SAFETY SERIES No. 35-G1

SAFETY GUIDES

safety series

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

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1994

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

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FOREWORD

The first nuclear research reactor went critical on 2 December 1942. At present, there are over 300 research reactors in operation in about 60 IAEA Member States. There has evolved a 50 year tradition of emphasis on nuclear safety in the design and operation of research reactors.

From the inception of the IAEA in 1957, there has been broad interest at the IAEA in the benefits to be derived from the safe operation of research reactors by IAEA Member States. These benefits can be gained not only in the traditional areas of nuclear power technology, radioisotope production, nuclear medicine and personnel training but also in the vital areas of materials development and environmental pollution control. To achieve these benefits, the safety of research reactors must be ensured. The IAEA has a long tradition in the area of research reactor safety.

The first publication of the IAEA on research reactor safety was as early as 1960 (Safety Series No. 4), and this subject has since received continuous attention. In 1971, Safety Series No. 35 on the Safe Operation of Research Reactors and Critical Assemblies was issued. A major revision of this Safety Series, with the same title, was published in 1984. While this publication provided practical guidance on safe operation, it did not deal with many other aspects which arise in the course of a research reactor project and which influence safety.

To remedy this shortcoming, basic principles and requirements for the safety of research reactors and critical assemblies have been compiled in two Safety Standards, which present codes for the safety of nuclear research reactors, on design (for the first time) and on operation (Safety Series Nos 35-S1 and 35-S2). These codes, which supersede those of Safety Series No. 35 of 1984, also include the essential safety requirements for siting, quality assurance and regulatory control of research reactors.

In addition to these two Safety Standards, the research reactor safety programme includes Safety Guides and Safety Practices which provide detailed guidance on safety in a number of areas, such as commissioning, utilization, safety analysis and assessment, radiation protection and operating procedures.

This Safety Guide presents guidelines, approved by international consensus, for the preparation, review and assessment of safety documentation for research reactors such as the Safety Analysis Report. While the Guide is most applicable to research reactors in the design and construction stage, it is also recommended for use during relicensing or reassessment of existing reactors.

NOTE ON THE INTERPRETATION OF THE TEXT

The appendix is considered to be an integral part of the Guide and it has the same status as the main text. However, annexes, footnotes and bibliographies are only included to provide additional information or practical examples that may be helpful to the user.

In this document the word 'shall' denotes a firm requirement, the word 'should' denotes a desirable option, and the word 'may' denotes permission (neither a requirement nor a desirable option).

In several cases, the wording 'shall consider ...' or 'shall ... as far as applicable...' is used. In these cases it is essential to give the matter in question careful attention, and the decision must be made in consideration of the circumstances of each case. The final decision must be rational and justifiable and its technical grounds must be documented.

CONTENTS

1. INTRODUCTION

BACKGROUND

101. This Safety Guide is part of the set of publications prepared within the framework of the IAEA Research Reactor Safety Programme (RRSP) which covers all the important areas of research reactor safety. This set includes Safety Standards, Safety Guides and Safety Practices in the IAEA Safety Series. The Safety Standards are the top level publications, which establish the objectives and recommend requirements that must be met to ensure adequate safety in all stages of the lifetime of a research reactor. The Safety Guides and Safety Practices deal with the above mentioned areas, providing recommendations on how to carry out the requirements established in the Safety Standards, giving guidance and presenting international practices in the areas of research reactor safety. In addition, a Safety Guide may introduce more specific consequential requirements, connected with those in the Safety Standard to which it refers. A list of publications related to the safety of research reactors is given at the end of this Guide.

102. Owing to the particular characteristics of research reactors, the safety aspects related to design and operation have been given special emphasis and have been incorporated in two Safety Standards, the Codes on the Safety of Nuclear Research Reactors on Design (Safety Series No. 35-S1) and on Operation (Safety Series No. 35-S2). These characteristics include the large variety of designs, the wide range of powers, the different modes of operation and purposes of utilization, the particularities of siting, and the differences among organizations operating research reactors, in particular concerning their resources. These characteristics require flexibility in the setting and the fulfilment of basic requirements when dealing with certain specific topics. These circumstances have been taken into account in the present Safety Guide.

103. The organizations involved in ensuring the safety of research reactors and the protection of the general public, the site personnel and the environment have a number of responsibilities which are interrelated. Most important are the performance of the safety analysis by the operator and its review and assessment by the regulator, as well as the production, submission and evaluation of other safety related documents during the licensing process or in other special circumstances, such as modifications and experiments. The present Safety Guide develops the general concepts in these areas which are presented in the two above mentioned Safety Standards. Therefore, this Safety Guide should be read in conjunction with them. Furthermore, this publication considers other aspects of reactor operation normally

included in the Safety Analysis Report (SAR), such as operational limits and conditions, commissioning and operating procedures, which are planned to be further developed in other publications of the set.

104. The use of the terms safety assessment, safety analysis and review and assessment in this Guide requires explanation. Safety assessment is used synonymously with safety evaluation, i.e. a process to evaluate the safety of the reactor. Therefore, the safety assessment of the reactor may include many and varied activities during the licensing process, such as analyses and preparation and submission of documents for review, and may involve several organizations. The term safety assessment is in general used in the RRSP instead of safety analysis, which in this Guide has a more specific meaning. Safety analysis is the analysis of incidents or hazards or risks. In this regard, the safety analysis is performed by the designer or the operator. The safety analysis is one of the most important parts of the reactor licensing process, and it is required to be included in the SAR and submitted to the regulatory body for review and assessment. Therefore, the term review and assessment is specifically used to describe the regulator's responsibility of assessing safety documentation submitted in support of a licence application.

OBJECTIVE

105. The objective of the present Safety Guide is to provide recommendations on how to fulfil requirements established for the safety assessment in the licensing process, such as the responsibilities and functions of the organizations involved (Safety Series No. 35-S2, paras 301–303 and 401–403) and steps towards the issue of the licence (Safety Series No. 35-S1, paras 308–313). In particular, this Guide gives guidelines for the performance of the safety analysis (Safety Series No. 35-S1, paras 509–517) and for the preparation of the SAR (Safety Series No. 35-S2, paras 501–505). Finally, detailed guidance is presented also on the performance of the review and assessment of the SAR by the regulatory body.

SCOPE

106. This Safety Guide is intended to have general applicability to any type of research reactor. However, the particular characteristics of research reactors, as mentioned in para. 102, require some flexibility in the fulfilment of the requirements and the application of the guidelines given in this Safety Guide. Therefore, it is necessary that users of this Safety Guide make a conscious and justifiable selection of the information offered. In addition, the extent of detail in the application of the

guidelines will depend on the discussions and agreements between the operating organization and the regulatory body, with the final decision to be made by the latter.

107. The present Safety Guide focuses mainly on research reactors of a capacity of up to a few tens of megawatts. Therefore, the amount of detail required in the SAR for small research reactors (below a few tens of kilowatts) and critical assemblies may be substantially reduced. Nevertheless, it is strongly recommended that the applicability of all items included in this Safety Guide be assessed. On the other hand, the requirements for a proper licensing process of high powered or otherwise advanced or complex research reactors are beyond the scope of this Safety Guide. For these cases, the IAEA safety publications for power reactors may provide adequate additional guidance.

108. Although this Safety Guide focuses mainly on newly designed and constructed research reactors, its content is applicable to any re-licensing process or reassessment requested by the regulatory body or decided on by the operating organization. In any case, the justification for the approach selected on the basis of this Safety Guide shall be provided to the regulatory body.

STRUCTURE

109. This Safety Guide addresses two interrelated issues: the safety assessment of the reactor and the preparation of the SAR for this purpose. It also provides general guidance on the necessary steps towards the licensing of a research reactor. The main reason for presenting such different topics in a single Safety Guide is their interrelationship and their joint importance in the licensing process. In addition, the Guide has been structured in the light of its function in the overall RRSP planned by the IAEA. The Safety Guide consists of a main text, an appendix and four annexes. The main text covers the overall safety assessment of the reactor, establishing specific requirements and giving guidance in three sections (Sections 2–4). These requirements and guidelines are considerably extended and supplemented by the material presented in the Appendix and in Annexes I–IV.

110. Section 2 describes the licensing process by which the safety of the facility and the issue of licences is controlled and determined. This section covers the roles and responsibilities of the organizations involved in the process, details of the information that the operating organization submits to the regulatory body for review and assessment, and the acceptance criteria against which the review and assessment are conducted.

111. Section 3 presents general guidance on the production of the SAR, in particular the preparation of the safety analysis by the operating organization. Frequent references are made to the Appendix for specific requirements, recommendations and related guidelines.

112. Section 4 gives general guidance on how the regulatory body should conduct the review and assessment of the safety of the facility. Reference to the Appendix is made for specific requirements and guidelines related to the overall SAR.

113. The Appendix is a comprehensive guide on the preparation of an SAR for a research reactor having the characteristics discussed in paras 102 and 106–108. It gives recommendations for the standard content of the SAR and establishes the specific basis for the production of safety related documents by the operating organization, and for their review, assessment and approval by the regulatory body. This recommendation is made in spite of the variety of existing practices in this area in IAEA Member States. The adoption of a standard format will particularly facilitate the review and assessment as well as the independent peer reviews that may be requested.

114. The Appendix is divided into 20 chapters, dealing with the standard specific topics that are addressed in an SAR for review and assessment by the regulatory body. The chapter headings of the Appendix are, in general, the headings that may be appropriate for the SAR. The areas in which basic information is required by the regulatory body, such as site characteristics, reactor (and safety system) descriptions, conduct of operations, commissioning, safety analysis, operational limits and conditions, quality assurance (QA), radiation protection and emergency planning, are emphasized. In particular, considerable attention is given to the safety assessment of modifications and experiments as related to the usage of the reactor.

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

2. REQUIREMENTS FOR SAFETY ASSESSMENT IN THE LICENSING PROCESS FOR A RESEARCH REACTOR

ASSESSMENT RESPONSIBILITIES AND CRITERIA

201. For a research reactor to be built (or to undergo a major modification), adequate assurance of the safety of the facility shall be provided to the population around the site where it operates. This assurance comes from the government, which ensures that an adequate legal and regulatory basis for assessing the safety implications of the project is available. This basis shall include the establishment of an independent regulatory body. The Code on the Safety of Nuclear Research Reactors: Design (Safety Series No. 35-S1), and the Code on the Safety of Nuclear Research Reactors: Operation (Safety Series No. 35-S2) establish general requirements for the framework of the system for ensuring safety, including the licensing process.[1]

202. The regulatory body shall be effectively independent of the operating organization. To be effective, the regulatory body shall be provided with the legal powers, statutory authority and resources needed to ensure that it can fulfil its responsibilities and functions. Such powers normally include the ability to regulate the research reactor project by the issue of licences, with attached conditions, and by inspections of performance and conformance.

203. The objective of the regulatory body is to ensure that the general public, the environment and the operating staff are protected from the possible adverse effects of the research reactor project. To accomplish this, the regulatory body shall establish safety policies, principles, guidance, criteria and regulations upon which to base its regulatory action. It shall also review and assess the safety information submitted by the applicant, and administer the relevant regulations (e.g. by issuing, amending or revoking licences or licence conditions), including carrying out compliance inspections and audits, taking enforcement action and providing information to other relevant bodies or to the general public, as appropriate.

204. Notwithstanding these responsibilities of the regulatory body, the responsibility for the safety of the facility and for demonstrating an adequate level of safety shall rest with the operating organization, not the regulatory body.

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[1] Further guidance on the development of a governmental regulatory body for nuclear power plants has been provided in Safety Series No. 50-C-G (Rev. 1), Code on the Safety of Nuclear Power Plants: Governmental Organization. This guidance should be adapted for application to research reactors.

205. One of the ways in which the operating organization demonstrates that it has achieved adequate safety is through the information normally incorporated in an SAR. This information also constitutes the prime basis for the regulatory decision on licensing the nuclear facility and the requirements against which the facility is inspected.

206. The licensing process may vary among Member States, but in all cases it follows the stages discussed in this section. Control over nuclear safety is maintained primarily through the issue of governmental licences which authorize, in stages, the development of the research reactor project and which place conditions on the licensee. Therefore, a primary task of the regulatory body is to determine whether or not to approve the application for a licence on the basis of its review and assessment of the proposals of the operating organization.

207. The content of the proposal may vary among Member States, depending upon their particular legal and regulatory system. However, the general principles and requirements should follow those established by the IAEA in this Safety Guide[2], in view of the fact that the depth of any information provided in support of a licence application should be commensurate with the potential hazard associated with the facility under consideration and the particular stage of the licensing process.

208. The major stages of the licensing process shall encompass the regulation of:

— Siting;
— Design and construction;
— Commissioning;
— Operation, including utilization and modification[3];
— Decommissioning.

209. Licensing is an ongoing process, starting at the stages of site planning and the feasibility study and continuing up to and including decommissioning of the reactor facility. While licensing steps and procedures vary among Member States, the first formal licensing action may be the approval of the safety concept and the design and issue of a construction licence. In some cases, only a single licence is issued for the

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[2] For further guidance on the subject see Safety Standard No. 50-C-G (Rev. 1) and Safety Guides 50-SG-G2, 50-SG-G3 and 50-SG-G8.

[3] Although utilization and modification of research reactors are processes comprising activities that are normally included in the operation (Safety Series No. 35-S2, paras 1201–1210 and 1301–1305), they may be considered as separate stages in the licensing process because their safety implications lead to a large number of review and assessment activities which are repeated many times during the reactor's lifetime (see paras 233–236 of the present Safety Guide).

project, but certain conditions are attached to it in order to effect control over subsequent stages. Despite these differences between national practices, a detailed demonstration of nuclear safety, which includes an adequate safety analysis, shall be submitted by the operating organization, and shall be reviewed and assessed by the regulatory body before it authorizes progress of the project to the next stage.

Operating organization

210. The operating organization has the overall responsibility for ensuring safety during all stages (listed in para. 208) of the lifetime of its facility. Compliance with the requirements imposed by the regulatory body shall not relieve the operating organization of the fundamental obligation to ensure the protection of site personnel, the general public and the environment. The operating organization shall demonstrate to the regulatory body that this responsibility will be discharged.

211. The operating organization shall submit the requested information to the regulatory body in good time. It shall be the responsibility of the operating organization to make arrangements with the vendors to ensure the availability of information. It shall also be the responsibility of the operating organization to provide new information and alterations to previously submitted information to the regulatory body.

212. The format and content of documents submitted by the operating organization in support of a licence application should be based on the information presented in this Safety Guide. However, the regulatory body may require additional information, depending upon the regulatory practices of the Member State.

213. The review and assessment of the information by the regulatory body is a continuous process. Sections of the SAR or other documents shall be submitted to the regulatory body at an early stage, in accordance with an agreed programme (see para. 222). This approach will permit a systematic assessment and approval procedure and will prevent unnecessary delays in the licensing process.

Regulatory body

214. The regulatory body shall be responsible primarily for determining that the proposed research reactor can be sited, constructed, commissioned, operated, utilized, modified and decommissioned without undue radiological risk to the site personnel, the general public and the environment. In connection with this, the regulatory body shall:

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

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

215. The primary basis for the review and assessment of the nuclear safety aspects of the proposed reactor is the information contained in the SAR submitted by the operating organization. The regulatory body shall make a determination regarding the following points:

- Whether the operating organization has provided the necessary and adequate information for the purpose and scope of the review and assessment (para. 401);
- Whether this information is in compliance with the requirements of all applicable rules and regulations;
- Whether this information is accurate; this may be done by independent checks of the design, including calculations, and by inspections of the programmes and facilities (e.g. QA);
- Whether the engineering solutions, and in particular any novel solutions, are feasible and capable of achieving the design objectives with regard to nuclear safety.

216. In collaboration with the operating organization, the regulatory body shall establish at an early date a schedule for the submission of documents in support of the licence application (para. 222). The regulatory body shall prepare a programme of review and assessment appropriate to the stages in the licensing process (paras 402—407). The regulatory body shall follow as closely as possible the development of the reactor facility, from the stage of site selection through design, construction, commissioning, operation and, where appropriate, modification of the facility.

Acceptance criteria

217. Each IAEA Member State should develop its own approach to acceptance criteria, depending upon its particular legal and regulatory framework. In some cases this may include quantified criteria and safety goals, whereas in other cases it may be restricted to achieving compliance with the requirements established by law or agreed upon between the operating organization and the regulatory body. In either case, the acceptance criteria shall demonstrate an adequate application of the principles for safe design and operation embodied in the IAEA Safety Standards. These

principles include the radiation protection objectives as stated in paras 202 and 203 of Safety Series Nos 35-S1 and 35-S2, which refer to the IAEA Basic Standards for Radiation Protection and the Recommendations of the International Commission on Radiological Protection.[4]

218. If both the operating organization and the regulatory body develop acceptance criteria to reflect the respective philosophies, the set of acceptance criteria agreed upon by both organizations must be satisfactory to the regulatory body.

219. Acceptance criteria should be stated for both the operational states of the reactor and the accident conditions considered in the design of the facility. Such criteria will vary among Member States; they may include considerations such as those listed below.

(a) Radiological criteria, such as:
   — ALARA levels;
   — Dose limits (or targets) for facility staff, including experimenters, workers at the reactor site and the general public;
   — Release limits to the environment; and
   — Risk criteria (where applicable).

(b) Performance criteria, including:
   — Limits to fuel cladding damage;
   — Limits to damage of the primary coolant system boundary;
   — Limits to damage of the containment systems;
   — Maintenance of core cooling; and
   — Frequency limits for certain anticipated operational occurrences and for particular accident conditions, including frequency limits for significant fuel cladding damage.

INFORMATION REQUIREMENTS DURING THE STAGES OF THE LICENSING PROCESS

220. The operating organization shall provide to the regulatory body all relevant information on the basic safety approach for the facility. The manner in which this information is requested, prepared and submitted for assessment differs among

[4] See Basic Safety Standards for Radiation Protection, 1982 Edition, IAEA Safety Series No. 9, and the 1990 Recommendations of the International Commission on Radiological Protection, Publication 60, Pergamon Press (1991), for the radiation protection principles. See also Basic Safety Principles for Nuclear Power Plants, IAEA Safety Series No. 75-INSAG-3, for the technical safety objective. (See the Selection of IAEA Publications Related to the Safety of Research Reactors at the end of this book.)

Member States. However, this information is normally included in the SAR; it is described comprehensively in the Appendix of this Safety Guide. Guidance for the preparation and presentation of the SAR is specified in Chapter 3 and guidance for its review and assessment is given in Chapter 4. The following paragraphs provide a summary of the information which is normally required for each of the stages of the licensing process. The sequential request of information may lead to successive updating, with each version of the report corresponding to a particular stage in the licensing process, as outlined in para. 209.

221. The production of the SAR should begin as early as possible in the project to allow the designers to derive the maximum benefit from the safety analysis, as well as to allow the regulatory body to become familiar with the design and the safety of the reactor. The amount of information provided in the SAR, at each stage, shall 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.

Schedule for the submission of information

222. A schedule should be drawn up which indicates the time-scale for the production of the different chapters of the SAR. Since the approval of one stage is normally required before commencement of the next stage, it is important that the SAR be made available for review and assessment on a time-scale that has been agreed upon by the regulatory body. Some estimate of the size and scope of the analyses should be conveyed to the assessor. In this time-scale, reasonable periods of time shall be allotted for each assessment phase such that each phase can be completed before commencement of the next phase (see paras 402 and 403).

Siting

223. Although site approval is not a formal licensing stage in some Member States, the operating organization shall be required, at this stage or at least before receiving agreement to proceed with construction, to provide sufficient information to demonstrate to the regulatory body that the proposed site is suitable for the type and design of the proposed research reactor. Possible fundamental difficulties that must be resolved during the subsequent stages of the licensing process shall be identified. Information on the site itself and preliminary information on the reactor and its interaction with the site shall be provided. In addition, a preliminary statement of the potential radiological impact on the site personnel, on the population in the surroundings and on the environment shall be provided.

Design and construction

224. To obtain a construction licence or an agreement for the start of construction, the operating organization shall submit information which demonstrates that the design will result in a safe facility and that construction will achieve the design intent. The information shall contain a description of the design of the reactor and the associated safety and process systems, and it shall present the results of the safety analyses which demonstrate the adequacy of the design of safety related structures, systems and components. This information should be submitted in the form of an SAR, which may be preliminary and subject to updating as the project proceeds.

225. Those aspects of the design which must be submitted to the regulatory body for assessment and approval before the design is finalized should be identified in order for the activities to proceed smoothly while the reactor is under construction. The information should be updated and submitted to the regulatory body as the detailed design and construction of the reactor proceed. In some cases, revised versions of the documents will be sufficient; in other cases, technical supplements may be appropriate.

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

227. The interaction between the operating organization and the regulatory body will be intensive at this stage, with the regulatory body inspecting the design and construction activities to confirm that the intentions stated in the SAR have been fulfilled and that management systems, such as those for QA, are operating effectively.

Commissioning

228. When construction is sufficiently advanced, the information contained in the SAR shall be updated in order for the operating organization to obtain the required authorizations for commissioning. The information referred to in paras 224–227 shall be updated and submitted to the regulatory body and will form the basis for the agreement for the start of commissioning.

229. The updated and revised SAR shall include the commissioning programme and demonstrate its adequacy (see Safety Series No. 35-S2, paras 801–809). The 'as-built' reactor, the analyses of postulated accidents and the capability of the safety systems to limit the consequences of postulated accidents shall also be fully documented in the above mentioned version of the SAR.

Operation

230. In its application for an operating licence, the operating organization shall submit all of the information referred to in the preceding sections. In addition, it shall submit information relating specifically to operation of the reactor, as required in Safety Series No. 35-S2.

231. The final version of the SAR shall be prepared for this stage. The results from the commissioning programme shall be included and assessed to demonstrate that the design intentions have been achieved.

232. From time to time, a review of the safety measures for the operation of the facility should be undertaken. For this review, a comparison of the existing SAR with the operating experience shall be made, including accidents, radiological information, modifications, experiments and other aspects of the operation. If required as a result of the above mentioned review, the operating organization shall submit to the regulatory body a request for an amendment of the licence. This request may include a revised SAR.

Utilization and modification

233. The operating organization shall submit to the regulatory body for review and assessment information on experiments and modifications that may affect reactor safety. The specific submission requirements will depend on the safety significance of the experiments and modifications. These requirements are described in Safety Series No. 35-S2. Specific guidance on the development of appropriate procedures for the control of experiments and modifications is provided in Safety Series No. 35-G2.

234. Experiments and modifications having major safety significance shall be subjected to design, construction, commissioning and safety analysis procedures equivalent to those for the reactor itself. Therefore, the operating organization shall prepare an ad hoc safety analysis report or shall revise the appropriate chapters of the existing SAR for the reactor and submit them to the regulatory body for review and assessment.

235. The safety analysis of an experiment or modification with major safety significance may need to be performed in stages. These stages could be: (1) design and procurement; (2) disassembly; (3) installation or implementation of the modification; (4) reassembly; (5) testing; (6) commissioning; and (7) validation of the design. The safety aspects of each phase of the project must be analysed. The

information submitted to the regulatory body may be arranged according to the phases of the project.

236. The regulatory body shall review and assess the submitted information against the relevant acceptance criteria, agreed upon by the operating organization and the regulatory body during the design stage. It is noted that the SAR provides boundaries of operational limits and conditions which have been demonstrated to be safe, and any experiments and modifications shall fall within these boundaries. Commissioning of the experiment or the modified facility shall take place and shall demonstrate compliance with the design intentions in the SAR. In addition, if changes to the SAR are made, it will be necessary to show that the original analysis is still valid.

Decommissioning

237. Decommissioning of a research reactor facility shall require agreement by the regulatory body. Detailed requirements for this subject can be found in Safety Series No. 35-S2, paras 1901–1905. Documentation[5] shall be required to describe the decommissioning process and to demonstrate that the remaining radiological hazards, if any, at the former site will be minimal, that any radioactive waste generated will be properly dealt with, and that any special hazards associated with the decommissioning process have been adequately analysed and assessed. Further details are provided in other relevant Safety Series.[6]

238. At some point in the decommissioning process (e.g. after removal of all fuel from the site) the SAR ceases to be a major working document. When required by the regulatory body, a decommissioning report shall be prepared.

3. PREPARATION OF THE SAFETY ANALYSIS REPORT

PURPOSE AND SCOPE

301. The SAR shall be prepared by the operating organization for the justification of the design and it shall be the basis for the safe operation of the research reactor.

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[5] This documentation is required when initiating the decommissioning process for the facility. At an earlier stage, different information on decommissioning is required for inclusion in the SAR (see paras A1901–A1903 of the Appendix).

[6] See IAEA Radioactive Waste Safety Standards (RADWASS) programme publications.

The document is an important link between the operating organization and the regulatory body, since it is the main document for the licensing of the reactor.

302. In addition, the preparation of an SAR also serves the following purposes:

(a) To aid the designer in confirming that individual systems are integrated correctly, since the reactor design and the development of the SAR are complementary, interactive processes;
(b) To ensure that the safety analysis has properly identified the safety issues relevant to the design and that safety analysis and design are consistent;
(c) To aid in the appreciation of the relevant design criteria, their limitations and requirements, and in the evaluation of the hazards posed by the facility;
(d) To aid operators in training and familiarization with the facility;
(e) To ensure the establishment of operational limits and conditions on certain parameters which have to be met at all stages in the life of the reactor in order to have an adequate protection of the margins of safety for the reactor.

303. The SAR shall give a detailed description of the reactor site, the reactor itself, the experimental facilities and all other facilities with safety significance. It shall provide a detailed description of the general safety principles and criteria applied to the design for the protection of the reactor, the operating personnel, the general public and the environment. It shall analyse the potential hazards associated with the operation of the reactor. It shall contain safety analyses of accident sequences and of the safety features incorporated in the design to avoid or to minimize accidents, or to mitigate their consequences through design and operating procedures.

304. The SAR shall provide a set of operational limits and conditions to be incorporated into the licence for operation. It shall also provide details of the conduct of operations intended by the operating organization, including its organization and the QA programme established for the design and operation of the facility. It shall also provide details of the emergency plan of the facility.

305. The topics listed in paras 303 and 304 have been deliberately stressed, but all topics treated in the Appendix of this Safety Guide should be adequately covered by the SAR. All of these topics should be prepared in accordance with the corresponding recommendations in the Appendix. However, some of the topics may be discussed in separate documents (e.g. operational limits and conditions, operational procedures, physical security, emergency planning). In this case these topics are treated briefly in the SAR, and a reference is made to the appropriate separate document. These considerations apply particularly to the SARs for existing reactors.

306. Some regulatory bodies require that the SAR be assessed by an independent peer review group. In this case, the results of the review may be reported directly to the regulatory body. Additionally, in some Member States the proposal for a research reactor project may be subjected to an open public debate. For these purposes, the operating organization may have to develop a non-technical version of the SAR which can be understood by the general public.

307. The SAR shall present adequate references which may be necessary for the review and assessment process. This reference material should be freely available to the regulatory body and should not be subject to any classification or limitation that would prevent adequate review and assessment. Such references need not be submitted together with the SAR, but they should be retained by the operating organization or the designers to be provided upon request.

308. Owing to the volume of documents required to support an SAR, a document control system should be established to manage the indexing and to control the issue of the separate documents that make up the SAR. This system shall control the updating, revision, issue or removal of reports in accordance with a QA programme so that the information is always up to date.

309. The reactor type, site and characteristics (design, power, usage) may influence the extent of the information that must be presented in the SAR. Accident scenarios for reactors with higher power levels will usually require more details on the site and on the safety features in order to protect against any significant releases to the environment.

310. For small, low risk facilities (such as critical assemblies or reactors with low power levels) these requirements are much less stringent. However, as the SAR is often the only comprehensive document produced, every topic discussed in the Appendix of this Guide should be considered. Although the extent of information on each topic would be limited, the scope of some topics (e.g. the protection of the operating personnel against overexposure in critical assembly facilities) may be much larger for small, low power facilities.

DEVELOPMENT OF THE SAFETY ANALYSIS

311. The development of the safety analysis and the design of the reactor are complementary processes that should be carried out interactively. The safety analysis is mainly used to enable the operator to understand the basis for the safe operation of

TABLE I. SELECTED POSTULATED INITIATING EVENTS

1. Loss of electrical power supplies
   - Loss of normal electrical power[a]

2. Insertion of excess reactivity
   - Criticality during fuel handling and loading (fuel insertion error)
   - Startup accident
   - Failure of control rod or control rod follower
   - Failure of control rod drive or system
   - Failure of other reactivity control devices (moderator, reflector, etc.)
   - Unbalanced rod positions
   - Failure or collapse of structural components
   - Cold water insertion
   - Moderator changes (e.g. voids, D_2O leakage into H_2O systems)
   - Influence from experiments and experimental facilities (e.g. flooding, voiding, temperature effects, insertion or removal of fissile or absorber materials)
   - Insufficient shutdown reactivity
   - Inadvertent control rod ejection
   - Errors in the maintenance of reactivity devices

3. Loss of flow
   - Failure of primary pump
   - Primary coolant flow reduction (e.g. valve failure, blockage in piping or heat exchanger)
   - Influence of experiment failure or mishandling
   - Failure of emergency cooling system
   - Primary coolant boundary rupture (pipe or vessel) leading to loss of flow
   - Fuel channel blockage
   - Improper power distribution due, for example, to unbalanced rod positions, in-core experiments or fuel loading
   - Coolant reduction due to core bypass
   - Malfunction of reactor power control
   - System pressure deviation from normal limits
   - Loss of heat sink (e.g. valve or pump failure, system rupture)

4. Loss of coolant
   - Primary coolant boundary rupture
   - Damaged pool
   - Pump-down of pool
   - Failure of beam tubes or other penetrations

TABLE I (cont.)

5. Erroneous handling or malfunction of equipment or components
   - Failure of fuel element cladding
   - Mechanical damage to core or fuel (e.g. handling of fuel, dropping of transfer flask on fuel)
   - Criticality in fuel storage
   - Failure of containment or ventilation system
   - Loss of coolant to fuel in transfer or storage
   - Loss or reduction of proper shielding
   - Failure of experimental apparatus or materials (e.g. loop rupture)
   - Exceeding fuel ratings

6. Special internal events
   - Internal fires or explosions
   - Internal flooding
   - Loss of support systems
   - Security incidents
   - Malfunctions of the reactor experiment
   - Improper access to restricted areas

7. External events
   - Earthquake (including seismically induced faulting and slides)
   - Flooding (including upstream dam failure, river blockage)
   - Tornadoes and tornado missiles
   - Hurricanes, storms and lightning
   - Explosions
   - Aircraft crash
   - Fire
   - Toxic spills
   - Transport accidents
   - Effects of adjacent facilities

8. Human error

[a] Although it is not an initiating event, it is recommended that consideration be given to the loss of normal power followed by the loss of emergency power in order to be sure that the consequences are acceptable under emergency conditions.

the reactor and to demonstrate to the regulatory body the way in which the design of the facility and the related operational procedures will contribute to the prevention and mitigation of accidents. The safety analysis shall include analyses of the response of the reactor to a range of postulated initiating events (such as disturbances in process variables, malfunctions and failures of equipment, postulated accidents of low probability, and human error). The safety analysis also contributes significantly to the selection of operational limits and conditions, as well as to design specifications for components and systems in relation to health and safety.

312. In the development of the safety analysis the general requirements established in Safety Series No. 35-S1 (paras 509–517) shall be taken into account. In order to fulfil the objectives of the safety analysis, the specific requirements established in Chapter A.16 (Safety Analysis) of the Appendix of this Guide shall also be taken into account.

313. The safety analysis should identify the design basis accident. In addition, accidents beyond the design basis accident may be analysed for purposes of emergency planning and accident management.

314. In the preparation of the set of postulated initiating events for the analysis, the list given in the Appendix of Safety Series No. 35-S1 shall be considered. This list is reproduced in Table I. Further detailed guidance on the methodology used is given in Annex I of this Guide. This Annex also gives guidance for the analysis of the event sequences triggered by the postulated initiating events and of accidents of low probability (external events and special internal events). In particular, the analyses shall clearly identify a number of assumed input parameters and initial conditions; these shall be presented in the SAR and will determine the basis for the selection of the operational limits and conditions. Annex II of this Guide gives examples of these parameters.

315. Annex I deals mainly with deterministic methods which are normally used for safety evaluations of research reactors. Deterministic techniques are characterized by conservatism and are based on defined sets of rules for event selection, analytical methods, parameter specification and acceptance criteria. Through the use of these methods, reasonable assurance is provided that the ultimate objective — to limit the release of radioactive materials — can be achieved without performing complex calculations, because these methods tend to overestimate the amount of radioactive release. The most severe of these releases (arising from the design basis accident or from a 'maximum credible accident') are taken into account in the selection of siting or in setting design requirements for engineered safety features of the reactor. The choice of these accidents is based on experience and engineering judgement, without the benefit of determining the probabilities of the event sequences.

316. Probabilistic techniques could be used to supplement the above mentioned evaluations. Probabilistic methodologies assume that all accidents are possible and that any number of simultaneous failures may occur, although the probabilities may be very low. Some accidents or accident combinations may have less dramatic consequences than those used in deterministic methodology, but, when they are weighed by their likelihood, they may represent a real risk and may impose different demands on design. In addition, the deterministic approach has difficulties in treating effectively system interdependences (e.g. common cause failure) which the probabilistic methods can address analytically and quantitatively. Application of these techniques also leads to significant improvements in the understanding of system behaviour and interactions, and of the role of operators under accident conditions. These techniques may be indicated for some specific cases, which could be discussed by the operating organization and the regulatory body. Detailed guidance on these topics is given in IAEA-TECDOCs 400 and 517 (list of IAEA publications given at the end of this Guide).

317. The results of the safety analysis of the facility shall be reflected in the SAR, taking into account the requirements established in Chapter A.16 (Safety Analysis) of the Appendix of the present Guide. Chapter A.16 also provides guidance for the comparison of the results with the acceptance criteria to determine the acceptability of the reactor.

4. PERFORMANCE OF THE REVIEW AND ASSESSMENT

PURPOSE AND SCOPE

401. Review and assessment of the information submitted by the operating organization in support of its licence application shall be performed by the regulatory body to determine whether the proposed facility can be sited, constructed, commissioned, operated, utilized, modified and decommissioned without undue radiological risk to the site personnel, the general public and the environment. Within this overall objective, specific objectives of the review and assessment are:

(a) To determine whether the site is adequate for the type, power and usage of the proposed facility;
(b) To determine before construction whether the proposed facility 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 applicant has the ability, reliability, resources, organizational structure and competent personnel to meet the regulatory requirements;
(d) To determine whether the construction remains consistent with the requirements of the regulatory body;
(e) To determine whether the commissioning programme is adequate and whether its results conform to the design intentions;
(f) To determine whether the operational limits and conditions are consistent with the regulatory requirements and whether an adequate level of operational safety can be ensured, including the provisions made for accident conditions;
(g) To determine whether the utilization and modifications of the facility 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

402. The programme for review and assessment should be established jointly by the regulatory body and the operating organization. This programme will take into account the schedule of tasks described in para. 222.

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

404. Before authorizing the construction of the facility, the regulatory body shall review and assess:

(a) The competence and capability of the operating organization to meet the licence requirements;
(b) The site characteristics, to confirm the acceptability of the site and the related data used in the design of the proposed facility;
(c) The basic design of the proposed facility, to confirm that it can meet the safety requirements;
(d) The QA programmes of the operating organization and its vendors;
(e) The design features related to physical protection which are important to safety;
(f) Information necessary for design verification.

405. Before authorizing the loading of nuclear fuel and initial criticality, the regulatory body shall complete the review and assessment of the operating organization's case for the safety of the facility as presented in the SAR, which includes:

(a) The commissioning programme;
(b) The as-built design of the reactor;
(c) The limits and conditions for operation during commissioning;
(d) The provisions for radiological protection;
(e) The adequacy of the operating instructions and procedures, especially the operating and emergency procedures, and of the administrative rules;
(f) The records and reporting systems;
(g) The training and qualification of plant personnel, including the levels of staff and their suitability for the work;
(h) The QA organization and programme for operation;
(i) The emergency plan;
(j) The accounting measures for nuclear and radioactive materials;
(k) The physical protection arrangements important for safety;
(l) The periodic testing, maintenance, inspection, control of modifications, and changes to specifications and surveillance.

406. Before authorizing routine operation of a facility at full power, the regulatory body shall complete the review and assessment of the operating organization's application, which takes into account:

— The results of commissioning tests;
— Any revised limits and conditions for operation; and
— Revised arrangements.

407. Before starting the implementation of proposals for experiments and modifications that are of major safety significance, the regulatory body shall review and assess the information submitted by the operating organization. This review and assessment will generally follow the same procedures as those for the original design (see paras 224–227).

Appendix

CONTENTS OF A SAFETY ANALYSIS REPORT

A.1. INTRODUCTION AND GENERAL DESCRIPTION OF THE FACILITY

A.101. This chapter of the SAR shall 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 facility.

General description of the facility

A.102. In this section, a summary of the principal characteristics of the facility and the site shall be provided. The general arrangement and layout of the facility should be described, starting with the core and continuing with the secondary and tertiary systems to convey an impression of the facility and its components. The features important to operational safety shall be clearly identified. If the facility has novel features or involves unusual approaches to safety analysis, these shall be outlined.

Historical review

A.103. The operational history of the facility shall be presented, including, for an existing facility, the major changes that have been made.

Comparison with other facilities

A.104. Any similarity with other facilities shall be discussed. The design similarities, safety precedents and case histories from other facilities that will be referenced in the SAR shall be itemized.

Identification of owners or agents

A.105. The owner of the facility, the architect/engineer, the prime contractors and the consultants shall be identified, noting whether they have had previous experience with nuclear research facilities.

Safety features

A.106. This section shall briefly state the basic safety principles adopted for the design, construction and operation of the reactor and the nuclear safety criteria for acceptance. The safety features, components or systems incorporated into the facility which will be described in more technical detail in the analysis shall also be identified.

Experimental programme

A.107. This section shall provide a brief description of the experimental programme to be pursued at the facility and the experimental facilities.

List of drawings

A.108. This section shall provide a list of drawings related to the arrangement of the facility and its equipment.

Material incorporated by reference

A.109. This section shall tabulate reference information supporting the SAR. This information may consist of, for example, computer codes and reports from reactor and fuel manufacturers.

Requirements for further technical information

A.110. This section shall identify those safety features or components for which further technical information is required in support of the issue of a licence but which has not been supplied in the SAR.

A.2. SAFETY OBJECTIVES AND ENGINEERING DESIGN REQUIREMENTS

A.201. This chapter of the SAR shall identify, describe and discuss the safety objectives and the engineering design requirements of the structures, components, equipment and systems important to safety.

Safety objectives and general design requirements

A.202. This section shall describe the safety objectives and the general design requirements followed in deciding the design of the reactor facility, considering the requirements for normal operation, anticipated operational occurrences and the accidents taken into account in the design. Safety objectives and design requirements for accident mitigation shall also be included. Other measures which can be used to mitigate accident conditions should be described in the appropriate chapters of the SAR.

A.203. A statement of the overall safety objectives shall be included. This should be followed by a brief description of the underlying safety objectives and general design requirements that are important to the design. Safety objectives are discussed in Section 2 of the main text, and general design requirements are discussed in

Section 5 of Safety Series No. 35-S1. These objectives and requirements may include the following:

(a) Quality assurance;
(b) High standard of engineering design and, in particular, conservative design margins, engineered safety systems (features), barriers to radionuclide transfer and protection of these barriers;
(c) Inherent safety features (those relying on intrinsic physical principles);
(d) Passive safety features (passive features do not actively change state);
(e) The extent to which unique or unusual features that may affect the consequences or the probability of releases are incorporated;
(f) The extent to which redundancy, diversity and independence are applied in the design of engineered safety features;
(g) Fail-safe features;
(h) Defence in depth applied in the design;
(i) Accident prevention;
(j) Accident management;
(k) Proven engineering practice and use of generally accepted standards;
(l) Assessment of human factors and dependent failures; and
(m) Radiation protection.

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

Specific design requirements

A.204. The specific design requirements applied shall be stated in this section. These requirements are discussed in detail in Section 6 of IAEA Safety Series No. 35-S1, and include:

(1) Design QA requirements, including the codes of practice utilized in the design;
(2) Monitoring of variables and control of reactor and system variables within their operating ranges;
(3) Reactor core integrity requirements;
(4) Protection against flow instabilities and suppression of power oscillations;
(5) Criteria for sharing of common structures, systems and components important to safety between facilities at the same site;
(6) Consideration of human factors and ergonomic principles to reduce the potential for human error and to relieve operator stress;
(7) Requirements for design analysis with validated techniques, models or codes;

(8) Reactivity control criteria, including:
    (a) Control redundancy;
    (b) Reactivity limits;
    (c) Shutdown margins; and
    (d) Design provisions to prevent, or to reduce the potential for, fuel loading errors;
(9) Core cooling criteria, including:
    (a) Requirements for adequate core cooling for all operational states and accident conditions; and
    (b) Requirements for coolant system integrity and protection of the boundary from leakage;
(10) Fuel design limits and material design criteria, including:
    (a) Fuel design bases for mechanical, chemical and thermal design;
    (b) Safety margins for fuel design parameters;
    (c) Methods of achieving a conservative safety margin for prototypical fuels;
    (d) Verification of fuel integrity; and
    (e) Design bases for mechanical, thermal and chemical design of reactor materials important to safety;
(11) Design criteria for reactor utilization, including:
    (a) Radiation protection for all operational conditions;
    (b) Maintenance of fuel design margins;
    (c) Design requirements to ensure that safety system settings are not adversely affected; and
    (d) Recognition of the interdependence between the reactor and any installed experimental equipment;
(12) Design criteria for the safety system and, where required:
    (a) Provision of systems for shutdown, fuel cooling and control of radionuclide releases;
    (b) Operating requirements;
    (c) Separation requirements for safety system and control functions; and
    (d) Fail-safe mode requirements;
(13) Reliability requirements, including:
    (a) Operational (process) system reliability;
    (b) Reliability targets for safety systems;
    (c) Requirements for safety system redundancy and unavailability;
    (d) Segregation for independence or diversity; and
    (e) Requirements for safety support systems;
(14) Design bases for equipment qualification for natural events, environmental conditions, fire protection and external hazards;
(15) Methods employed for protection against dependent failure;
(16) Capability for surveillance and maintenance of safety related equipment; and

(17) Radiation protection measures in the design, including:
   (a) Reducing exposure by design features;
   (b) Control of releases;
   (c) Control of radioactive materials;
   (d) Prevention of inadvertent criticality; and
   (e) Monitoring of fuel and waste storage areas.

Classification of structures, components and systems

A.205. If any scheme has been devised for the classification of structures, components and systems for purposes of analysis or design, such as for seismic safety, QA or nuclear safety, the basis for the classifications and the list of classes should be presented in this section of the SAR.

External events

A.206. In this section the design criteria for the resistance of structures, systems and components to external events should be presented. These may include:

(a) Wind and tornado loadings;
(b) Water level (flood);
(c) Protection against missiles from internal and external sources, including aircraft;
(d) Seismic hazard and seismic analysis; and
(e) Fire and explosions.

Additional information on siting requirements is presented in Section 4 of Safety Series No. 35-S1.

Codes and standards

A.207. In this section, all codes and standards to be employed in the design of structures, systems and components shall be listed. Justification for their use shall be provided, particularly if they are relevant for nuclear safety.

A.208. If different codes and standards are used for different aspects of the same item or system, the consistency between them shall be demonstrated. Typical areas covered by codes and standards are:

— Mechanical design, including stress analysis and fracture mechanics;
— Structural design;
— Earthquake resistant design;
— Selection of materials;
— Fabrication of equipment and components;
— Inspection of fabricated and installed systems, components and structures;

— Thermohydraulic and neutronic design;
— Electrical design;
— Design of instrumentation and control systems;
— Shielding and radiological protection;
— Fire protection;
— Inspection, testing and maintenance as related to design;
— Design and production of fuel.

A.209. For items important to safety for which no appropriate established codes or standards exist, an approach derived from existing codes or standards for similar equipment should be applied. In the absence of such codes and standards, the results of experience, tests, analysis, or a combination thereof, may be applied, and an explanation of these results should be given.

Technical design methods

A.210. This section should describe methods for design and analysis of systems and components, including design transients, computer programs used, experimental stress analysis, and any programmes for dynamic testing and analysis of the mechanical systems and components. Particular attention should be paid to items important to safety.

Design for internal fire protection

A.211. This section should describe the design requirements for fire protection inside the facility. It should include passive features, such as isolation, separation, selection of materials, the building layout and zoning, the location of fire barriers, and the safety system layout and protection (including separation of safety related redundant systems). The fire protection system is described in para. A.1008.

Qualification of components

A.212. This section should describe the design bases for qualification of components to resist such environmental factors as vibration, thermal expansion, radiation, corrosion, dynamic effects, mechanical loadings and high pressure, high temperature, humidity, water, steam, chemicals, low temperature or vacuum. Qualification tests and analyses that have (or will be) performed should be described.

Conclusions

A.213. This section shall provide the conclusion that the facility 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 facility, including qualification of components.

A.3. SITE CHARACTERISTICS

A.301. This chapter of the SAR shall provide information on the geological, seismological, hydrological and meteorological characteristics of the site and the vicinity, in conjunction with the present and projected population distribution, land use, site activities and planning controls. The purpose is to indicate how these site characteristics have influenced the facility design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint. Additional information on siting can be found in Section 4 of Safety Series No. 35-S1.

A.302. Information should be provided in sufficient detail to support the analysis and conclusions of Chapter A.16 (Safety Analysis) to demonstrate that the reactor facility can be safely operated at the proposed site. For many low power reactors, which present very limited hazards, the amount of detail provided in this chapter can be substantially reduced.

A.303. 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.304. The location of the facility site shall be specified and an area map should be provided which indicates:

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

A.305. This section shall describe the applicant's legal rights with respect to all areas that lie within the designated controlled area, as well as any activities unrelated to the facility's operation that will be permitted in the controlled area.

External effects

A.306. This section shall describe the site related phenomena and characteristics, of both natural and man induced origin, which must be taken into account to assess the suitability of the site for the reactor facility.

A.307. This section shall describe the appropriate methods adopted for establishing the external events which will constitute the design basis events for important natural phenomena and man induced effects; further information on design criteria for protection against these effects should be given in Chapter 2 (para. A.206).

Geology and seismology

A.308. The geology of the site and its environs shall be described in this section in sufficient detail to identify effects that could present a hazard to the facility.

A.309. Information that is used to establish the seismic design basis, such as earthquake return frequency and ground motion, shall be presented in this section, as well as information for:

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

Meteorology

A.310. This section shall provide a meteorological description of the site and its surroundings, including wind speed and direction, air temperature, precipitation, humidity, atmospheric stability parameters and prolonged inversions. Seasonal and annual frequencies of weather phenomena should be provided, including, where applicable, hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, freezing rain, snow, ice and sandstorms.

Hydrology and oceanography

A.311. The surface and underground hydrology of the site and its environs shall be described in this section, including the location, size, flow, water use and other characteristics of nearby freshwater courses. The location and characteristics of man made structures should be indicated, including darns and diversion channels, and any flood control measures.

A.312. A description of the groundwater hydrology in the vicinity of the facility should be presented, including the main characteristics of the water bearing formations and their interaction with surface waters, and data on the use of groundwater in the region.

A.313. If the reactor is to be built near the coast, oceanographic and hydrographic information, including a bathymetric map of the near-shore area in front of the location of the reactor, shall be given.

A.314. Natural phenomena to be considered in the SAR may include, where appropriate:

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

Nearby industrial, transport and military facilities

A.315. All present or projected industrial, transport or military facilities which could pose a hazard to the reactor facility shall be described in this section, for example: significant manufacturing or chemical plants, refineries, storage facilities, mining and quarrying operations, military bases or sites, transportation routes (air, land, water), transport facilities (rail lines, docks, anchorages, airports), oil and gas pipelines, drilling operations and wells, and underground storage facilities. The potential adverse effects that these facilities can have on the reactor building, e.g. aircraft crashes or other transport accidents, shall be described.

A.316. Foreseeable significant changes in land use should be considered, including expansion of existing facilities or activities, or construction of high risk facilities.

Radiological impact

A.317. This section shall describe ecological aspects and, in particular, the biological aspects of transfers of radioactive material and their impact on man. Most of this detail may not be required for low hazard, low power reactors. In this case, only a brief summary should be given under each heading. If a radiological impact section is not provided, justification for deleting this SAR section shall be provided.

A.318. Information should be included which, in combination with details of radioactive discharges and of the radionuclide behaviour/transfers presented in other chapters, will permit an assessment of the doses to individuals and the population, and of contamination of biological chains and food chains. This information should cover the entire region likely to be affected, taking into account topographical, hydrological and meteorological characteristics.

Population distribution

A.319. The population distribution around the facility and in the region, including seasonal and daily variations, shall be presented in this section. In particular, information on existing or projected population distributions around the facility should be collected and kept up to date during the lifetime of the facility.

Natural environment, land and water usage

A.320. The characteristics of the regional ecology and the uses of land and water shall be summarized in this section, including:

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

Baseline radiological levels

A.321. This section should include a description of the radioactivity due to both natural and artificial substances in air, water and ground (including underground), and in flora and fauna. If there was a nuclear installation on the site, a brief description of any incidents that have led to residual activity at the site shall be provided.

Atmospheric dispersion of radioactive materials

A.322. This section shall describe the models used to assess the atmospheric dispersion of radioactive material released under operational states and accident conditions of the reactor, in accordance with the policies of the operating organization and the regulatory body. It should be stated whether the dispersion estimates are based on representative meteorological data or on conservative, worst weather assumptions. The scope of the models should include any unusual site and regional topographic features, and facility characteristics that may affect atmospheric dispersion. The accuracy and validity of the models, including the suitability of input parameters, source configuration and topography, should be discussed.

A.323. This section should provide the results of calculations of atmospheric diffusion parameters at the site boundary and at off-site locations, or refer to radionuclide atmospheric concentrations and dose calculations presented in Chapter A.12 (Operational Radiological Safety) and Chapter A.16 (Safety Analysis).

Dispersion of radioactive materials through surface waters and groundwater

A.324. This section shall indicate locations near the facility where radionuclide releases could be discharged and where they could enter surface waters or groundwater. The results of hydrological and hydrogeological investigations carried out to assess, to the extent necessary, the dilution and dispersion characteristics of bodies of water should be presented.

A.325. The models used to evaluate the possible impact of contamination of surface waters and groundwater on the population shall be described. Where appropriate, the results of off-site dose calculations should be provided or reference to such calculations should be made in Chapter A.12 (Operational Radiological Safety) and Chapter A.16 (Safety Analysis).

Mitigation

A.326. This section should discuss the results of investigations carried out to assess the need for, or the extent of, mitigation measures such as accident management or emergency measures which may be required in the event of accidents at the facility, in accordance with the policies of the regulatory body. Reference should be made to Chapter A.16 (Safety Analysis) and Chapter A.20 (Emergency Planning and Preparedness) where appropriate, to support the evaluation.

A.327. This section should consider:

- Population distributions and projected population changes in the region surrounding the facility;
- Present and projected land and water use in the region;
- Potential radioactive source terms, and doses to the population from direct radiation fields and from airborne/aqueous pathways;
- Potential contamination of the food chain;
- Potential doses to site personnel;
- The need to control activities unrelated to facility operation in the controlled area or to evacuate persons engaged in these activities;
- The capability of the appropriate authorities to implement emergency measures if required; and
- The feasibility of emergency plans (if required), taking into account the population distribution, national and international boundaries, special groups (e.g. hospitals), special geographical features (e.g. islands), and communication and transport facilities.

Conclusion

A.328. This section shall provide the conclusion regarding the acceptability of the site for the research reactor under consideration. If further analysis is required to support the conclusion of acceptability, site characteristics shall be identified, and reference to other appropriate sections of the SAR should be made. It should be stated that the radiological risk to the population from accident conditions, including those which may require implementation of mitigation measures, is acceptably low and in accordance with national requirements.

A.4. BUILDINGS AND STRUCTURES

Reactor building

A.401. This section shall contain a description of the reactor building and internal structures (e.g. reactor pools and internals, supporting structures, cranes, ventilation system), emphasizing those characteristics of the building which assist in maintaining acceptable radiation levels on and off the site during all operational states. Information on the requirements for the reactor building is presented in Section 6 of Safety Series No. 35-S1.

A.402. The description shall include the design basis of the reactor building and internal structures, together with the design basis of the building penetrations (air locks, doors, etc.) regarding their resistance to internal and external events (see paras A.211 and A.307).

A.403. The design and operation of the ventilation systems, including requirements for containment or confinement, shall be described. If applicable, distinction shall be made between the system used during normal operation and the system used for emergencies. The specific efficiencies of the air filters and iodine traps shall be given.

A.404. The design and operation of reactor building subsystems, such as the system for controlling the release of fission products, shall be described.

A.405. The design and operation of cranes or other lifting devices shall be described.

A.406. The descriptions required in paras A.401–A.405 shall be supported by drawings, including flow and instrumentation diagrams.

A.407. Permissible limits and testing and inspection requirements for the subsystems shall be described, in particular those for ensuring the prescribed leaktightness/leak rates.

Auxiliary structures

A.408. This section shall include a description of auxiliary buildings and structures important to reactor safety.

A.5. REACTOR

A.501. This chapter of the SAR shall provide all the necessary information to demonstrate that the reactor is capable of fulfilling its safety functions. These functions are:

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

A.502. This section shall provide information pertaining to operational states, including the portions of the safety analysis dealing with them. The consequences of failures and accidents are treated in Chapter A.16 (Safety Analysis).

Summary description

A.503. The chapter shall start with a summary of the functional, technical and operational characteristics of the reactor. Drawings, flow sheets and tables should be provided for illustration and support. Annex III presents items which should be considered in the description. The description shall indicate the dependent and interrelated safety functions of the main reactor components.

Fuel elements

A.504. Basic information on fuel design and fuel properties shall comprise:

(a) Fuel material, enrichment, composition and metallurgical state (oxide, alloy, etc.);
(b) Material (type, composition, etc.) of all other fuel parts, such as cladding, spacers, fittings and burnable poisons;
(c) Fuel geometry, dimensions, tolerances, etc. (together with drawings);
(d) The material properties required for the analyses mentioned in paras 505–508;
(e) The maximum temperatures to which the fuel elements can be subjected without deformation (due to blister formation or mechanical weakening);
(f) Fuel element instrumentation, if any.

A.505. An analysis shall be provided which shows that the fuel elements can withstand the thermal conditions to which they are subjected throughout their normal operational life cycle. This life cycle should comprise not only nuclear applications in the reactor core but also periods of storage, handling and transport.

A.506. An analysis shall be provided which shows that the fuel elements can withstand the mechanical forces to which they are subjected (hydraulic forces, differential thermal expansion effects, etc.), without breach of mechanical integrity or undue deformation. The anticipated effects should be quantified.

A.507. An analysis shall be provided which shows that the fuel element cladding can withstand the chemical environment to which it is subjected during use and storage, taking into account the effects of temperature and irradiation.

A.508. An analysis shall be provided which shows that the intended irradiation conditions and limits (fissioning density, total fissions at the end of life, etc.) are acceptable and will not lead to undue deformation or swelling of components which may contain fissile materials. The anticipated upper limit of the eventual deformation (expressed as minimum cooling channel width) shall be provided for the thermal safety analysis.

A.509. These analyses and information should be supported by a report on experimental measurements and irradiation experience, and shall include the entire fuel cycle (storage, transport, etc.).

Reactivity control system

A.510. Information shall be provided which demonstrates that the reactivity control mechanisms and their drive system can fulfil their designated safety functions during all foreseeable operating conditions. Only the technical safety functions (such as insertion capability) shall be addressed; the reactivity aspects shall be treated in the section on nuclear design, paras A.513 and A.514; the incorporation of the protection and power regulating systems into the instrumentation is treated in Chapter A.8 (Instrumentation and Control).

A.511. Basic information shall be provided on the design of reactivity control mechanisms and drive systems, including the materials, redundancy and diversity aspects, the anticipated performance characteristics (such as drive speed and insertion time), fail-safe features, etc.

A.512. An analysis shall be provided which shows that the reactivity control system will function properly under all operational states of the reactor and that it will maintain its reactor shutdown capability under all foreseeable accident conditions, including failures of the control system itself.

Nuclear design

A.513. An analysis shall be provided which shows that the nuclear conditions in the reactor core are acceptable throughout the anticipated core cycle. The analysis shall include the steady state and the dynamic nuclear and thermal characteristics of the reactor.

A.514. Basic information on the nuclear design shall include:

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

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

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

A.515. The basic information must be supported by reference to the calculation methods and codes, experimental verification of the basic input data, or other information that can support the validity of the nuclear properties, details of which are supplied in this section.

A.516. An analysis shall be provided which shows that the effectiveness, speed of action and shutdown margin of the reactor shutdown system are acceptable and that a single failure in the shutdown system will not prevent the system from completing its safety functions when required. A sufficient shutdown margin shall be provided so that the reactor can be made and maintained subcritical under all operational states and accident conditions.

Thermal and hydraulic design

A.517. Information shall be provided to prove that, during operational states, adequate core cooling capacity will be available to keep the reactor fuel in a thermally safe condition and that an adequate thermal safety margin will be available to prevent or minimize fuel damage in accident conditions.

A.518. Basic information on thermal and hydraulic core design shall include:

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

(b) The power distribution in all core components which may contain fissile materials, as derived from the nuclear design characteristics provided in para. A.514(b).

A.519. The information shall be qualified by reference to analyses, experimental measurements, fabrication specifications, etc., from which it is derived, thus providing a quantitative assessment of the uncertainties in each of the safety relevant parameters that have been quantified.

A.520. An analysis shall be provided which proves that the maximum thermal load to which any fuel element in the reactor is subjected during any operational state does not exceed the available cooling capacity, whether by forced or natural convection. The limiting criteria that are to be applied for this analysis may be related to nucleate boiling, flow instability, inlet vortexing, departure from nucleate boiling, etc. (depending on the reactor type and operating conditions), and should be verified and qualified with regard to each other.

A.521. The analysis shall lead to determination of a thermal safety margin for the core, both for 'best estimate' conditions (based upon nominal thermohydraulic conditions) and for 'conservative' conditions (taking into account the uncertainty values mentioned in para. A.519).

A.522. The assessment shall take into account changes to safety relevant fuel parameters that may be caused by mechanical deformation, irradiation swelling, etc., as mentioned in paras A.506 and A.508.

Reactor materials

A.523. Information shall be provided which shows that all materials which have been selected for the construction of safety relevant components and structures can withstand the nuclear and chemical environment to which they are subjected, without unacceptable worsening of the performance of the safety functions of such components and structures. Ageing effects due to the deterioration of properties, as well as irradiation damage shall be included.

A.524. Items which should be considered include:

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

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

A.525. The information shall 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 the essential material properties shall be described.

A.6. REACTOR COOLANT SYSTEMS AND CONNECTED SYSTEMS

A.601. This chapter of the SAR shall provide a description of the reactor coolant systems which transfer the heat from the reactor to the ultimate heat sink. The description shall contain the main design and performance characteristics. It shall be supported by schematic flow diagrams and an elevation drawing of the coolant systems.

Primary cooling system

A.602. The design and operation of the primary cooling system shall be described in detail. The design and performance characteristics of the main components (pumps, valves, heat exchangers, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials of which the components are made and the effects of irradiation on these materials shall be specified. The primary vessel, together with factors resulting from service conditions, such as corrosion, fatigue and thermal stress cycling, shall be described.

A.603. Methods utilized for leak detection and measures for minimizing the loss of the primary coolant shall be described. The potential consequences of the loss of the primary coolant shall be discussed.

A.604. The chemistry data for the primary coolant shall be presented, including the effects of irradiation of the primary coolant.

Secondary cooling system (ultimate heat sink)

A.605. The design and operation of the secondary cooling system shall be described in detail. The design and performance characteristics of the main components (pumps, valves, heat exchangers, cooling towers, piping) shall be tabulated. A flow and instrumentation diagram shall be included, as well as drawings of the main components. The materials of which the components are made and corrosion control measures shall be specified.

A.606. If the reactor uses a closed intermediate cooling system between the primary and the secondary cooling systems, this shall also be described.

Moderator system

A.607. The design and operation of the moderator system shall be described in detail. The calculation of the heat generated in the moderator shall 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 of which the components are made should be specified; the effects of irradiation and corrosion should be discussed. Material ageing should also be discussed.

Emergency core cooling system

A.608. The design and operation of the emergency core cooling system shall be described in detail. The accidents with which this system must cope should be mentioned, and analyses should be provided which show that the system fulfils the requirements. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials of which the components are made should be specified, and the effect of irradiation, if any, should be discussed, as well as any environmental effects. The procedures of inspection and testing of the emergency core cooling system shall be described.

Decay heat removal system

A.609. The design and operation of the decay heat removal system, including the ultimate heat sink, shall be described in detail. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The

materials of which the components are made should be specified; the effects of irradiation, if any, and of corrosion should be discussed, as well as unfavourable environmental conditions for the ultimate heat sink.

Primary purification system

A.610. The design and the operation of the primary purification system shall be described in detail, including the procedures for resin exchange and the shielding used to protect persons during this operation. This may be described in this section, or reference may be made to Chapter A.10 (Auxiliary Systems).

A.611. The design and performance characteristics of the main components (pumps, valves, filters, resins, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials of which the components are made should be specified. The means for monitoring the performance and for renewing the system's ability to purify the coolant should be described.

Primary coolant make-up system

A.612. The design and operation of the coolant make-up system may be described here, or reference may be made to Chapter A.10 (Auxiliary Systems). The relevant chemistry data of the coolant should be presented, including details of new water treatment and degassing and demineralizing processes. Means for controlling the make-up water chemistry and the way in which the design ensures that the primary coolant does not enter the potable water system should be discussed.

A.7. ENGINEERED SAFETY FEATURES

A.701. This chapter of the SAR shall identify and provide a summary of the types, locations and functions of the engineered safety features (ESFs) provided in the facility. Examples of ESFs are the emergency core cooling system and the confinement/containment system. The requirements of these systems and supplementary features are discussed in paras 635–645 of Safety Series No. 35-S1.

A.702. The design basis and various modes of operation of the ESFs should be discussed in detail. The accidents with which these systems must cope should be presented and analyses should be provided which demonstrate that the systems fulfil the requirements. The subsystems which are essential for the proper operation of the ESFs shall be described (e.g. uninterruptable power supply for the emergency core cooling system). The extent to which the ESFs are automated and the conditions for which manual override is warranted should be clearly indicated.

A.703. Information shall be provided on:

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

A.704. Reference shall be made to the relevant chapters in the SAR or to other documents where the ESFs are further described.

A.8. INSTRUMENTATION AND CONTROL

A.801. This chapter of the SAR shall provide information regarding the instrumentation and control (I&C) systems of all safety systems and of safety related items and systems. The information provided shall emphasize those instruments and associated equipment which affect reactor safety. The requirements for I&C are discussed in paras 646–651 of Safety Series No. 35-S1.

A.802. All I&C and supporting systems (with emphasis on safety systems and safety related systems), including alarm, communication and display instrumentation, shall be listed, and considerations of instrumentation errors shall be included. Adequate schematic diagrams shall also be provided.

A.803. Information on provisions for testing the I&C system shall also be included.

Reactor protection system

A.804. The requirements for the reactor protection system are discussed in paras 626–634 of Safety Series No.35-S1. The reactor protection system, including all its components, shall be described in detail. A schematic diagram shall 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.805. The adequacy of the protection system to shut down the reactor in a safe manner (e.g. by providing redundancy) and to bring the reactor into a safe condition shall be described. A reliability analysis of the protection system should also be presented.

A.806. For computer based digital protection systems, software verification and validation shall be included.

A.807. The means for detecting failures within the reactor protection system shall be described.

A.808. This section shall describe the methods used to prevent adverse environmental conditions (temperature, humidity, high voltage, electromagnetic fields, etc.) from influencing the reactor protection system, and methods to protect against tampering.

Reactor power regulating system

A.809. All elements of the reactor power regulating system shall be described (design criteria and reliability analysis). All interfaces between the power regulating system and the reactor protection system should be identified and analysed to confirm that they do not lead to degradation of safety.

Alarm system

A.810. The alarm system which indicates abnormal facility status and failures within the safety systems shall be described.

Interlocks

A.811. All interlocks which are provided for the safety of the reactor and the relevant logic shall be listed and described.

Other instrumentation systems required for safety

A.812. All other instrumentation systems required for safety (e.g. fire protection instrumentation) should be described.

Control room

A.813. This section should include a description of the instrumentation systems which are provided in the reactor control room for indicating the status of the protection system, the reactor power regulation system and other important systems.

A.814. It should be discussed whether sufficient information and means are available in the reactor control room to enable the operator to carry out the required safety actions.

A.815. The control actions required in emergencies, including actions to be performed in the emergency control room, if provided, should be discussed.

A.9. ELECTRIC POWER

A.901. This chapter of the SAR shall describe the AC and DC power supplies, with emphasis on their dependability and on their relationship to safety. The descriptions should be supported by adequate diagrams. The adequacy of each class of power supply shall be demonstrated.

Normal AC power supply

A.902. This section shall describe the normal AC power supply, emphasizing the design and performance characteristics.

Emergency AC power supply

A.903. This section shall describe the design and operation of the emergency power supply, emphasizing the connection to the normal power supply.

A.904. The description shall include:

(a) The reliability of the system;
(b) The starting load requirements of the equipment powered by the system;
(c) The starting time of the system and the time sequence for connecting loads;
(d) The starting method (automatic/manual).

Uninterruptible power supplies

A.905. The design and operation of the AC and DC uninterruptible power supplies, including the connection to the emergency power supplies, shall be described. The capacities of the power source shall be specified and compared with the requirements of the safety related loads.

Cable and routing

A.906. Information shall be provided on the types of cable used. The adequacy of the measures employed to separate the cables in order to maintain redundancies, prevent cross-talk and provide fire protection shall be demonstrated.

A.10. AUXILIARY SYSTEMS

A.1001. This chapter shall provide information concerning the auxiliary systems included in the facility. A description of each system, the design bases for the system and for critical components, a safety evaluation demonstrating how the system satisfies the requirements of the design basis, the testing and inspection to be performed to verify the system capability and reliability, and the required instrumentation and

control should be provided. In cases where auxiliary systems are not related to the protection of the public against exposure to radiation, enough information should be provided to allow understanding of the design and function of the auxiliary system; emphasis should be placed on those aspects that might affect the reactor and its safety features or that might contribute to the control of radioactivity inside the facility.

Fuel storage and handling

A.1002. This section shall describe systems for storing fresh and spent fuel, for cooling and cleaning the spent fuel pool (if applicable), and for handling and, if necessary, cooling the fuel as it moves within the facility. The quantity of fuel to be stored and the means for maintaining a subcritical array, even during adverse seismic conditions, shall be provided.

A.1003. Fresh fuel handling and storage, including the tools and systems used, shall be described. A brief description of the operating procedures for fuel handling should also be given (see para. A.1310).

A.1004. Information concerning the management of irradiated fuel should be provided, i.e. the activity, decay rate, refuelling frequency and storage requirements.

Water systems

A.1005. All water systems of the facility that have not been described previously shall be discussed in this section. These may include the primary purification system, the service water system, the cooling system for reactor auxiliaries and the primary coolant make-up system. In each case, the information provided should include the design bases, a system description, flow and instrumentation diagrams, a safety evaluation, if required, testing and inspection requirements, and instrumentation requirements.

Process auxiliaries

A.1006. All auxiliary systems associated with the reactor process system and the experimental facilities, such as compressed air, process sampling and equipment and floor drainage systems, shall be discussed in this section. The discussion should include the design bases, a system description, a safety evaluation, testing and inspection requirements, and instrumentation requirements.

Air conditioning, heating, cooling and ventilation systems

A.1007. The ventilation systems for all areas except the reactor building (see Chapter 4) shall be discussed. A system description should also be provided.

Fire protection

A.1008. A description and a safety analysis of the fire protection system shall be provided in this section, including information on procedures and maintenance activities. Reference could also be made to the design methods (para. A.211).

Other auxiliary systems

A.1009. In this section, the design bases, system descriptions and a safety analysis shall be provided for the other auxiliary systems important to safety.

A.11. REACTOR UTILIZATION

A.1101. This chapter shall provide information demonstrating that reasonable provisions have been made that the experimental facilities and experiments do not pose a significant risk to the facility, the staff and the general public. Additional guidance may be found in Safety Series No. 35-S1 and S2, and in the Safety Guide on Safety in the Utilization and Modification of Research Reactors, Safety Series No. 35-G2.

Experimental facilities

A.1102. This section shall provide a description of the design basis and of the design, as well as a safety analysis for all experimental facilities associated directly or indirectly with the reactor. Such facilities may include the beam tubes, the thermal column, in-core or moderator facilities, boreholes, experimental loops, etc.

A.1103. 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 shall be given to the methods that will be utilized to review and approve new experimental facilities which are outside the scope of the facilities discussed in the SAR.

Experimental programme

A.1104. This section shall describe the expected experimental use of the reactor, including the operational limits and conditions for the experiments.

A.1105. Materials that will not be allowed to be used in experiments in or near the reactor core shall be specified, together with materials that may only be utilized under additional safety conditions.

A.12. OPERATIONAL RADIOLOGICAL SAFETY

A.1201. This chapter shall describe, for normal operational conditions:

(a) The radiation protection programme, including the radiation protection policies of the operating organization;
(b) Sources of radiation at the facility;
(c) Facility design for radiological safety;
(d) Waste management systems;[7]
(e) Dose assessment for normal operation;
(f) Conclusions.

A.1202. The estimated radiation exposure of the staff and the general public in accident conditions shall be analysed in Chapter A.16 (Safety Analysis). Exposure from anticipated operational occurrences shall be within the bounds laid down in the accident analysis and, therefore, shall also be described in Chapter A.16. Radiological emergency planning is described in Chapter A.20 (Emergency Planning and Preparedness), and irradiated fuel management should be treated in Chapter A.10 (Auxiliary Systems).

Radiation protection programme

Radiation protection policy of the operating organization

A.1203. This policy statement shall endorse the radiation protection objective as stated in paras 202 and 203 of Safety Series Nos 35-S1 and 35-S2, which refer to the IAEA Basic Safety Standards for Radiation Protection and the recommendations of the International Commission on Radiological Protection. In particular, this section shall summarize the legal dose limits for both occupationally exposed personnel and the general public, as well as the operational emission limits based on these dose limits. The regulatory requirements for maintaining exposures and releases of radioactive waste and effluents below these legal dose limits shall be described, as well as the reference level of the doses and releases established by the operating organization to assist the reactor management in ensuring that the radiation doses and operational emissions are as low as reasonably achievable (ALARA) and below the above mentioned limits.

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

A.1204. The programme for radiation protection established and implemented by the operating organization of the facility, including the ALARA programme, shall be described, as well as the emission control philosophy at the facility, including the organizational policy concerning the control and monitoring of releases and the evaluation of trends.

Organization, staffing and responsibilities

A.1205. This section shall describe the administrative organization of the radiation protection management and staff, including the authority and responsibility associated with each position identified and the experience and qualifications of the personnel responsible for the health physics programme. As appropriate, the functional responsibilities of the health physics group in areas such as radiation protection advice, support, training, monitoring, dosimetry and laboratory services, and administrative control of radioactive material shall be included. Reference should also be made to the relevant QA programme applicable to the radiation protection activities.

Facilities, equipment and instrumentation

A.1206. The health physics facilities and equipment, such as laboratories for radioactive analysis, contamination control equipment and decontamination facilities, shall be described, including the location of these facilities, as well as the arrangements for maintenance and calibration of health physics instruments and for personnel monitoring (e.g. film badge and thermoluminescent dosimetry service).

A.1207. This section shall describe the radiation and contamination monitoring stations, including fixed hand and foot monitors, portal monitors (if used) and portable activity monitors located at these stations. The portable and laboratory equipment and instrumentation for performing radiation and contamination surveys, for contamination control between different access zones, for airborne radioactivity monitoring/sampling and for personnel monitoring shall also be described.

A.1208. Information shall be provided on the protective clothing and equipment routinely used at the facility, including respiratory protective equipment.

A.1209. Special equipment available for use in an emergency when high dose rates may prevail, and any special training of facility personnel in the use of this special equipment are described in the Emergency Plan (see para. A.2003).

A.1210. If separate documentation has been prepared to describe the health physics programme, this documentation may be referred to, and only a brief summary may be given in this section.

Procedures and training

A.1211. An overview of the written procedures for the radiological protection programme shall be provided. Such procedures shall be prepared in accordance with the relevant QA programmes and may include:

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

A.1212. Reference shall be made to the operating procedures which include provisions for controlling the doses to operating personnel for normal operation and maintenance, in-service inspection and refuelling. Reference shall also be made to the operating procedures which include provisions for monitoring of systems that collect, contain, store or transport radioactive liquids, gases and solids. Any procedures relating to experimental facilities, isotope production and laboratory activities shall be referenced.

A.1213. This section shall describe the methods and procedures for controlling and evaluating exposures of experimenters and other personnel (e.g. contractors and students) likely to have only a cursory knowledge of radiation protection procedures at the facility.

A.1214. Reference should also be made to the emergency operating procedures described in Chapter A.20 (Emergency Planning and Preparedness) for emergency situations at the facility when dose rates may be high.

A.1215. This section shall give a brief description of the facility radiation protection training programme for the radiation protection management and staff, and for other facility personnel, including contractors and students.

Effluent monitoring programme

A.1216. This section shall describe the effluent monitoring programme carried out on the site and off the site. If off-site monitoring of effluents is done by the operator of the facility, the arrangements and responsibilities shall be discussed.

Audit and review programmes

A.1217. This section shall describe the provisions for controlling the conduct of the radiation protection programme and its review.

Radiation sources at the facility

A.1218. All normal potential radiation sources (contained and airborne) due to reactor operation and all potential radiation sources throughout the facility that can be identified shall be catalogued in this section. These sources are used as bases for shielding calculations, design of ventilation systems, dose assessment, waste management and determination of effluent releases.

A.1219. For typical sources that are shielded or contained, information shall be provided on their form, location, geometry, isotopic content and activity. For typical liquid and airborne sources, information shall be provided on their form, location, isotopic content and concentrations.

A.1220. Examples of sources of radiation or radiation fields can be found in Annex IV.

A.1221. This section shall provide drawings of the facility, showing the location of all typical sources.

Facility design for radiological safety

A.1222. In the description of the design considerations for the facility and equipment it shall be demonstrated that external and internal radiation exposures of facility personnel and the general public are based on the radiation protection policy described in para. A.1203. It shall be described how the design philosophy reduces the exposure of personnel, minimizes the undesirable production of radioactive material, reduces the need and the time spent for maintenance and operational activities with the possibility of internal or external exposure, and keeps the releases of radioactive material to the environment as low as reasonably achievable.

Access control and zoning

A.1223. This section shall describe how the layout of the reactor facility provides for the necessary segregation of radioactive material from facility personnel and the general public and how it prevents other hazards. This layout may include zones which are classified according to their potential for radioactive contamination and/or radiation exposure. Drawings shall be provided showing the facility layout and zoning and the controlled access areas. The section shall also describe the access control measures which guard against approach by personnel to areas of high radiation fields and potentially contaminated areas, or which prevent the placement of a radiation source (e.g. spent fuel or activated/irradiated material) in an area where personnel is present.

Shielding and protective features

A.1224. The shielding required for the reactor, the associated facilities (e.g. beam tubes) and the radiation sources identified in paras A.1218–A.1221 shall be described, including the radiation levels external to the shielding at locations where occupancy may be required, as well as the materials used, the criteria for penetrations of the shielding and the calculational methods. The section shall also describe other protective features, such as geometric arrangements (e.g. distance) or remote handling to ensure that the exposure of reactor personnel and of the general public are within the specified requirements and based on the ALARA principle, as well as the methods ensuring that beam tubes and other experimental facilities are adequately shielded against radiation streaming during experimental use.

Ventilation for radiological protection

A.1225. This section shall discuss the radiological protection aspects of the ventilation system on the basis of the description of the system in Chapter A.4 (Buildings and Structures) or Chapter A.7 (Engineered Safety Features).

Radiation monitors

A.1226. This section shall describe the permanent monitoring systems for area radiation, effluents and airborne radiation, including information on:

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

A.1227. This section shall describe the criteria and methods for ensuring that representative samples are obtained from the areas being monitored.

A.1228. The radiation monitoring system or other systems that could be used during accident conditions should be described. Reference should be made to Chapter A.16 (Safety Analysis) for use of the system in the safety analysis, and to Chapter A.20 (Emergency Planning and Preparedness) for emergency measures regarding the application of monitoring under accident conditions.

Waste management systems[8]

Solid waste

A.1229. This section shall describe the treatment of solid waste including, as applicable:

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

Liquid waste

A.1230. This section shall describe the treatment of liquid sources that are considered to be waste, including:

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

A.1231. The criteria for determining whether processed liquid waste will be recycled or discharged shall be described, including the expected effluent concentrations by radionuclide release and the total annual release to the environment. The dilution factors upon release should be given.

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

Gaseous waste

A. 1232. This section shall describe the treatment of gaseous sources that are considered to be waste, including:

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

A. 1233. If applicable, the design provisions to handle gaseous material with a potential for explosion should be described.

Dose assessment for normal operation

Doses to the general public

A. 1234. This section shall demonstrate that the combined effect of direct radiation and of releases of radioactive material from the facility does not result in off-site doses to the general public that exceed regulatory limits. In addition, measures to reduce exposures on the basis of the ALARA principle should be described.

A.1235. If previous sections of this chapter have demonstrated that the radioactive releases are a small fraction of the operational emission limits and are acceptable, and that direct and indirect radiation is also within acceptable limits, this section shall provide only a summary of the total effect of all pathways of radiation: airborne, liquid, direct and indirect radiation exposure.

A.1236. If radioactive releases have not been treated in terms of operational emission limits, then this section shall include a calculation of the individual doses at the facility boundary and at off-site locations due to the effect of all releases. It should be shown that the combined effect of all releases meets the regulatory requirements for doses to the public.

A.1237. This section shall state the criteria to be used for determining that the releases of gaseous and liquid waste are at an acceptable rate. The effluent concentrations by radionuclide release and the total annual release shall be included, together with the methods, parameters and assumptions used in calculating these quantities.

A.1238. In addition, for gaseous effluents, all points of release to the environment shall be identified, providing for each quantity:

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

Occupational doses

A.1239. This section shall present a diagram showing the radiation fields in normally occupied areas of the facility and in areas where maintenance activities will be performed. Estimated annual occupancy data for the facility radiation areas shall be used to show that the expected doses are acceptable for the major functions, such as reactor operation, conduct of experiments, normal maintenance, radwaste handling, refuelling and in-service inspection. An estimate of the annual dose at the boundaries of the restricted area shall be provided.

A.1240. This section shall 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 facility staff shall be provided.

Conclusion

A.1241. This section shall give a conclusion regarding the acceptability of the operational radiological safety programmes and the design features at the facility.

A.13. CONDUCT OF OPERATIONS

A.1301. This chapter shall describe the organizational structure and the way in which the operating organization will conduct operations of the facility; this shall include the staffing, review and audit of operations, operating procedures, maintenance, testing and inspection, security, and records and reports. General requirements and additional guidance on the above topics can be found in Safety Series No. 35-S2.

Organizational structure

A.1302. The structure of the operating organization of the facility shall be described in this section. The key personnel and the groups at the various operating levels of the facility should be illustrated in an operational diagram. The functions, authority and responsibility of key personnel in the operating organization shall be described.

A.1303. The organizational functions for which it is planned to use off-site or external groups shall be indicated.

A.1304. This section shall provide data for the personnel required during the different operational states of the reactor.

Staff qualification and training

A.1305. This section should describe the qualifications of key personnel.

A.1306. This section should indicate the kind of training required for various personnel and how often the required training will be provided. All licensing or qualification requirements for the staff should be discussed. Training requirements for facility users and instructions for visitors, if any, should be given.

Review and audit

A.1307. This section shall describe the methods for the review and audit of the safety aspects of facility operations, as well as the composition and qualifications of the review and audit group, the rules for group meetings, and the items to be reviewed by the group, such as changes to the licence, the operational limits and conditions, the procedures and the facility; new tests; experiments and procedures; and evaluation of unplanned events.

A.1308. Information on the audit function of the group should be provided, including the items to be audited, the intervals between audits, and how the audit findings are addressed by the facility management within the QA programme for facility operation (see Chapter A.18, Quality Assurance).

Operating instructions and procedures

A.1309. This section shall describe the operating procedures or provide an overview of the operating manual which contains these procedures.

A.1310. These written instructions and procedures should include information on the following items:

- Reactor startup, operation and shutdown;
- Loading, unloading and movement of fuel and irradiated material;
- Inspection and tests of items important to safety, in particular the safety systems;
- Setting up, testing and conducting performance tests of experiments with safety significance;

— Maintenance, in particular concerning major components or systems important to safety;
— Radiation protection;
— Response to anticipated abnormal occurrences, system or component failures and accident conditions;
— Effluent monitoring and environmental surveillance;
— Emergency situations;
— Physical security; and
— Fire protection.

The SAR shall describe how to perform major, minor and temporary modifications to procedures.

Maintenance, testing and inspection

A.1311. This section shall describe the conduct of facility maintenance and periodic testing, and the inspection programme for reactor equipment and components. An overview is sufficient if the detailed programme is given in supplementary documents. The maintenance, testing and inspection programme should provide information on:

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

Physical security

A.1312. The measures established to protect against sabotage and unauthorized removal of fissile and radioactive material shall be described, including regulations of access to the facility and the security systems.

A.1313. The physical security of the facility may be described in a separate document, which would be confidential.

Records and reports

A.1314. This section shall provide information on the facility system for controlling operational records, data and reports that are important for safety. These records may comprise data on:

(a) Reactor operation (log-book, strip charts, check-lists, automatic data readout);
(b) Operational status (type and number of operational components and of components out of service);

(c) Maintenance, inspection and testing protocols;
(d) Records of modifications;
(e) Irradiation of samples and radionuclides produced;
(f) Movement of fissile material;
(g) Radiation levels;
(h) Radiation exposure (external and internal), radiation doses to personnel and records on medical examinations;
(i) Effluent and environmental monitoring;
(j) Safety related component failures and occurrences;
(k) Documents on training and retraining.

A.1315. This section shall give the minimum time interval for which records are to be stored in accordance with the QA programme for the facility operation (see Chapter A.18, Quality Assurance).

A.14. ENVIRONMENTAL ASSESSMENT

A.1401. This chapter of the SAR should provide a summary of the environmental reports for construction, operation and modification of the facility.

A.1402. This section should briefly discuss the following points, in connection with the related information included in Chapter A.3 (Site Characteristics):

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

A.1403. Some licensing actions may have little or no environmental effect. In these cases, the decision to take these actions should be stated and briefly justified.

A.15. COMMISSIONING

A.1501. This chapter of the SAR shall describe the technical aspects of the commissioning programme for the facility in sufficient detail to show that the functional requirements of structures, systems and components will be verified adequately. Complete details of commissioning may be provided in a separate commissioning document.

Commissioning programme

A.1502. This section shall provide the following information concerning the commissioning programme:

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

Specific requirements

A.1503. This section shall contain a description of how the commissioning information from similar operational facilities will be utilized. The method for reporting the commissioning results to the regulatory body shall be described, including a resolution regarding non-conforming or unexpected results.

A.1504. This section shall describe the method for updating the SAR, if required, to include the results of commissioning tests.

A.1505. For existing facilities, this section shall summarize the commissioning programme and results.

Commissioning of modifications

A.1506. The information outlined in the foregoing shall also be included in a separate SAR involving modifications to existing facilities.

A.16. SAFETY ANALYSIS

A.1601. The safety analyses presented in this chapter form the focal point of the SAR. In previous chapters, the structures, systems and components important to safety should be evaluated for their susceptibility to malfunctions and failure. In this chapter, the effects of anticipated process disturbances and postulated component failures and human errors (postulated initiating events) shall be described, including their consequences, to evaluate the capability of the reactor to control or accommodate such situations and failures.

A.1602. To ensure completeness of presentation and to facilitate the review and assessment by the regulatory body, this chapter of the SAR shall contain the following information:

(a) *Introduction:* The general approach and methods used in the safety analyses (paras A.1603 and A.1604);
(b) *Reactor characteristics:* The reactor parameters and initial conditions used in the safety analyses (paras A.1605–A.1609);
(c) *Selection of initiating events:* The spectrum of events initiating accidents considered in the analyses (paras A.1610–A.1612);
(d) *Evaluation of individual event sequences:* The results of the analyses (paras A.1613–A.1645):
(e) *Summary:* A summary of significant results and conclusions regarding acceptability (paras A.1646 and A.1647).

Introduction

A.1603. This section shall provide an overview of the methods and approaches used in the safety analysis. The information provided should be sufficient for a reviewer to obtain a basic understanding of the methods used and of the general nature of the criteria used to assess the acceptability of the results. Annex I may be of some assistance in completing this section, but the level of detail given in this annex is not required here.

A.1604. This section shall provide a brief summary, under the following headings:

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

Reactor characteristics

A.1605. This section should summarize the reactor parameters and the initial conditions used in transient analysis (paras A.1618–A.1623). These parameters and the permitted operating bands will form the basis of the operational limits and conditions in Chapter A.17 (Operational Limits and Conditions).

Core parameters

A.1606. A summary of the reactor parameters and the ranges for specified operating conditions considered in the safety analysis shall be given. Although these values may be tabulated in various other sections of the SAR, they should be summarized here to assist in the review and assessment of the safety analysis. These parameters should include:

— Core power;
— Core inlet temperature;
— Fuel element cladding temperature;
— Reactor system pressure;
— Core flow;
— Axial and radial power distribution and hot channel factor;
— Reactor kinetics;
— Fuel and moderator temperature reactivity coefficients;
— Void reactivity coefficient;
— Available shutdown reactivity worth; and
— Insertion characteristics of reactivity control and safety devices.

A.1607. A range of values shall be specified for reactor parameters that vary with fuel burnup, refuelling or other factors.

A.1608. The permitted operating band on the reactor system parameters shall be specified, including the permitted fluctuations in a given parameter and the associated uncertainties. The most adverse conditions within the operating band shall be used as initial conditions for transient analysis.

Assumed functions of the reactor protection system

A.1609. The settings of all protection system functions that are used in the safety analysis shall be listed. Typical protection system functions are reactor trip, isolation valve closures and backup cooling.

Selection of initiating events

A.1610. This section shall list the postulated initiating events that are treated in the safety analysis. The list shall be comprehensive, and justification for the rejection of particular initiating events shall be provided. Annex I provides some information on methodologies. The selection should consider the points mentioned in paras A.1611 and A.1612.

A.1611. Each postulated initiating event should be assigned to one of the following categories, or grouped in some other manner consistent with the type of reactor under study:

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

A.1612. The initiating events in each group shall be evaluated to identify the events that would be limiting, and the events selected for further analysis shall be indicated. Such events would include those having potential consequences that are limiting for all other initiating events in the group.

Evaluation of individual event sequences

A.1613. The detailed information listed below should be given for each initiating event selected in para. A.1612. This information is organized under the following headings:

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

A.1614. The extent of the quantitative information that should be given for these topics will differ for the various initiating events and depends on the reactor type. For those situations where a particular initiating event is not limiting, only the qualitative reasoning which led to that conclusion need be given, together with a reference to the section presenting an evaluation of the more limiting initiating event. Further, for those initiating events which require a quantitative analysis, it may not be necessary to provide such an analysis for each topic. For example, there are a number of events initiating a reactor transient that result in minimal radiological consequences. The SAR should merely present a qualitative evaluation to show that this is the case. A detailed evaluation of the radiological consequences need not be performed for each such initiating event.

Identification of causes

A.1615. For each event evaluated, a description of the occurrences that led to the initiating event under consideration should be included.

Sequence of events and system operation

A.1616. The step by step sequence of events, from the event initiation to the final stabilized condition, shall be described. The following should be provided for each event sequence:

(a) Identification of significant occurrences on a time-scale, e.g. flux monitor trip or start of insertion of control rods;
(b) Indication of correct and incorrect functioning of normally operating reactor instrumentation and controls;
(c) Indication of both the correct functioning of reactor protection and safety systems and their failure to function;
(d) Indication of the required operator actions;
(e) Evaluation of dependent failures and human errors;
(f) Qualitative evaluation of sequence probabilities (if employed); and
(g) Justification for exclusion of sequences that are outside the design basis.

A.1617. Not every postulated initiating event need be completely analysed and described. The event sequences which are the limiting or bounding sequences in each class and which have been selected for further analysis shall be indicated.

Transient analysis

A.1618. A detailed analysis of the core and system performance shall be given. The methods used to characterize the reactor core and system performance under accident conditions shall be discussed and the important results of the analysis presented. The discussion should include, where appropriate, an evaluation of the parameters that may affect the performance of barriers restricting the transport of radioactive material from the fuel to the environment (e.g. fuel cladding, primary cooling system and building/systems providing confinement).

A.1619. *Computational models.* The computational models employed, including digital computer programs or analogue simulations used in the analysis, shall be identified. It should be confirmed that the models are applicable for the expected range of operational parameters, yield conservative predictions, represent all important physical phenomena and have been validated properly. This section should provide only a summary of the mathematical models and digital computer programs

or lists, referring to detailed descriptions in documents that are available to the regulatory body. The following should also be provided:

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

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

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

A.1620. *Input parameters and initial conditions.* The input parameters and initial conditions used in the analysis shall be clearly identified. Annex II provides a representative list of these items. However, the initial values of other variables and additional parameters should be included in the SAR if they are used in the analysis of the event being analysed.

A.1621. *Results.* The results of the analysis shall be presented and described in the SAR. Key parameters should be given as a function of the time of the transient or accident. The following are examples of parameters that should be included:

— Reactivity;
— Thermal power;
— Heat fluxes;

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

A.1622. Uncertainties in the results shall be pointed out and discussed.

A.1623. The margins between the predicted values of various core parameters and the values of these parameters that would represent the boundaries of acceptable conditions should be given.

Classification of damage states

A.1624. The transient analysis may show that the fuel design limits have been exceeded, resulting in some fuel cladding damage. An estimate of the type of damage, the quantity of fuel affected and other factors (such as fuel and cladding temperatures, coolant characteristics and chemical interactions) should be provided.

A.1625. Some event sequences may result in different radiological hazards, including failure of experiments or of irradiation/activation facilities and mechanical damage to the cladding of the irradiated fuel. An estimate of the form and content of the hazardous material, together with any physical parameters that further characterize its nature, shall be provided. Any regrouping of the sequences within the class according to the type and extent of the radiological hazard shall be described. Sequences that result in no hazard should be excluded, and the remaining sequences which are bounding or limiting for each category of hazard should be selected for analysis of the releases of radioactive material.

Derivation of source terms

A.1626. The source terms, if any, for each bounding sequence mentioned in the previous section shall be described. Such a description should include the quantity of radioactive material that might be released from the reactor, its physical and

chemical form, and any other factors necessary to completely specify its potential dispersion to the environment.

A.1627. This section shall indicate whether detailed calculations of realistic release fractions have been performed or whether conservative release fractions, in accordance with the practices of the regulatory body, have been employed, such as an arbitrary source term which is larger than that expected for probable accident sequences (e.g. to demonstrate the effectiveness of the building/confinement or to show that the resulting doses to critical groups meet regulatory requirements).

A.1628. Mathematical models used in determining and analysing the source term should be summarized and information on validation presented. The information given in paras A.1629–1631 should be provided for each limiting event sequence, where appropriate.

A.1629. *Assessment of releases to the reactor building.* The quantity of radionuclides released inside the building, the isotopic content and other physical factors characterizing the releases should be described for each relevant event sequence. The parameters and assumptions used in the analysis should be presented, including:

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

A.1630. *Assessment of releases from the reactor building.* The quantity of radionuclides released to the environment, the isotopic content and other physical factors characterizing the release should be given for each of the event sequences that results in releases to the reactor building. Both airborne and aqueous releases should be considered. The parameters and assumptions used in the analysis should be presented, including:

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

A.1631. *Assessment of other hazards.* Descriptions should also be given of accidents which might result in significant exposure of staff members or of the general public to direct radiation fields associated with any releases that are contained within the reactor building (see also para. A.1638). Examples are:

— Inadvertent criticality;
— Releases from an experiment or an irradiation facility which are contained but which present a radiation hazard;
— Aqueous spills or releases of other radioactive material which are contained locally; and
— Loss of shielding.

Evaluation of the radiological consequences

A.1632. This section shall discuss the calculational methods used to determine the 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 analysis to be performed by the regulatory body.

A.1633. If no radiological consequences are associated with a given event sequence, this section should simply contain a statement to that effect.

A.1634. *Methods for analysis of the radiological consequences.* The methods used to analyse the radiological consequences that might result from reactor accidents shall be presented in this section. The assumptions and the methods used in determining the radiological consequences shall be supported by providing adequate information, where appropriate, by referring to other sections within the SAR, or by referring to documents readily available to the regulatory body.

A.1635. Information on the modelling of radiological consequences should include the following:

— A description of the mathematical or physical models employed, including any simplifications or approximations introduced in the analysis;
— A summary of the digital computer programs or analogue simulations used in the analyses, referring to detailed descriptions;
— Information on the validation of the calculational methods; and
— Consideration of uncertainties in the calculational methods, the equipment performance, instrumentation response characteristics or other intermediate effects taken into account in the evaluation of the results.

A.1636. *Dose results.* This section shall present the results of the dose calculations, giving the effective dose equivalent at the site or the exclusion boundary and, if

necessary, the effective dose equivalent for the general public at greater distances from the site. In these cases, the dose to the most highly exposed member of the public should be given, as well as the doses during the accident to the control room operators and the personnel at other places on the site, where appropriate.

A.1637. Consideration should be given to direct radiation fields, from both aqueous and atmospheric releases, and to the possibility of ground contamination.

A.1638. *Direct radiation fields.* Direct radiation fields associated with releases occurring within the facility and which could result in radiation doses should be described, together with estimates of doses to critical groups. The parameters and assumptions used in the analysis should be justified, including:

- The quantity of radionuclides released, their geometrical configuration and the time-scale of the release;
- Radionuclide decay and precursor production;
- Shielding parameters, buildup factors and scattering (e.g. sky shine); and
- Distance to critical groups and the time-scale over which doses are calculated.

A.1639. *Aqueous releases.* This section should summarize the assessment of aqueous releases and, where appropriate, dispersion of releases in surface waters and groundwaters, contamination of biological chains and food-chains, and the consequent doses to individuals and the population. Reference should be made to paras A.311–A.314 for data on hydrological and hydrogeological characteristics of surface water and groundwater. The discussion of potential hazards should include:

- Direct radiation from released fluids;
- Evaporation or airborne resuspension of radionuclides from the released fluids;
- Ground contamination; and
- Contamination of aquifers on and off the site.

A.1640. The parameters and assumptions used in the analysis should be justified, including:

- Radionuclide removal by liquid hold-up or recirculation systems;
- Potential discharge points, the inventory of radionuclides released, their concentration in the fluid, the release rate and the mode of release (continuous, intermittent);
- Radionuclide decay and precursor production;
- Dilution and dispersion characteristics, including the migration and retention characteristics of soils, radionuclide movement in hydrogeological formations, the reconcentration ability of sediments and biota, and other effects which may be needed to determine radionuclide movement and exposure pathways;

— Direct and indirect pathways for radioactive contamination of the food-chain; and
— Radionuclide uptake by humans and consequent doses.

A.1641. Special attention should be paid to ascertaining those characteristics important for the determination of food-chain transport.

A.1642. If aqueous releases to surface or groundwater aquifers are judged to be credible, the provisions for containment of any liquid releases within the facility shall be described and the possibility of failure of these provisions discussed.

A.1643. *Atmospheric releases.* This section shall present the doses to facility staff and to the general public after an airborne release of radioactive material from the facility, taking into account atmospheric dispersion, where appropriate.

A.1644. The parameters and assumptions used in the analysis shall be presented and shown to be conservative, including:

— The source term, characterizing it in terms of the radionuclide inventory, the physical and chemical form, and any other factors necessary to completely specify the dispersion of radioactive material to the environment, including buoyancy;
— Mode of release (single burst, intermittent, continuous);
— Location of release and characteristics, including stack height and diameter;
— Distance to receptors and the intervening terrain;
— Meteorology data, including wind speed and direction, and data on inversions and on atmospheric stability;
— Wake effects of the building;
— Diffusion parameters;
— The physical and chemical form of radionuclides at the receptor location, and whether they are airborne or deposited; and
— Results of dose calculation (immersion, ingestion and/or ground shine).

A.1645. *Ground contamination.* This section shall discuss possible ground contamination, by either direct dispersion of particulate radioactive material or deposition from airborne or aqueous releases. The surface contamination by radionuclides shall be estimated and the doses (due to ground shine and ingestion) assessed.

Summary

A.1646. This section shall summarize the important results of the safety analysis, including a brief description of the dominant accident sequences. Significant conclu-

sions arising from the analyses should be presented. The effect of uncertainties of the results should be discussed and evaluated.

A.1647. The results of the analyses shall be compared with the appropriate acceptance criteria. It should be shown that the criteria discussed in paras 217–219 have been met. An evaluation of the results should demonstrate that the design is acceptable and should confirm the validity of the operational limits and conditions discussed in Chapter A.17 (Operational Limits and Conditions).

A.17. OPERATIONAL LIMITS AND CONDITIONS

A.1701. This chapter of the SAR shall contain the operational limits and conditions (OLCs) important to safe reactor operation which have been derived from the safety analysis. The OLCs represent an envelope of parameters, developed by the operating organization, which will protect the reactor, the staff, the general public and the environment from undue exposure if they are not exceeded. Therefore, it is essential that the OLCs are understood by the responsible operating personnel. The OLCs include safety limits, safety system settings, limiting conditions for safe operation, and surveillance and administrative requirements. Additional information is contained in paras 601–608 of Safety Series No. 35-S2.

A.1702. The OLCs are based on an agreement between the reactor operator and the regulatory body and form an important part of the requirements for authorization of the operation of the facility by the regulatory body. Changes to the OLCs shall require a revision of the SAR and assessment and approval by the regulatory body.

A.1703. Because of the important role of the OLCs in safe operation, each OLC must be selected and appropriately substantiated by a written statement of the reason for its adoption. This information shall either be presented in a separate document or included in this chapter of the SAR. In the first case, the information on the OLCs given in the SAR could be a summary of this separate document. In both cases, the information on each OLC should cover the following points:

(a) The objectives to be met by the establishment of OLCs (e.g. prevention of situations that might lead to accident conditions).
(b) The applicability of the OLCs, for example to physical variables related to physical barriers, such as fuel plate temperature or pool water level, or to conditions of these barriers. Sometimes the applicability refers to the equipment set-up, such as the minimum number of measuring channels being operable.
(c) The specification(s) of the OLC, for example the value that may not be exceeded, or specific conditions on equipment.

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

Safety limits

A.1704. The safety limits for important process variables or parameters shall be stated and justified by the analyses provided in the SAR. Safety limits normally involve operational parameters such as fuel and fuel cladding temperatures, reactor coolant temperature, reactor pressure, reactor power, coolant flow rates and, for pool reactors, the water level above the core. These safety limits are derived primarily from Chapters A.5 (Reactor) and A.16 (Safety Analysis).

Safety system settings

A.1705. Safety system settings shall be provided for those variables and parameters which, if not controlled, could result in a safety limit being exceeded. This section shall identify the safety system settings and provide an analysis showing that the safety limits will not be exceeded. In determining safety system settings, consideration should be given to items such as calibration error, measurement accuracy and system response time. Safety system settings are derived primarily from Chapters A.5 and A.16.

Limiting conditions for safe operation

A.1706. This section shall present the limiting conditions for safe operation, which shall provide acceptable margins between normal operating values and safety system settings. Although in many cases the limiting conditions are established by the administration, they are constraints on equipment and operational characteristics which are identified in the SAR as being important to safety and which should be adhered to during operation of the facility. In some cases, when the process variables or parameters reach a limiting condition for safe operation, they may initiate alarms to enable the operating personnel to take appropriate action in order to prevent safety system settings from being exceeded. Some examples of limiting conditions for safe operation are as follows:

- Core configuration and design limitations (reactivity coefficients, burnup limits, inspection, etc.);
- Minimum number, design and performance of reactivity control mechanisms;
- Fuel design parameters (enrichment, fuel type, cladding type, etc.);
- Maximum reactivity insertion rate;

— Minimum operational reactor measurement and control systems and safety set points;
— Equipment required to achieve confinement or containment;
— Operations that require confinement or containment;
— Minimum operating equipment for ventilation systems;
— Equipment and performance of the emergency power supply system;
— Minimum operational equipment for radiation and effluent monitoring;
— Limits on effluent releases;
— Limitations on experiments (reactivity, materials, etc.);
— Other design limitations important to safety.

Surveillance requirements

A.1707. This section shall discuss the surveillance requirements regarding the frequency and scope of tests, showing that the performance levels set by the safety limits and the limiting conditions for safe operation are being met. The requirements for monitoring, inspection, operability checks and calibrations shall be included and the actions to be taken if a system fails should be described. The conditions for continuing operation during repair or the acceptability of substitution of equipment for failed equipment should be stated.

Administrative requirements

A.1708. This section shall contain the administrative and organizational requirements, as well as the organizational structure and responsibility, the staffing requirements, the review and audit of facility operation procedures, the review of operational events, and reports and records. These limiting conditions and administrative requirements are derived primarily from Chapter A.13 (Conduct of Operations).

A.18. QUALITY ASSURANCE

QA programmes

A.1801. The operating organization is responsible for the preparation and implementation of an overall QA programme that will ensure conformance with every aspect of safety. The principles and scope of this programme shall be established in accordance with the general requirements of Safety Series Nos 35-S1 and 35-S2, and with other national standards.

A.1802. This section shall describe the QA programme or refer to a description of it. A summary shall be provided of the items, services and processes to which the QA programme shall apply and of the organizational structure within which the

QA activities are to be planned or implemented. The level of control and verification of quality shall also be defined, and the means available for achieving it shall be described.

A.1803. This section shall describe or refer to the particular QA programme that has been established for the phases of design, procurement, construction, commissioning and operation, as appropriate. The QA programme shall be consistent with the requirements of the research reactor project, its objectives, status and characteristics, and it shall be acceptable to the regulatory body.

QA procedures

A.1804. This section shall describe or refer to the planning, implementation and control of essential activities related to the QA programme to ensure that the specific requirements, such as regulatory requirements, design and construction criteria, and acceptance criteria, are correctly applied and achieved. In particular, the responsibility and authority of the personnel concerned with QA should be defined.

A.1805. This section shall describe the procedures covering specific activities under the QA programme, such as non-conformances, design changes, design deviations and concessions, and the analysis of their impact on safety requirements.

A.1806. This section shall describe the procedures covering the operating activities performed under the QA programme. Examples are activities related to reactivity and criticality management, thermal safety of the core, safety of experimental devices, reactor modifications, component and material manipulations and human surveillance.

A.1807. This section shall describe how the SAR and supporting documents are identified and filed, and how long the documents are retained, or a reference to such a description shall be given.

Management status of the QA programme

A.1808. This section shall provide a summary report on the current status of management of the QA programme and the status of achievement of the required standards of quality and safety.

A.19. DECOMMISSIONING

A.1901. This chapter of the SAR shall provide information on the facility design and the operational procedures to facilitate the decommissioning process. The design basis related to decommissioning should be described.

A.1902. The aspects of the facility design that facilitate decommissioning should be discussed, such as selection of material to reduce activation and to provide for easy decontamination, detachment and handling (remotely where required) of activated components, and adequate facilities for processing of radioactive waste.

A.1903. This section shall discuss the aspects of facility operation that facilitate decommissioning, such as design provisions and operational practices to reduce activation of material, and maintenance of records of facility construction and contamination.

A.20. EMERGENCY PLANNING AND PREPAREDNESS

Emergency plan

A.2001. This section shall contain or refer to a plan which will provide reasonable assurance that actions can and will be taken to mitigate emergencies that might occur at the facility; this plan shall be prepared by the operating organization. However, safety precautions taken in the design and operation of the reactor will greatly reduce the risk of an accident. Additional information on this plan may be obtained from Safety Series No. 35-S2.

A.2002. This section shall demonstrate that the emergency plan is based on accidents analysed in the SAR and on other accidents postulated only for emergency planning purposes.

A.2003. This section shall provide information on actions to be taken in the reactor building, on the site and off the site. The information shall cover the following items:

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

Emergency procedures

A.2004. This section shall demonstrate that the emergency plan will be implemented by emergency procedures. These procedures shall contain the specific actions which will be taken to mitigate the consequences of emergencies.

A.2005. This section shall contain information on the arrangements for periodic review of the emergency plan and the implementing procedures ensuring that the requirements of new experiments or facility modifications are included.

A.2006. The emergency procedures shall contain guidance on limits to exposure of personnel performing rescue missions or missions to reduce the consequences of an emergency.

Annex I

SAFETY ANALYSIS APPROACH AND METHODS

I-101. Annex I outlines some considerations which may be useful in developing a safety analysis for a research reactor. The well accepted basic approach to this is to consider initiating events for credible accidents, using a deterministic method to estimate the maximum possible releases to the environment. Probabilistic methods may be used to evaluate which accident sequences are of higher likelihood; they will be useful also for evaluating the relative rankings of risks and hence for providing countermeasures. They may also be used for identifying hidden weaknesses of the design and for quantifying the value of possible improvements or modifications. However, probabilistic safety assessment (PSA) is not treated in this Safety Guide since deterministic methods are used. For further information on applications of PSA to research reactors, see IAEA-TECDOCs 400 and 517 (list of IAEA publications given at the end of this Guide).

I-102. These considerations cover a wide spectrum of research reactors and thus may contain information which is not applicable to all reactors. Consequently, these considerations are not intended to be requirements but are provided for additional guidance.

Methods for identification and selection of initiating events

I-103. Postulated initiating events are occurrences that may lead to reactor fault sequences or accident scenarios. They originate from component failures, system malfunctions, human error or external events and special internal events.

I-104. The method used to identify postulated initiating events and to select sets of particular events for further analysis shall be established. This method should ensure that the list of initiating events is as complete as possible, that initiating events are grouped in some logical fashion to simplify the analysis, and that limiting or bounding initiating events in each group are selected for further analysis. Such a method could include one or more of the following:

(a) *Lists of initiating events in research reactors.* A list of possible initiating events in research reactors is given in Table I (see also para. 314).
(b) *Engineering evaluation.* Potential sources and types of radiological hazards within the facility are identified, and a systematic review of the facility design, operations and site factors is made to identify occurrences that could lead to radiological hazards.

(c) *Operational experience.* Past experience from the facility or from similar facilities, including examination of safety reports, can be used to develop or supplement the list of initiating events.

(d) *Logical analysis.* An example is a top-down logical model known as a Master Logic Diagram, which is similar to a fault tree.

I-105. Methods used to reject particular initiating events and to exclude them from further analysis shall be determined. Such methods could lead to rejection of the following initiating events:

(a) *Inconsistent or inappropriate initiating events:* Initiating events that can be justified as falling outside the scope of the analysis or that are found by inspection to be trivial.

(b) *Incredible initiating events:* Initiating events that are not possible for the reactor under study.

(c) *Very rare initiating events:* Initiating events whose frequency of occurrence may be so low that they could be candidates for rejection on probabilistic grounds (e.g. aircraft crashes), using statistical data or conservative estimates. Combinations of mutually independent initiating events, each having a low frequency of occurrence, would also fall under this category.

I-106. Certain methods can be used to group the initiating events as follows:

(a) Initiating events that require similar safety functions which determine the design parameters of the safety systems;

(b) Initiating events that have a similar influence on the reactor behaviour or on structures or components for which similar calculational models are used;

(c) Initiating events that can assist in the selection of limiting cases for analysis in each group; and

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

One possible grouping is shown in para. A.1611 of the Appendix.

I-107. The method developed to select limiting initiating events for further analysis should include those having potential consequences that are limiting for all other initiating events in the group.

Methods for event sequence analysis

I-108. A method shall be developed to evaluate the step by step sequence of events, from the initiation of the event to the final stabilized condition. This method should include any rules or conventions regarding the extent to which reactor systems, including the reactor protection system, are assumed to function. If there

is a possibility of fuel cladding failure, then other barriers to spread of activity should be considered. The outcome of the event sequence has to be considered, not only if all systems function correctly but also if some of them fail. Consideration should be given to the types of events that will be evaluated using this method, and the types of events that will be evaluated by other methods (see paras I-116 to I-120).

I-109. Methods shall be established to investigate event sequences. The sequences should include the response of the reactor and the reactor systems, as well as human interactions; possible event sequences for the case where each safety system functions well and for the case where a system fails should be described. The following points should be considered:

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

I-110. Rules or conventions should be established to determine the response of reactor systems; these rules or conventions should refer to:

(a) The effect of single, random failures;
(b) System qualification (or lack of qualification) under accident conditions;
(c) Safety and protection systems, including reliability in quantitative terms;
(d) Support systems, such as normal and emergency electric power and the cooling water;
(e) Redundant trip parameters;
(f) Actions of systems that are independent;
(g) Operator action (e.g. response time, display of information on a console); and

(h) Carrying out of frequency or probability evaluations to assess the system response, the extent to which such evaluations will be used and the methods to be employed (including validation).

I-112. Rules or conventions should be developed to determine event sequences that are beyond the design basis and thus are excluded from further analysis. Such rules could be based on:

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

I-113. The effects of dependent failures (e.g. common cause or cross linked effects) and human error shall be considered; this includes:

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

I-114. The frequency or probability of event sequences may be evaluated; this would help to determine which sequences should be excluded from the design basis or to assess the relative risk presented by various sequences. This evaluation should include:

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

I-115. Limiting or bounding event sequences in each class should be selected for further analysis in order to reduce the number of events to be analysed using analysis methods of core transients. Consideration should be given to:

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

Methods for evaluation of external events and special internal events

I-116. General methods used to evaluate particular external and internal events, such as earthquakes, tornadoes or sudden catastrophic rupture of reactor pressure retaining components or reactor internals, should be presented in the appropriate chapter of the SAR. It may be difficult to model the effects of such events, or analyses may be highly speculative. Further guidance regarding protection against these events is given in Chapters A.2 and A.3 of the Appendix.

I-117. In general, design qualification is an accepted practice for protection against external events once siting questions have been resolved (i.e. if the site does not present hazards for which there is no adequate protection). The method for establishing the design bases for particular external phenomena can be summarized as follows:

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

I-118. Design qualification may prevent failure of pressure retaining components. In this case the appropriate chapter of the SAR should describe the design and construction standards used (e.g. acceptable engineering codes and practices) to prevent structural failures and to preserve the required safety functions. Reference may be made to the appropriate chapters of the SAR (see Chapters A.2 and A.3 of the Appendix).

Qualitative evaluations

I-119. Consideration should be given to the conditions under which qualitative evaluations are used in the safety analysis to treat particular event sequences, for example:

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

I-120. Such qualitative arguments should be used with caution and the regulatory body should be consulted concerning acceptability.

Acceptance criteria

I-121. The significant results of the safety analysis must be compared with the acceptance criteria (see paras 217–219).

I-122. The SAR should present not only the acceptance criteria appropriate to the safety analysis but also the results of the comparisons referred to in para. I-121.

Annex II

EXAMPLES OF INPUT PARAMETERS AND INITIAL CONDITIONS

II-101. Examples of input parameters and initial conditions which should be identified in the safety analysis are:

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

Annex III

EXAMPLES OF ITEMS TO BE CONSIDERED IN THE REACTOR DESCRIPTION

III-101. *Summary description:* A brief description of the following aspects of the reactor should be provided:

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

III-102. *Reactor structures:* A detailed description of the following items is required:

(a) Reactor pool/vessel;
(b) Core support, grid plate;
(c) Reactor bridge;
(d) Reflector;
(e) Shielding (including movable shielding for beam tubes);
(f) Supports for core instrumentation;
(g) Beam tubes;

(h) In-core test facilities;
(i) Natural circulation provisions.

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

III-103. *Reactivity control system, reactor shutdown system:* The function of the mechanical and electrical design shall be described. The description shall include the materials and dimensions and shall be supported by drawings. The reactivity control mechanisms and their instrumentation, such as their position or status (coupled/decoupled), should be presented, together with their insertion time and interlocks. The effects of corrosion, fatigue, neutron doses, etc., on the lifetime of the mechanical and electrical components shall also be discussed. The safety related design parameters should be presented, such as:

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

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

III-104. *Fuel elements:* The fuel used shall be specified, including the uranium enrichment and the type of fuel (alloy, oxide, aluminide, carbide). The description of the fuel elements shall be supported by drawings, and the main characteristics of the fuel elements should be presented:

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

Experience with the fuel used should be described.

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

III-105. *Reactivity control mechanisms:* The reactivity control mechanisms shall be described, giving the main dimension, the neutron absorber material used, and information on the experience with these or similar reactivity control mechanisms. The description should be supported by drawings.

Annex IV

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

IV-101. Examples of possible radiation sources or radiation fields in a research reactor are:

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

DEFINITIONS

The definitions presented in this Guide are intended principally for use in the IAEA's safety related documents for research reactors and do not necessarily conform to definitions adopted elsewhere for other use. In all cases these definitions are identical with, or at least consistent with, those used in the IAEA Nuclear Safety Standards (NUSS) for nuclear power reactors.

acceptable limits

Limits acceptable to the regulatory body.

accident conditions

Deviations from operational states in which the releases of radioactive material are kept to acceptable limits by appropriate design features. These deviations do not include severe accidents.[9]

accident management

Accident management is the taking of a set of actions:

- During the evolution of an event sequence, before the design basis accident of the facility is exceeded, or
- During a beyond design basis accident, without core degradation, or
- After core degradation has occurred, to bring the facility to a controlled safety state and to mitigate any consequence of the accident.

anticipated operational occurrences

All operational processes deviating from normal operation which are expected to occur once or several times during the operating life of the reactor and which, in view of appropriate design provisions, do not cause any significant damage to items important to safety or lead to **accident conditions**.

applicant

The organization that applies for formal granting of a licence to perform specified activities related to the siting, design and construction, commissioning, utilization and modification and decommissioning of the facility.

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[9] A severe accident is an accident which is beyond **accident conditions** and is a concept used exclusively for nuclear power reactors.

authorization

Granting of written permission to perform specified activities.

commissioning[10]

The process during which the reactor components and systems that have been constructed are made operational and verified to be in accordance with design assumptions and to have met the performance criteria; it includes both non-nuclear and nuclear tests.

construction[10]

The process of manufacturing and assembling the components of a nuclear research reactor facility, the erection of civil works and structures, the installation of components and equipment, and the performance of associated tests (not included in the commissioning stage).

decommissioning[10]

The process by which a reactor is permanently taken out of operation.

design basis accident

Accident conditions against which the research reactor facility is designed according to established design criteria.

disposal

The emplacement of waste in a repository, or at a given location, without the intention of retrieval. Disposal also covers the approved direct discharge of waste to the environment, with subsequent dispersion.

engineered safety features (*see* safety systems)

[10] The terms siting, design, construction, commissioning, operation, modification and decommissioning are used to delineate major stages of the licensing process of research reactors (see Safety Series No. 35-S1, para. 309). Several stages may coexist, for example construction and commissioning, commissioning and operation.

experiment (*or* **experimental device**)

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

interim storage (storage)

Storage of radioactive material such that: (a) isolation, monitoring, environmental protection and human control are provided; and (b) subsequent action, involving treatment, transport and disposal or reprocessing, is expected.

licensee

The holder of a licence issued by the regulatory body to perform specific activities related to the nuclear research reactor facility. The applicant becomes the licensee after he has received a licence issued by the regulatory body.

modification (*or* **reactor modification**) *(see footnote 10)*

A deliberate change in or an addition to the existing reactor configuration, with potential safety implications, intended for continuation of the reactor operation. It may involve **safety systems** or safety related items or systems, procedures, documentation or operating conditions.

normal operation

Operation of a research reactor and associated **experimental devices** within specified **operational limits and conditions**, including startup, power operation, shutting down, shutdown, maintenance, testing and refuelling (*see* **operational states**).

operating organization

The organization authorized by the regulatory body (or by the government) to operate the reactor facility.

operation *(see footnote 10)*

All activities performed to achieve the purpose for which the nuclear research reactor facility was constructed, including maintenance, refuelling and other associated activities.

operational limits and conditions

A set of rules which set forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of the research reactor facility.

operational states

The states defined under **normal operation** and anticipated operational occurrences.

protection system

A system which encompasses all electrical and mechanical devices and circuitry, from sensors to actuation device input terminals, involved in generating signals associated with the protective function (*see also* **shutdown system**).

protective action

Action of the **protection system** calling for the actuation of particular safety actuation devices such as those of the **shutdown system**.

quality assurance

All planned and systematic actions necessary to provide adequate confidence that an item or service will satisfy given requirements for quality.

radioactive waste management

All activities, administrative and operational, that are involved in the handling, treatment, conditioning, transportation, storage and disposal of waste.

reactor management

The members of the **operating organization** who have been delegated responsibility and authority for directing the operation of the research reactor facility.

research reactor

A nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes.[11]

safety (*or* nuclear safety)

The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the general public and the environment from undue radiation hazards.

safety analysis report

A document provided by the **applicant** to the regulatory body containing information concerning the nuclear research reactor facility, its design, safety analysis and provisions to minimize the risk to the public, the operating personnel and the environment.

safety limits

Limits on process variables within which the operation of the research reactor facility has been shown to be safe.

safety margin

The difference between **safety limits** and **operational limits**. It is also sometimes expressed as the ratio of the two values.

safety related items or systems

Items or systems important to safety which are not **safety systems**.

safety systems[12]

Systems important to safety, provided to ensure the safe shutdown of the reactor or heat removal from the core, or to limit the consequences of **anticipated operational occurrences** and **accident conditions**.

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[11] In this publication, the term **research reactor** also includes associated experimental facilities and critical assemblies.

[12] The functions of **safety systems** are initiated upon receipt of a signal from the protection system or manually. Some aspects of **safety systems** are often referred to as engineered safety features, particularly in the context of emergency heat removal and confinement.

safety system settings

The points of actuation of appropriate automatic protective devices which are intended to initiate action to prevent a **safety limit** from being exceeded in the event of **anticipated operational occurrences** and **accident conditions**.

shutdown margin

The negative reactivity provided in addition to that necessary to maintain the reactor in a subcritical condition without a time limit, with the most reactive control device removed from the core and all experiments which can be moved or changed during operation in their most reactive condition.

shutdown reactivity

Amount of reactivity by which the reactor is subcritical, with the control devices introducing maximum negative reactivity.

shutdown system

The system necessary to execute the shutdown of the reactor by rapid reactivity reduction either manually or on the receipt of a signal from the protection system.

site

The area containing the reactor building defined by a boundary and under effective control of the **reactor management**.

siting *(see footnote 10)*

The process of selecting a suitable site for a nuclear research reactor facility, including appropriate assessment and definition of the related design bases.

storage (*see* **interim storage)**

ultimate heat sink

The atmosphere or a body of water or a combination of these to which residual heat is transferred.

utilization (*or* reactor utilization) *(see footnote 10)*

The use of the reactor or of **experiments** or **experimental devices** during operation of the reactor.

waste management (*see* radioactive waste management)

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