITIAL CONDITIONS

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

— Moderator (and coolant) temperature coefficient of reactivity;

— Moderator void coefficient of reactivity;

— Fuel temperature coefficient of reactivity;

— Effective prompt neutron lifetime;

— Delayed neutron fraction(s);

— Average heat flux;

— Maximum heat flux;

— Minimum departure from nucleate boiling ratio;

— Minimum critical heat flux ratio;

— Margin to onset of significant void;

— Margin to onset of flow instability;

— Axial power distribution;

— Radial power distribution;

— Hot channel factor;

— Core coolant flow rate;

— Core coolant inlet and exit temperatures;

— Core coolant inlet and exit pressures;

— Hot channel coolant exit temperature;

— Maximum fuel centre-line temperature;

— Fuel cladding temperature;

— Reactor coolant system inventory;

— Coolant level in the reactor vessel or tank;

— Coolant level in the components (e.g. delay tank);

— Heat exchanger mass flow rate and temperature;

— Fuel burn-up (e.g. exit burn-up, ratio of peak to average burn-up);

— Control rod worth (e.g. differential and total, shutdown margin);

— Maximum reactivity insertion rate;

— Maximum reactivity excess;

— Minimum shutdown margin;

— Maximum reactivity worth for experiment.

II–2. For subcritical assemblies, input parameters and initial conditions will depend on specific design features of the research reactor and their importance to safety (e.g. input parameters related to reactivity insertion and delayed neutron fraction and maximum reactivity worth for experiments).

Annex III

ITEMS TO BE CONSIDERED IN THE DESCRIPTION OF THE RESEARCH REACTOR

III–1. This Annex provides examples of items to be considered in the description of the research reactor.

SUMMARY DESCRIPTION

III–2. A brief description of the following aspects of the research reactor can be provided (examples of items that might not be applicable to some subcritical assemblies, depending on their design, are shown below with an asterisk (*)):

(a) The purpose of the research reactor (neutron source, irradiation facilities, material testing).

(b) The type of research reactor (e.g. pool, tank):

— Type of fuel;

— Moderator;

— Reflector;

— Core configurations (i.e. fuel elements, reflector elements, reactivity control mechanisms, experimental devices, nuclear instrumentation);

— Reactivity control mechanisms for power regulation (control rods or shim rods)*;

— Reactivity control mechanisms for shutdown (safety rods)*.

(c) Coolant*.

(d) Mechanical reactor design:

— Reactor vessel or reactor pool;

— Core support structures;

— Reactor bridge;

— Beam tubes and in-core test facilities;

— Natural circulation provisions (e.g. flapper valves, coolant gate)*.

(e) Shielding.

(f) Summary table of main design and performance characteristics:

— Rated power*;

— Neutron flux;

— Core coolant flow*;

— Core inlet and outlet temperatures*;

— Power density*.

REACTOR STRUCTURES

III–3. A detailed description of the following items can be provided:

(a) Reactor pool and/or vessel;

(b) Core support or grid plate;

(c) Reactor bridge;

(d) Reflector;

(e) Shielding (including movable shielding);

(f) Supports for core instrumentation;

(g) Beam tubes;

(h) In-core test facilities;

(i) Provisions for natural circulation*.

The description includes materials and dimensions, supported by drawings. The effects of ageing such as corrosion, fatigue and neutron irradiation on the lifetime of items important to safety are described.

REACTIVITY CONTROL MECHANISMS, REACTOR SHUTDOWN SYSTEM

III–4. The description of the function of the mechanical design and the electrical design includes the materials and dimensions, and is supported by drawings. The reactivity control mechanisms and their instrumentation, such as their position or status (coupled and/or decoupled), are presented, together with the insertion time and interlocks. The effects of ageing such as corrosion, fatigue and neutron irradiation on the lifetime of the mechanical and electrical components are also described. The following safety related design parameters are presented:

— Speed of control rods*;

— Insertion time of shutdown rods*;

— Maximum number and height of withdrawal of rods*.

Measures to avoid ejection of the control rods and shutdown rods are also described*.

FUEL ELEMENTS

III–5. The fuel used, including the uranium enrichment and the type of fuel, is specified. The description of the fuel elements, supported by drawings, and the main characteristics of the fuel elements are presented, including the following:

(a) Thickness of cladding;

(b) Length of active zone;

(c) Width of coolant channel;

(d) Number of fuel plates and/or pins;

(e) Cladding material;

(f) Uranium loading.

If fuel elements are used that contain channels for the movement of neutron absorbing blades or neutron absorbing rods, or integral burnable neutron poisons they are described. A summary of the experience with the fuel is provided*.

REACTIVITY CONTROL SYSTEMS

III–6. In addition to the description of the reactivity control systems, supported by drawings, the main dimensions and information on the neutron absorber material used and on the experience with these or with similar reactivity control systems are provided*.

Annex IV

TYPICAL RADIATION SOURCES IN A RESEARCH REACTOR

IV–1. Examples of possible radiation sources in a research reactor are the following:

— The fission product inventory of the reactor core;

— Spent fuel storage;

— Concentration of fission products, activation products and corrosion products in the pool or the reactor coolant system and in related systems such as the purification system;

— Equipment, systems and piping containing activation sources;

— Solid and liquid radioactive waste and radioactive waste management facilities, and leakage or spills from these facilities;

— Gaseous radioactive material from the pool, cooling systems, cover gas systems, reflector systems and experimental facilities connected to ventilation systems, or any leakage from these systems;

— Filters from the ventilation systems;

— Airborne radioactive material in areas normally occupied by personnel;

— Experimental facilities with the potential to generate activated material or other radioactive material, or facilities for the storage and handling of such material, including sample activation and/or irradiation facilities, in-core experiments and hot cells;

— Material irradiated by the research reactor;

— Neutron startup sources;

— Sources for testing and calibration of radiation monitoring equipment.

IV–2. Applicability of these items to subcritical assemblies will vary, depending on the design of the research reactor and the hazards associated with the research reactor. However, the main radiation sources in subcritical assemblies are typically fuel, neutron source and sources for testing and calibration of radiation monitoring equipment.

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