IAEA Safety Standards

for protecting people and the environment

Format and Content of the Safety Analysis Report for Nuclear Power Plants

Specific Safety Guide

No. SSG-61

IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

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

https://www.iaea.org/resources/safety-standards

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

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

FORMAT AND CONTENT OF THE SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

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

IAEA SAFETY STANDARDS SERIES No. SSG-61

FORMAT AND CONTENT OF THE SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

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FOREWORD

by Rafael Mariano Grossi
Director General

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

Safety Fundamentals

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

Safety Requirements

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

Safety Guides

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

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

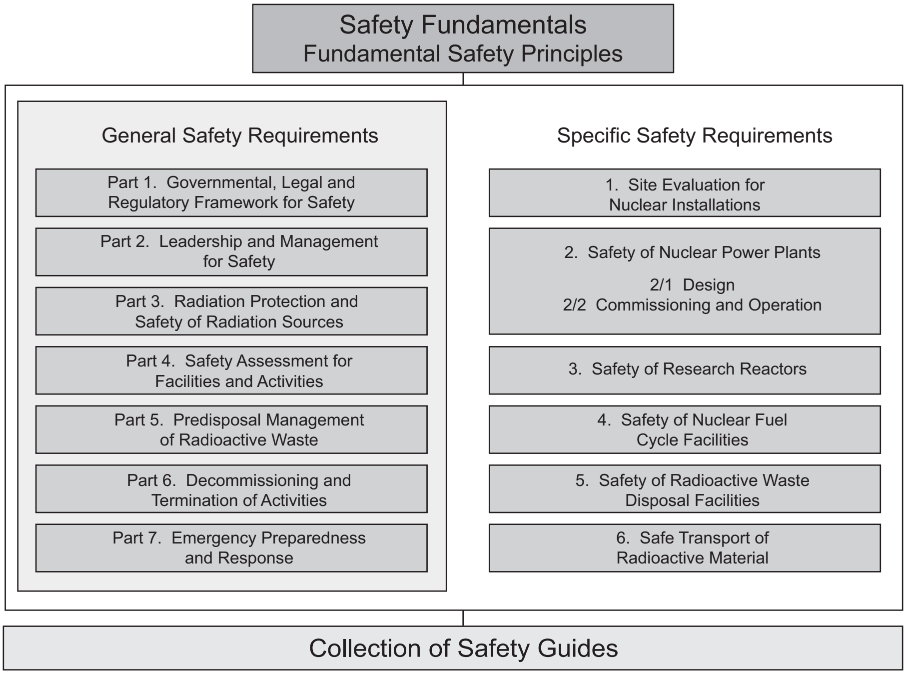

FIG. 1. The long term structure of the IAEA Safety Standards Series.

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

APPLICATION OF THE IAEA SAFETY STANDARDS

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

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

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

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

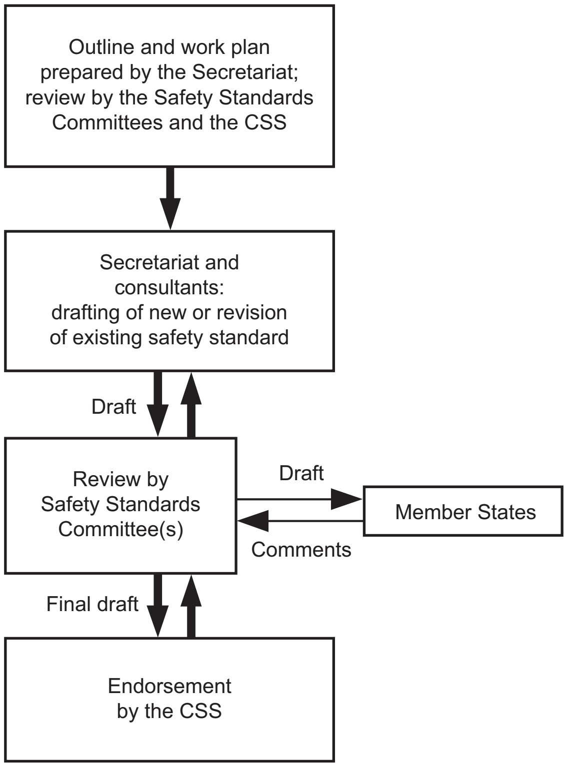

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

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

INTERPRETATION OF THE TEXT

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

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

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

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

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. For an operating organization to obtain regulatory approval to build and operate a nuclear power plant, an authorization[1] is required to be requested from and granted by the relevant regulatory body. In accordance with paras 4.33 and 4.34 of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [1], the regulatory body is required to issue guidance on the format and content of documents to be submitted by the applicant in support of applications for authorization, and the applicant is required to submit or make available to the regulatory body, in accordance with agreed timelines, all necessary safety related information as specified in advance or as requested in the authorization process.

1.2. The information to be submitted by the applicant should be presented mainly in the form of a report, hereinafter referred to as the 'safety analysis report'. Further requirements on the documentation of the safety assessment for a facility in the form of a safety analysis report; on the objectives, scope and level of detail of this report; and on updating the safety analysis report are established in Requirement 20 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2].

1.3. This Safety Guide is a revision of IAEA Safety Standards Series No. GS-G-4.1, Format and Content of the Safety Analysis Report for Nuclear Power Plants, which it supersedes.[2] The revision reflects good practices and experience in the use of safety analysis reports for newly built nuclear power plants in different States; it also reflects recent progress made in approaches to safety assessment.

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[1] The authorization is expected to be granted by the issue of a licence or permit by the regulatory body. Consequently, the term 'licensing' is also used in this Safety Guide.

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

1.4. Since the publication of the previous version of this Safety Guide, several IAEA Safety Requirements publications have been revised to establish enhanced requirements for the safety of nuclear power plants, in particular the following:

(a) IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3];
(b) IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [4];
(c) IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [5].

The most significant changes made to this Safety Guide are those corresponding to the new safety requirements established in SSR-2/1 (Rev. 1) [3], in particular the requirements regarding design extension conditions, the strengthening of the independence and effectiveness of the different levels of defence in depth, the robustness of the plant against extreme external hazards, and the practical elimination of event sequences that could lead to an early radioactive release or a large radioactive release. The importance of addressing these changes was also strongly highlighted by the feedback of experience and lessons from the accident at the Fukushima Daiichi nuclear power plant.

1.5. The recommendations provided in this Safety Guide aim to maintain consistency between the content of the safety analysis report and the safety requirements established in the IAEA safety standards. In addition, applicable national and multinational guidance documents (e.g. Refs [6–9]) were taken into account in the development of this Safety Guide.

1.6. The terms used in this Safety Guide are to be understood as defined and explained in the IAEA Safety Glossary [10].

OBJECTIVE

1.7. The objective of this Safety Guide is to provide recommendations on the structure and content of the safety analysis report to be submitted by the operating organization in support of an application to the regulatory body for authorization of the siting, construction, commissioning, operation and decommissioning of a nuclear power plant. This Safety Guide is intended to facilitate both the development of the safety analysis report by the operating organization and the checking of the completeness and adequacy of the safety analysis report by the regulatory body. The content of the safety analysis report recommended in this

Safety Guide is designed to ensure that the information provided about the safety of the nuclear power plant is comprehensive and is sufficient to demonstrate compliance with the relevant IAEA safety requirements and recommendations.

SCOPE

1.8. This Safety Guide is intended mainly for use in the authorization process for nuclear power plants, although it may, in parts, have a wider applicability to other nuclear installations or facilities. In accordance with current practices, this Safety Guide applies also to the authorization process for units of a multiple unit nuclear power plant.

1.9. This Safety Guide was written to apply to water cooled reactors, in particular to light water reactors, although many sections and subsections may also be applicable for other reactor types. The particular contents of the safety analysis report for these reactor types will depend on the specific design of the nuclear power plant, which will determine how the sections and subsections described in this Safety Guide are included in the safety analysis report.

1.10. This Safety Guide assumes that the safety analysis report will be developed in accordance with the different stages of authorization of the nuclear power plant and that the report will be updated on a regular basis to reflect the configuration of the nuclear power plant at each stage of its lifetime. Consequently, it is expected that the structure of the safety analysis report will be maintained as far as possible throughout its development process, from siting to decommissioning of the nuclear power plant.

1.11. Although intended mainly for use for new nuclear power plants, the recommendations presented in this Safety Guide should also be used, as far as practicable, for existing nuclear power plants when the operating organization reviews the existing safety analysis report to identify any areas in which improvements to the safety analysis report may be appropriate. Such improvements should focus on extending the scope and enhancing the quality of the information provided in the safety analysis report, rather than on changing the structure of the safety analysis report.

STRUCTURE

1.12. This Safety Guide has two main parts: one containing general recommendations relating to the safety analysis report and one devoted to the structure and content of individual chapters of the safety analysis report. The general recommendations are set out in Section 2 and cover the following issues:

(a) The role of the safety analysis report;
(b) Safety rules of different origins;
(c) The structure and outline of the safety analysis report for various stages of the lifetime of the nuclear power plant;
(d) The structure of the safety analysis report;
(e) A unified description of the design of plant structures, systems and components (SSCs);
(f) The use, review and updating of the safety analysis report during plant operation;
(g) Formal aspects of the safety analysis report;
(h) The relationship of the safety analysis report to other licensing documents;
(i) The treatment of sensitive information;
(j) The structure of the safety analysis report for different types of nuclear installation.

1.13. Section 3 provides specific recommendations on the structure and content of each of the chapters of the safety analysis report and is further supported by two appendices. Appendix I indicates the type of information to be provided in each chapter of the safety analysis report at different stages of the licensing process. Appendix II presents a unified content and structure for the information to be provided for the different SSCs described in the safety analysis report.

1.14. An example of a detailed list of contents of a safety analysis report is provided in the Annex.

1.15. The structure proposed in this Safety Guide, including the subdivision of the safety analysis report into different chapters, should not be interpreted as having to be followed verbatim. In each specific case, the operating organization should agree with the regulatory body on the content, structure, form of presentation, storage and use of the safety analysis report.

2. GENERAL CONSIDERATIONS

ROLE OF THE SAFETY ANALYSIS REPORT

2.1. The safety analysis report is a key licensing document, developed by the operating organization and used by the regulatory body in assessing the adequacy of plant safety in all stages of the lifetime of a nuclear power plant to determine the suitability of the licensing basis. The safety analysis report, compiled either as a single document or as an integrated set of documents that collectively constitute the licensing basis of the plant, should provide an adequate demonstration that the nuclear power plant meets all applicable safety requirements.

2.2. At later stages of the lifetime of the plant, the safety analysis report should also adequately demonstrate that the plant has been built and commissioned as intended; that any changes in design, construction and commissioning have been properly addressed; and that the safety aspects of interactions between technical, human and organizational factors have been duly considered throughout the report.

2.3. In addition to providing a documented demonstration that the plant has been designed to appropriate safety standards, the safety analysis report should be able to demonstrate that the plant will be operated safely and should provide related reference material for the safe operation of the plant. While it might not be feasible to present all the relevant information in the safety analysis report itself, the information should be presented in such a way that the regulatory body can conduct the review and assessment process with only a limited need for additional documentation.

SAFETY RULES OF DIFFERENT ORIGINS

2.4. A nuclear power plant is a strictly regulated nuclear installation, subject to a number of applicable rules, including international conventions, national laws and regulations, international or regional safety standards and nuclear security guidance, regulations of the country of origin, quality standards, and technical norms. These rules include those addressing the classification of SSCs and those addressing fire protection, radiation protection, civil construction, and occupational health and safety. The safety analysis report should present the whole set of applicable rules, including principles for their hierarchical application, with a specified process to resolve any potential differences that might arise between rules of different origin.

STRUCTURE OF THE SAFETY ANALYSIS REPORT FOR VARIOUS STAGES OF THE LIFETIME OF A NUCLEAR POWER PLANT

2.5. It is common practice in many States to develop different versions of the safety analysis report for different licensing stages of the nuclear power plant. Although the approach, title, content and structure of the safety analysis report for different licensing stages vary among States, there are typically three report development stages:

(a) Initial safety analysis report, which includes the basis for the site authorization;
(b) Preliminary safety analysis report (often abbreviated to PSAR), which includes the basis for the authorization of construction;
(c) Pre-operational safety analysis report, which includes the basis for the authorization of the commissioning and operation of the nuclear power plant.

During operation of the nuclear power plant, the pre-operational safety analysis report should be further complemented by additional information, leading to the issue of the operational safety analysis report or final safety analysis report (often abbreviated to FSAR).

2.6. The structure of the safety analysis report proposed in this Safety Guide is best suited to the preliminary, pre-operational and final safety analysis reports. Nevertheless, the same structure of the safety analysis report should be maintained as far as possible throughout its development, from the initial safety analysis report to the pre-operational safety analysis report. In general, more information will be generated from operating experience. As a guiding principle, any new version of the safety analysis report should provide updated and revised information on the topics outlined in the previous issues of the safety analysis report and should explain and justify any significant differences from previous safety considerations. The level of information expected in the individual chapters of the safety analysis report at different licensing stages is indicated in Appendix I.

2.7. At the stage of the initial safety analysis report, information about the nuclear power plant might be limited, while information about the site is likely to be reasonably complete. Although the future reactor design might not have been selected yet, the impact of the future nuclear power plant on both the site and its environment should be based on a reasonable estimate, for example using a bounding case approach[3]. Rather than describing the safety features of the future nuclear power plant, the initial safety analysis report should describe relevant safety principles and requirements and should, to some extent, indicate how these requirements will be complied with. Since in many cases the initial safety analysis report consists of a summary of requirements, and these requirements are typically not described in detail, it may be practicable to combine several sections of a given chapter of the safety analysis report into one integrated section.

2.8. The preliminary safety analysis report should contain sufficiently detailed information, specifications and supporting calculations to assess and demonstrate that the plant can be constructed, commissioned, operated and decommissioned in a manner that is acceptably safe throughout its lifetime. The preliminary safety analysis report should demonstrate that the requirements specified in the initial safety analysis report are met. The safety features incorporated into the design should be described, with due regard to any site specific aspects.[4]

2.9. The pre-operational safety analysis report should contain revisions of, and provide more specific information on, the topics outlined in the preliminary safety analysis report. The pre-operational safety analysis report should take into account all modifications implemented during the design and construction stages of the nuclear power plant and should provide a justification of any differences from, or revisions to, the safety considerations or the design intent as set out in the preliminary safety analysis report. The pre-operational safety analysis report should provide a justification of the final detailed design of the plant and present a demonstration of its safety. In addition, the pre-operational safety analysis report should address in greater detail (i.e. than in the preliminary safety analysis report) issues relating to the commissioning and operation of the plant during the pre-operational stage. The pre-operational safety analysis report should also provide more up to date information on the licensing basis for the plant.

2.10. Initially, the final safety analysis report should be prepared as an update of the pre-operational safety analysis report. Additional information obtained during the operational stage of the nuclear power plant should be incorporated

[3] The bounding case approach includes the identification of important physical and chemical parameters that might affect the environment for the nuclear power plant considered and the use of those parameters with the highest impact value.

[4] In some cases (e.g. in States deploying a given reactor design in several units), the amount of information to be provided in the preliminary safety analysis report might depend on the extent to which the proposed reactor design conforms to a generic or standard design for which the licensing process has been followed previously, including the associated safety analysis report.

periodically into the final safety analysis report. This information should include any plant modifications with their justification. Particular attention should be given to documenting information that is relevant to the decommissioning of the nuclear power plant.

2.11. This Safety Guide considers periodic updates of the approach and associated conditions regarding the future decommissioning of the nuclear power plant (see paras 3.21.1–3.21.10). However, it does not specifically address the scope of the safety analysis report for an advanced decommissioning phase, when the nuclear fuel has been removed from the plant after a suitable cooling period.

STRUCTURE OF THE SAFETY ANALYSIS REPORT

2.12. The safety analysis report should be structured as follows:

— Chapter 1: Introduction and general considerations;
— Chapter 2: Site characteristics;
— Chapter 3: Safety objectives and design rules for structures, systems and components;
— Chapter 4: Reactor;
— Chapter 5: Reactor coolant system and associated systems;
— Chapter 6: Engineered safety features;
— Chapter 7: Instrumentation and control;
— Chapter 8: Electrical power;
— Chapter 9: Auxiliary systems and civil structures;
— Chapter 10: Steam and power conversion systems;
— Chapter 11: Management of radioactive waste;
— Chapter 12: Radiation protection;
— Chapter 13: Conduct of operations;
— Chapter 14: Plant construction and commissioning;
— Chapter 15: Safety analysis;
— Chapter 16: Operational limits and conditions for safe operation;
— Chapter 17: Management for safety;
— Chapter 18: Human factors engineering;
— Chapter 19: Emergency preparedness and response;
— Chapter 20: Environmental aspects;
— Chapter 21: Decommissioning and end of life aspects.

2.13. The Annex to this Safety Guide provides an example detailed structure for each chapter of the safety analysis report. The main objective of the Annex is to indicate the expected comprehensiveness of information to be provided in the safety analysis report.

2.14. The proposed structure of the safety analysis report incorporates several chapters that have often been covered by separate documents. Examples of such chapters are those on operational limits and conditions (OLCs) for safe operation, management for safety, emergency preparedness and response, environmental aspects, and decommissioning and end of life aspects. While in general it is acceptable to have separate documents to complement the safety analysis report, at least for new nuclear power plants all such additional documents should be either summarized or referenced in the safety analysis report to ensure completeness, the appropriate use of confidential information[5] and consistency with other licensing documents. The specific approach may differ for different stages of the safety analysis report. For example, including environmental aspects is relevant for the initial safety analysis report and uses information usually available from the report on the environmental impact assessment, while in subsequent safety analysis reports the radiological impact on people and the environment should be comprehensively covered by the safety analysis included in chapter 15 of the safety analysis report.

UNIFIED DESCRIPTION OF THE DESIGN OF PLANT STRUCTURES, SYSTEMS AND COMPONENTS

2.15. In general, all plant SSCs that have the potential to affect safety should be described in the safety analysis report. The type of information about each SSC to be included in the safety analysis report depends on the particular type and design of the reactor selected for construction; however, this information should be sufficient to review these SSCs in terms of their compliance with national laws and regulations. For some types of reactor, many of the sections within the chapter descriptions in Section 3 of this Safety Guide will be entirely relevant, while for other reactor types those sections may not apply directly.

2.16. Descriptions of all the SSCs important to safety should be provided, together with a demonstration of the conformance of these SSCs with the relevant design requirements. The level of detail in each description should be commensurate with the importance of the structure, system or component to safety. To help ensure consistency and completeness in the descriptions of all the SSCs important

[5] See also paras 2.29 and 3.13.29.

to safety, a common structure with a more detailed specification of the intended content is provided in Appendix II.

USE, REVIEW AND UPDATING OF THE SAFETY ANALYSIS REPORT DURING PLANT OPERATION

2.17. Use of the safety analysis report should not be limited to the licensing process and to providing public assurance about the safety of the plant prior the operation. The safety analysis report should also be continuously used by the operating organization to manage safety. It is essential that the operating organization accomplishes the safety objectives embodied in the safety analysis report by developing appropriate management for safety, including procedures and instructions. The safety analysis report serves to identify the limits and conditions for safe plant operation, which provide the basis for the development of operating procedures and instructions.

2.18. Since the safety analysis report is an essential part of the overall justification of the safety of the nuclear power plant, it should always reflect the state of knowledge of the methods for safety assessment as well as the status of the plant configuration. The safety analysis report should therefore be reviewed at appropriate intervals and should be updated accordingly. The updating of the safety analysis report should reflect, as appropriate, all safety related activities performed during the lifetime of the nuclear power plant, including the following:

(a) Hardware modifications;
(b) Findings from inspections;
(c) Procedural changes;
(d) Maintenance findings;
(e) Periodic safety reviews or alternative arrangements (see para. 2.8 of IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [11]);
(f) Analysis of operating events;
(g) Analysis of applicable experience from other nuclear power plants and other industries, as appropriate;
(h) Ageing management of the SSCs;
(i) Changes to analytical techniques, standards and criteria;
(j) Regulatory requirements.

2.19. The safety analysis report should be consistent with the plant configuration over the plant lifetime. Therefore, the safety analysis report should be updated in a timely manner to reflect plant modifications that have an impact on safety, in accordance with paras 11.2 and 11.3 of IAEA Safety Standards Series No. SSG-71, Modifications to Nuclear Power Plants [12]. It is essential that all activities that could affect the validity of the safety analysis report be clearly identified and controlled by procedures that include a requirement to review the impact of each activity. The full impact of any modification on the safety of the nuclear power plant should be evaluated and, where so required, submitted to the regulatory body for approval before being implemented.

2.20. Changes incorporated into the safety analysis report should be performed in accordance with the procedures established by the operating organization and should be easily traceable (e.g. revision number and date of issue indicated on every new or modified page); this includes those changes incorporated during the review of the safety analysis report by the regulatory body.

FORMAL ASPECTS REGARDING THE DOCUMENTATION OF THE SAFETY ANALYSIS REPORT

2.21. The safety analysis report should document the safety of the nuclear power plant with a scope and level of detail sufficient to support the conclusions reached and to provide an adequate input to the review undertaken by the regulatory body. The depth of description provided in the safety analysis report should reflect the requirement for the report to be a key reference document and for it to be sufficiently detailed to be understandable by itself.

2.22. In accordance with Requirement 5 of GSR Part 1 (Rev. 1) [1], the operating organization has the prime responsibility for safety. Consequently, if the safety analysis report is developed by a third party (e.g. by the nuclear power plant vendor), it should contain sufficiently detailed information, either in the report itself or in referenced documents, to allow for an independent verification. This verification should be conducted either by the operating organization or by another qualified organization on its behalf (see paras 4.64, 4.66 and 4.67 of GSR Part 4 (Rev. 1) [2]). Irrespective of the process followed for the development and verification of the safety analysis report, the operating organization remains responsible for the content, comprehensiveness and quality of the safety analysis report.

2.23. The information included in the safety analysis report should be presented in a clear and concise way. Each subject should be treated in sufficient depth and should be documented to permit a reviewer to independently evaluate the safety level. Tables, drawings, graphs and figures should be used wherever they contribute to the clarity and brevity of the report.

2.24. The information contained in the safety analysis report should be self-contained to a reasonable extent. Any important supporting material should be referenced in the safety analysis report. These supporting materials serve to enhance the review process and the subsequent usability of the safety analysis report, and should be easily accessible to the regulatory body, which will use the information for its review and assessment process. Use of external references (e.g. detailed design documents, references to standards, detailed analysis reports, code validation reports, source material for probabilistic safety assessment) in the safety analysis report and their extended use are inevitable. Less important external references are usually not submitted to the regulatory body with the safety analysis report, but they should be made available on request. Discussions in relation to lower level documents (e.g. operational procedures, emergency operating procedures, severe accident management guidelines; see IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [13]), as appropriate, are also useful.

2.25. A user friendly format significantly facilitates the use and review of the safety analysis report. Therefore, the safety analysis report should be made available in an electronic format, which would ideally contain cross-references and links between the various chapters and sections of the safety analysis report.

RELATIONSHIP OF THE SAFETY ANALYSIS REPORT TO OTHER LICENSING DOCUMENTS

2.26. In addition to the safety analysis report, other documents are used in the licensing process. Typical examples include reports on the environmental impact assessment, probabilistic safety assessment studies, emergency plans and decommissioning plans. In some States, information from these reports is part of the safety analysis report.

2.27. Some of the information contained in the safety analysis report might be the same as that required for other licensing documents. In such cases, the same information needs to be incorporated (to an appropriate extent) in parallel in several different documents. These documents might have been developed in response to different legislative requirements, and each of them should be essentially self-contained.

2.28. It should be ensured that there is consistency and continuity in the information provided in different licensing documents as well as in subsequent stages of the safety analysis report. If a subsequent stage of the safety analysis report indicates different results to those in the report from the previous stage (e.g. because the information has improved or modifications have been made), the incorporated changes should be explained and justified.

TREATMENT OF SENSITIVE INFORMATION

2.29. Certain parts of the safety relevant information may be of a sensitive or confidential nature. The operating organization should decide either to limit the presentation of such information in the safety analysis report or to adopt other information security measures. These measures could include limiting access to certain parts of the safety analysis report to ensure that the information that is publicly available will not contain data that could be misused (e.g. for malicious acts endangering nuclear power plant safety or nuclear security), lead to a violation of intellectual property rights, or compromise business or sensitive information. At the same time, the operating organization should ensure that measures to protect intellectual property rights or business or sensitive information do not impede a comprehensive review of the safety analysis report by the regulatory body; the regulatory body should have access to all the information necessary to perform its function. In addition to the safety analysis report used in the licensing process, consideration should be given to the preparation of a safety analysis report that does not contain any sensitive information for the purpose of communication and consultation with interested parties such as the public.

STRUCTURE OF THE SAFETY ANALYSIS REPORT FOR DIFFERENT TYPES OF NUCLEAR INSTALLATION

2.30. This Safety Guide is intended to be used for nuclear power plants. Nevertheless, some sections of this Safety Guide may be applied to other nuclear installations, such as nuclear fuel cycle facilities. In such cases, it should be taken into account that common or similar SSCs are used in different facilities under different operating conditions.

2.31. In the majority of cases, the nature and the magnitude of the risk associated with other installations is not comparable with that of a nuclear power plant. Correspondingly, the scope and content of the safety analysis report for some nuclear installations may be significantly simplified compared with the safety analysis report for a nuclear power plant. The structure and content of the safety analysis report will depend on the specific type and design of the nuclear installation concerned. This will, in turn, determine how different sections of this Safety Guide can be used in the development of the safety analysis report.

3. CONTENT AND STRUCTURE OF INDIVIDUAL CHAPTERS OF THE SAFETY ANALYSIS REPORT

CHAPTER 1: INTRODUCTION AND GENERAL CONSIDERATIONS

Introduction

3.1.1. The safety analysis report should start with an introduction that includes the following:

(a) Identification of the purpose of the nuclear power plant, making reference to the case for justification (e.g. in terms of meeting the demand for energy and the choice of the nuclear option);
(b) A statement of the main purpose of the safety analysis report;
(c) Information about the preparation process for the safety analysis report, the major contributors to the preparation (e.g. vendors), and the use of information that has been previously reviewed by the regulatory body, if applicable;
(d) A description of the structure of the safety analysis report, the objectives and scope of each of its chapters, and the connections between them;
(e) A description of the national and international guidance applied in the preparation of the safety analysis report, with justification of any deviations from this guidance.

Project implementation

3.1.2. The information provided in the project implementation section should include a description of the existing authorization status of the plant, with an indication of future project milestones, as appropriate.

Identification of interested parties regarding design, construction and operation

3.1.3. The primary contractors for the design, construction and operation of the nuclear power plant should be specified in this section. The principal consultants and external service organizations (e.g. those providing audits of the management system) should also be identified. The division of responsibilities between the designers, the owner, the constructors and the operating organization should also be described.

Information on the plant layout and other aspects

3.1.4. Drawings of the general layout of the entire plant (including multiple unit plants) should be included in the plant layout section, together with a presentation of the physical and geographical location; connections with the electricity grid; and means of access to the site by rail, road and water.

3.1.5. The main interfaces and boundaries between on-site equipment and equipment and systems external to the plant should be described. In addition, it should be clearly specified which external equipment is under the responsibility of the operating organization and which is under the responsibility of other organizations.

3.1.6. This section might also refer to sensitive information (i.e. in a separate document; see para. 2.29) on the provisions made for the nuclear security of the plant. Such information might also include a description of the steps that would be taken to provide protection in the event of a malicious act on the site or off the site.

General plant description

3.1.7. This section should provide a general description of the plant, including the overall safety philosophy, the safety concepts to be applied and a general comparison with appropriate international practices. It should enable the reader to gain an adequate general understanding of the plant without having to refer to subsequent chapters of the safety analysis report.

3.1.8. This section should briefly present (e.g. in a table) the principal elements of the plant, including the number of units, the type of reactor, the principal characteristics of the plant, the type of nuclear steam supply system, the type of nuclear fuel, the type of containment structure and associated systems, the thermal power levels in the core, the corresponding net electrical power output for each thermal power level, the type of ultimate heat sink, and any other characteristics necessary for understanding the main technological processes included in the design.

Comparison with other plant designs

3.1.9. If applicable, this section should include information about the reference plant (e.g. location and brief summary of relevant data). If the plant design is new, unique or special ('first of a kind'), the plant design should be compared with designs that have previously been authorized, so as to identify the main differences and assist in the justification of any modifications and improvements that have been made. This comparison should focus on new safety features that differ from previous designs, such as the use of redundant, diverse, simplified, inherent, passive or other innovative means of fulfilling safety functions.

Drawings and other more detailed information

3.1.10. Basic technical and schematic drawings of the main plant systems and equipment should be included in this section. The drawings should be accompanied by a brief description of the main plant systems and equipment, together with their purposes and interactions. References should be made, where necessary, to other chapters of the safety analysis report that present detailed descriptions of specific SSCs.

Modes of normal operation of the plant

3.1.11. All operating modes of the nuclear power plant should be described: startup, power operation, shutting down, shutdown (including long term shutdown), maintenance, testing, refuelling, and any other allowable modes of normal operation, including load following operation. The permissible periods of operation at different power levels in the event of a deviation from normal operating conditions should be specified.

Principles of safety management

3.1.12. This section should briefly introduce the management of safety as an integral component of the management of the operating organization. It should be confirmed that the operating organization will be able to fulfil its responsibility to operate the plant safely throughout its operating lifetime. The principles of safety management should be described.

Additional supporting or complementary documents to the safety analysis report

3.1.13. This section should provide a list and summary of the topical reports that are incorporated, by reference, as part of the safety analysis report. Typically, the results of tests and analyses (e.g. results of manufacturers' material tests and qualification data) may be submitted as separate reports.

Conformance with applicable regulations, codes and standards

3.1.14. This section should provide an overview of the relevant regulations, codes and standards that collectively represent the safety rules used in the design, including information on the use of relevant IAEA safety standards. If these regulations, codes and standards have not been prescribed by the regulatory body, a justification of their appropriateness should be provided. Any deviations from existing regulations, codes and standards should be described in this section, together with a demonstration that the deviations will not be detrimental to safety.

CHAPTER 2: SITE CHARACTERISTICS

3.2.1. Chapter 2 of the safety analysis report should provide information on the geological, seismological, volcanic, hydrological, meteorological and geotechnical characteristics of the site and the surrounding region. It should also provide information on the characteristics of external human induced hazards in conjunction with information on the radiological dispersion characteristics of the site and surrounding environment, and on the present and projected population distribution and land use relevant to the safe design and operation of the plant. Sufficient data should be included to permit an independent evaluation.

3.2.2. The information provided in chapter 2 of the safety analysis report should be periodically updated (typically every ten years), with account taken of the latest information and knowledge, to provide a basis for evaluating the safety implications of any changes.

3.2.3. Site characteristics that might affect the safety of the plant should be investigated, and the relevant results of the corresponding assessment should be included in this chapter of the safety analysis report. Relevant requirements are provided in SSR-1 [5], and relevant recommendations and guidance are provided in the following publications:

(a) IAEA Safety Standards Series No. NS-G-3.1, External Human Induced Events in Site Evaluation for Nuclear Power Plants [14];
(b) IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [15];
(c) IAEA Safety Standards Series No. NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants [16];
(d) IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [17];
(e) IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [18];
(f) IAEA Safety Standards Series No. SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations [19];
(g) IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations [20].

3.2.4. This chapter of the safety analysis report should provide information concerning the site evaluation as support for the design, design assessment and periodic safety review and should include potential changes to relevant site parameters expected over the lifetime of the plant. This information should include the following:

(a) The collection of site reference data for the plant design (e.g. geological, seismological, geotechnical, volcanic, hydrological and meteorological);
(b) The site specific hazard evaluation for external events of natural origin (e.g. earthquakes, meteorological events, flooding, geological and volcanic hazards, hazards from biological organisms, surface deformation relating to tectonic (i.e. faulting) and non-tectonic causes) and of human induced origin (e.g. aircraft crashes, chemical explosions from activities performed at nearby industrial facilities and other facilities);
(c) The design targets in terms of the probability of recurrence of external events, with account taken of their severity and associated uncertainties;
(d) An evaluation of the impact of the site related issues to be considered in the parts of the safety analysis report on emergency preparedness and response and accident management;
(e) The arrangements for the monitoring of site related parameters throughout the lifetime of the plant;
(f) The potential for specific hazards to give rise to impacts simultaneously on several units in the case of a multiple unit site.

3.2.5. A description of any considerations from the site survey stage concerning the site exclusion or acceptance criteria applied in the preliminary screening of the site for suitability should be provided in this chapter of the safety analysis report.

3.2.6. Site related information represents an important input to the design process and may be one of the sources of uncertainty in the final safety evaluation. The measures employed to take into account such uncertainties should be considered in this chapter of the safety analysis report.

Geography and demography

3.2.7. This section should specify the site location, including both the area under the control of the operating organization and the area surrounding the site in which there is a need for consultation with interested parties on the control of activities that could affect plant operation (e.g. aircraft flights, associated flight exclusion zones). This should include facilities and activities in the surrounding area that could pose a hazard to the plant (e.g. pipelines, roadways, waterways).

3.2.8. Information on activities with the potential to affect plant operation should include relevant data on the population distribution and density (including, where applicable, transient populations) and on the distribution of public and private facilities (e.g. airports, harbours, rail transport centres, pipelines, roadways, waterways, factories and other industrial sites, schools, hospitals, police services, firefighting services, municipal services) around the site.

3.2.9. This section should also cover the public uses of the land and water resources in the surrounding area and should include an assessment of any possible interaction with the plant and the implications for off-site protective actions in an emergency.

Evaluation of site specific hazards

3.2.10. This section should present the results of a detailed evaluation of natural and human induced hazards at the site that should be taken into account in the design of SSCs. The description should include due consideration of the envisaged evolution of these hazards during the expected lifetime of the nuclear power plant. SSR-1 [5] establishes requirements for the evaluation of specific external hazards.

3.2.11. The screening criteria used for each hazard (including the envelope, probability thresholds and credibility of events) and the expected impact of each hazard in terms of the originating source, the potential propagation mechanisms and the predicted effects at the site should be described in this section.

3.2.12. Hazards identified as potentially affecting the site can be screened out if they would be incapable of challenging the safety of the plant or if they are considered, with a high degree of confidence, to be extremely unlikely. The arguments in support of the screening process should be justified and described in this section of the safety analysis report.

3.2.13. The target probability levels for design against external hazards should be defined, and a comparison with the acceptable limits should be presented. Attention should be paid to the external hazards that could lead to common cause failures of the safety systems and the safety features for design extension conditions.

3.2.14. The evaluation presented in this section should also take into account unlikely natural hazards exceeding those considered for design, derived from the hazard evaluation for the site, to ensure adequate margins to avoid cliff edge effects. The reliability of the heat transfer to the ultimate heat sink should be given special attention.

3.2.15. This section should confirm that appropriate arrangements are in place to periodically update the evaluations of site specific hazards in accordance with the results of updated methods of evaluation, monitoring data and surveillance activities.

3.2.16. This section should also include results from the evaluation of potential combinations of site specific hazards that could affect the safety of the nuclear power plant.

3.2.17. Where administrative measures are employed to mitigate the adverse effects of hazards (especially for human induced events), information should be presented on their implementation, together with the roles and responsibilities for their enforcement.

Proximity of industrial, transportation and other facilities

3.2.18. This section should describe the locations and transport routes that represent potential risks for the plant and the results of a detailed evaluation of the effects of potential accidents at industrial, transportation or other facilities in the vicinity of the site. Projected developments in the vicinity over the envisaged lifetime of the nuclear power plant relating to this information should also be presented and updated, as required, in future stages of the safety analysis report.

3.2.19. Any identified risks considered in determining the design basis should be included to help determine whether any additional measures are necessary to mitigate the adverse effects of potential incidents.

Activities at the plant site that might influence the safety of the plant

3.2.20. In this section, any processes or activities at the site that, if incorrectly carried out, could affect or influence the safe operation of the plant should be presented and described. Examples of such processes or activities include vehicular transport in the plant area; storage of fuels, gases and other chemicals; and activities potentially leading to intakes of or contamination by harmful particles, smoke or gases (e.g. intakes of air through ventilation systems).

3.2.21. Measures for site protection (e.g. dams or dykes for flood control and drainage) and any modifications to the site (e.g. soil substitution, modifications to the site elevation) are usually considered at the site characterization stage, and their assessment in relation to the design basis should be included in this section of the safety analysis report.

Hydrology

3.2.22. This section of the safety analysis report should present sufficient information to enable evaluation of the potential implications of hydrological conditions at the site for the plant design and safe operation, with special attention devoted to conditions that potentially affect residual heat removal to the ultimate heat sink. Cooling water channels and reservoirs to be used for cooling the plant should be described. Low water conditions and the possibility of using groundwater sources in extraordinary situations should also be considered.

3.2.23. The conditions that should be taken into account in this section include potential floods resulting from phenomena such as abnormal ice effects and heavy rainfall, as well as runoff floods from watercourses, reservoirs, adjacent drainage areas and site drainage. This section should also include consideration of flood waves resulting from dam failures; flooding caused by landslides; ice jams and other ice related flooding; and seismically generated, water based effects on and off the site. For coastal and estuary sites, evaluations should include storm surge, tsunamis and seiches. For both coastal and riverine flooding, reasonable combinations of hazards (e.g. tides, strong wind) and potential effects of climate change should be considered.

3.2.24. The information given in this section should be prepared in such a way as to enable the assessment of (i) the transport of radionuclides in groundwater and the surface water system and (ii) the dispersion of radionuclides through the environment. This information should also include a characterization of the hydrogeological subsurface properties and surface water features to enable an assessment of the measures taken to preclude the release of radionuclides to the environment.

Meteorology

3.2.25. This section should provide a description of the meteorological aspects relevant to the site and its surrounding area, with account taken of regional and local climatic effects. Data derived from on-site meteorological monitoring or other meteorological stations should be documented.

3.2.26. This section should include information relevant to the assessment of (i) the hazards from meteorological events potentially affecting the plant and (ii) the transport of radioactive material to and from the site and the dispersion of radionuclides through the environment.

3.2.27. The extreme values of meteorological parameters or meteorological events — such as temperature; humidity; rainfall; wind speeds for straight and rotational winds, including tornadoes (owing to the sudden pressure drop that accompanies the passage of the centre of a tornado); waterspouts (owing to their potential to transfer large amounts of water to the land from nearby water bodies); dust storms; sandstorms; and snow loads and ice (see SSG-18 [18]) — should be evaluated in relation to the design, with account taken of the envisaged evolution of such extreme parameters over the lifetime of the nuclear power plant. The potential for lightning and windborne debris to affect plant safety (including the design basis missile hazard from hurricanes and tornadoes) should be considered, where appropriate.

Geology, seismology and geotechnical engineering

3.2.28. This section should provide information concerning the geological, tectonic, seismological and volcanic characteristics of the site and a sufficiently large region surrounding the site. The evaluation of seismic hazards should be based on a suitable seismotectonic model, substantiated by appropriate seismological evidence and geological or seismological data. The results of this evaluation that will be used further in other sections of the safety analysis report (including structural design and seismic qualification of components) should be described in sufficient detail. The potential for volcanic phenomena to affect plant safety should be considered, where appropriate.

3.2.29. Site reference data relating to the geotechnical properties of soil and rock underlying the site (both static and dynamic properties, including damping and modulus degradation properties) should be elaborated on in this section. Geological hazards such as slope instability, subsidence or uplift of the site surface, soil liquefaction, instability of subsurface materials, and the long term performance of subsurface materials and foundations over the lifetime of the plant should be characterized in this section. The processes for the following should be described: the collection of data for the design of foundations, the evaluation of the effects of site response and soil–structure interaction, the construction of earth structures and buried structures, the evaluation of the effects of groundwater conditions, and the evaluation of soil improvements at the site.

3.2.30. This section should present the relevant data for the site and the associated ranges of uncertainty, including the spatial variability used in the site seismic response analysis and in the structural design. Reference should be made to the technical reports that provide a detailed description of the conduct of the investigations and their planned extensions, as well as of the origin of the data collected through site surveys on a regional basis or through bibliographic surveys.

3.2.31. The design of subsurface material and buried structures, as well as site protection measures, if relevant, should also be documented. A description of projected developments relating to the information described in paras 3.2.28–3.2.30 should also be provided and should be updated as required.

Site characteristics and the potential effects of the nuclear power plant in the region

3.2.32. The characteristics of the site and the surrounding environment relevant to the dispersion of radioactive material in water, air and soil should be described in this section. The relevant requirements for evaluating the dispersion of radioactive material are established in section 6 of SSR-1 [5].

Radiological conditions due to external sources

3.2.33. This section should describe the radiological conditions in the environment at the site and in the surrounding area, with account taken of the radiological effects of other nuclear installations on the site and any other external radiation sources. The radiological conditions should be described in sufficient detail to serve as an initial reference point and a basis for future assessments of radiological conditions at the site and in the surrounding area.

3.2.34. A description should be provided of the available radiation monitoring systems and the corresponding technical means for the detection of any radiation or radioactive contamination. If appropriate, this section may reference other relevant sections of the safety analysis report concerned with the radiological aspects of licensing the plant.

Site related issues in emergency preparedness and accident management

3.2.35. The issues regarding feasibility of emergency preparedness in terms of plant accessibility and transport of any equipment necessary in an emergency, including a severe accident, should be described in this section, with account taken of all reactor units and other nuclear and non-nuclear installations on the site, as applicable. The information provided should include the availability of adequate access and egress roads for the evacuation of personnel, including access to and around the site, and supply networks in the vicinity of the site.

3.2.36. The availability of local transport networks, communications networks and other infrastructure external to the site, during and after an external event, and issues regarding the feasibility of implementing emergency response actions should be described in this section.

3.2.37. The need for any necessary administrative measures should be identified, together with the relevant roles of bodies and response organizations other than the operating organization.

Monitoring of site related parameters

3.2.38. The strategy for monitoring site related parameters and the use of the results in preventing, mitigating and forecasting the effects of site related hazards should be described in this section.

3.2.39. The provisions to monitor site related parameters affected by earthquakes and surface faulting, geological and volcanic phenomena, meteorological events, flooding, geotechnical hazards, hazards from biological organisms and human induced hazards (e.g. aircraft flight activities, chemical explosions, activities at nearby industrial and other facilities) should be described in this section. These provisions may be used for the following purposes:

(a) To provide the information necessary for operator actions taken in response to external events;
(b) To support the periodic safety review at the site;
(c) To develop models for the dispersion of radionuclides;
(d) To confirm the completeness of the set of site specific hazards taken into account.

3.2.40. This section should contain a description of the on-site meteorological monitoring programme, which can potentially be used for updating meteorological data in the future, for predicting the dispersion of radioactive substances during plant operation, or for providing early warning against extreme meteorological events. The monitoring of demographic and hydrological conditions over the lifetime of the plant should also be described in this section (see SSR-1 [5]).

3.2.41. Long term monitoring programmes should include the collection of data from site specific instrumentation and data from specialized institutions for use in comparisons to detect significant changes from the design basis, for example changes due to the possible effects of climate change.

CHAPTER 3: SAFETY OBJECTIVES AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.3.1. Chapter 3 of the safety analysis report should outline the general design concepts, requirements, codes and standards applicable for different kinds of SSC and the approach adopted to meet the safety objectives. The compliance of the actual design with all these elements should be demonstrated in more detail in other chapters of the safety analysis report, in particular in those devoted to a description of different SSCs.

General safety design basis

3.3.2. The overall safety philosophy and general approaches for ensuring safety should be presented in this section. In addition to any national requirements and associated regulatory guidance, these approaches should be based on the requirements for the design of nuclear power plants established in SSR-2/1 (Rev. 1) [3].

Safety objectives

3.3.3. This section should summarize the overall safety philosophy, safety objectives and high level principles used in the project. These should be based on the relevant safety principles set out in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [21].

Safety functions

3.3.4. This section should identify the plant specific safety functions that are necessary to fulfil the main safety functions and how their fulfilment is ensured by the plant's inherent features, in accordance with Requirement 4 of SSR-2/1 (Rev. 1) [3] and depending on the nature of the facility or activity. The corresponding SSCs necessary to fulfil those safety functions should be introduced.

3.3.5. If the main safety functions are subdivided into more detailed specific safety functions and functional criteria, with the objective of facilitating their use, they should be listed here, for example heat removal, which is considered a safety function necessary not only for the safety of the reactor core but also for the safety of any other part of the plant containing radioactive material that needs to be cooled, such as spent fuel pools and storage areas.

Radiation protection and radiological acceptance criteria

3.3.6. This section should describe in general terms the design approach adopted to meet the fundamental safety objective (see para. 2.1(a) of SF-1 [21]) and to ensure that, in all plant states, radiation doses due to any radioactive release are kept below authorized limits and as low as reasonably achievable (see also paras 2.6 and 2.7 of SSR-2/1 (Rev. 1) [3]).

3.3.7. Relevant radiological acceptance criteria for nuclear power plant workers and for the public, assigned for each plant state (normal operation, anticipated operational occurrences, design basis accidents and design extension conditions), and the consistency among the various criteria, should be introduced in this section.

General design basis and plant states considered in the design

3.3.8. The general approach to defining the design basis should be described, with account taken of operational states, accident conditions, and impacts from both external and internal hazards. The information provided should include the operational states and accident conditions under which a given structure, system or component will need to fulfil a safety function.

3.3.9. This section should describe the capability of the plant to cope with a specified range of operational states and accident conditions. Modes of normal operation of the plant should be specified. Plant states considered in the design should be listed and grouped into categories. In addition to normal operation, these categories should include anticipated operational occurrences, design basis accidents, design extension conditions without significant fuel degradation and design extension conditions with core melting.

3.3.10. The basis for the categorization of plant states (typically, frequencies or other associated characteristics) should be explained. Postulated initiating events (whether of internal origin or caused by internal and external hazards, if relevant) should be listed. This categorization should be commensurate with the content of chapter 15 of the safety analysis report.

Prevention and mitigation of accidents

3.3.11. This section should describe the measures taken to prevent and to mitigate the consequences of accidents and to ensure that the likelihood that an accident will have harmful consequences is extremely low (see paras 3.30 and 3.31 of SF-1 [21]).

Defence in depth

3.3.12. This section should describe the approach adopted to incorporate the defence in depth concept into the design of the plant. It should be demonstrated that the defence in depth concept has been applied at all stages of the lifetime of the nuclear power plant, for all plant states and for all safety related activities, in accordance with paras 2.12–2.18 of SSR-2/1 (Rev. 1) [3]. It should also be demonstrated that measures have been taken for adequate robustness and independence of levels. Particular emphasis should be placed on describing how the independence of safety systems and safety features for design extension conditions with core melting is approached.

3.3.13. It should be demonstrated that there are physical barriers to the release of radioactive material and systems to protect the integrity of the barriers and that measures are taken to ensure the robustness of these provisions at each level of defence in depth.

3.3.14. Where appropriate, any operator actions envisaged to be necessary to mitigate the consequences of an event and to assist in the fulfilment of the safety functions essential for defence in depth should be described.

3.3.15. Where appropriate, any off-site support envisaged to be necessary should be described.

Application of general design requirements and technical acceptance criteria

3.3.16. This section should include a high level description of the deterministic design principles. Where aspects of the design are based on conservative deterministic principles (such as those embodied in international standards, internationally recognized industrial codes and standards, and regulatory guides), the use of such design approaches should be elaborated in this section of the safety analysis report, with reference made to the specific applicable codes and standards.

3.3.17. The scope of implementation of the single failure criterion and how compliance with this criterion is achieved in the design should be described in this section of the safety analysis report. This section should also include results from the consideration of the possibility of a single failure occurring while a redundant train of a system is undergoing maintenance or is impaired by internal or external hazards.

3.3.18. The provisions to comply with Requirements 21 and 23–26 of SSR-2/1 (Rev. 1) [3] for protection against common cause failures should also be described in this section of the safety analysis report.

3.3.19. Any other relevant approaches aimed at ensuring safety should be specified in this section. Such approaches typically include the following, as applicable:

(a) Simplification of the design;
(b) Passive safety features;
(c) Gradually responding plant systems;
(d) Fault tolerant plant and systems;

(e) Operator friendly systems;
(f) Equipment that employs the 'leak before break' concept.

3.3.20. Any specific technical acceptance criteria used in the design that are associated with the integrity of individual barriers against the release of radioactive material should be listed here. If probabilistic safety objectives or criteria have been used in the design process, these should also be specified in this section.

Practical elimination of the possibility of event sequences arising that could lead to an early radioactive release or a large radioactive release

3.3.21. This section should describe the approach used to identify the conditions that could lead to an early radioactive release or to a large radioactive release and should summarize the design and operational provisions implemented to ensure that the possibility of such conditions arising has been 'practically eliminated'[6] (see para. 5.31 of SSR-2/1 (Rev. 1) [3]).

3.3.22. In this section, reference should also be made, as appropriate, to other sections of the safety analysis report where relevant confirmatory analyses are presented (e.g. chapter 15 of the safety analysis report; see paras 3.15.1–3.15.68).

Safety margins and avoidance of cliff edge effects

3.3.23. This section should summarize the approach taken to ensure adequate margins to prevent cliff edge effects relating to damage to barriers against releases of radioactive material to the environment (see para. 5.73 of SSR-2/1 (Rev. 1) [3]).

3.3.24. This section should specifically describe the approach and assumptions for deterministic safety analyses (conservative or realistic), selected to demonstrate adequate safety margins, including the use of sensitivity studies to demonstrate the avoidance of cliff edge effects in the analyses applicable for design extension conditions.

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[6] Footnote 16 of SSR-2/1 (Rev. 1) [3] states: "The possibility of certain conditions arising may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise."

3.3.25. The section should also describe the approach used to demonstrate safety margins for internal or external hazards. For natural hazards, it should be described how adequate safety margins are ensured for hazards that exceed those considered in the design (see para. 5.21A of SSR-2/1 (Rev. 1) [3]).

Design approaches for the reactor core and for fuel storage

3.3.26. This section should describe the design approaches adopted to demonstrate the performance of the safety functions in the reactor and in the fuel storage areas, in particular in the spent fuel pool. These design approaches could imply differences in implementation of defence in depth, different specification of derived safety functions, different monitoring means and substantial differences in the time evolution of accidents. In accordance with Requirement 4 of SSR-2/1 (Rev. 1) [3], shielding of the irradiated fuel elements is required. More detailed descriptions of design provisions should be included in the relevant sections of chapters 4 and 9 of the safety analysis report (see paras 3.4.1–3.4.10 and 3.9.1–3.9.24); information to be provided regarding the evolution of accidents and the availability of sufficient margins should be included in chapter 15 of the safety analysis report (see paras 3.15.1–3.15.68). Further recommendations regarding fuel storage are provided in IAEA Standards Series No. SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [22].

Considerations of interactions between multiple units

3.3.27. For multiple unit sites, this section should describe any sharing of systems among the units as well as any interconnections among the units. It should be confirmed that Requirement 33 of SSR-2/1 (Rev. 1) [3] is met.

3.3.28. Any interconnections between units to further enhance safety should be explicitly described in this section, and the positive and negative effects of such interconnections should be explained.

3.3.29. A description should be provided of any interconnections or services provided by shared systems that will be severed when one or more units are shut down for an extended period and kept in a safe storage state (e.g. in preparation for future decommissioning). In addition, the results of analyses that consider the impact on other operating units of severing the interconnections and shared services should be provided.

Design provisions for ageing management

3.3.30. This section of the safety analysis report should define the design life of items important to safety and should describe how relevant mechanisms of ageing and wear were taken into account in the design of the nuclear power plant to ensure the adequate performance of the most important plant components. Special attention should be devoted to the reactor pressure vessel, in particular to the effects of neutron embrittlement.

3.3.31. It should be described how adequate margins are maintained, with account taken of degradation mechanisms relevant to ageing, including those caused by testing and maintenance, by plant states during a postulated initiating event and by plant states following a postulated initiating event.

3.3.32. It should be described how ageing effects caused by environmental factors (e.g. vibration, irradiation, humidity, temperature) over the expected service life of items important to safety have been considered in the qualification programme for such items. Reference should be made to a comprehensive ageing management programme (see paras 3.13.1–3.13.30).

Classification of structures, systems and components

3.3.33. This section of the safety analysis report should provide information on the approach adopted for the categorization of safety functions, for the identification of the SSCs necessary to fulfil these safety functions and for the safety classification of these items (see Requirement 22 of SSR-2/1 (Rev. 1) [3] and IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [23]). The information should include details of the following:

(a) The methodology and criteria applied for safety classification;
(b) The categorization of the safety functions;
(c) The safety classification of the SSCs;
(d) The associated engineering, design (e.g. environmental qualification, seismic categorization) and manufacturing rules for different safety classes of SSCs;
(e) The verification of the classification.

3.3.34. If there is a potential for structures or systems to interact, then details should be provided of the way in which it has been ensured in the design that a plant provision of a lower class or category cannot unduly impair the role of plant provisions with a higher classification.

3.3.35. A list of the main SSCs important to safety, together with their related safety functions, safety classification, seismic categorization and associated safety requirements, should be included either in an annex to, or as a reference in, the safety analysis report.

Protection against external hazards

3.3.36. An indicative list of external hazards to be considered should be provided in chapter 2 of the safety analysis report. This section of chapter 3 should provide a list of the external hazards specifically considered in the design. It should also describe the quantitative design parameters of individual hazards, relevant design criteria, codes and standards, methods of assessment, and the general design measures to ensure that the SSCs important to safety are adequately protected against the detrimental effects of the hazards considered in the plant design.

3.3.37. Hazards of natural origin and human induced hazards relevant to the given site should be described (see IAEA Safety Standards Series Nos SSG-67, Seismic Design for Nuclear Installations [24], and SSG-68, Design of Nuclear Installations against External Events Excluding Earthquakes [25]). As stated in para. 5.15B of SSR-2/1 (Rev. 1) [3]: "For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously."

3.3.38. As stated in para. 5.17 of SSR-2/1 (Rev. 1) [3]: "Causation and likelihood shall be considered in postulating potential hazards." Combinations of events and failures, such as induced effects caused by primary external hazards, for example flooding following an earthquake, are also required to be considered (see para. 5.32 of SSR-2/1 (Rev. 1) [3]). More generally, combinations of various types of load, including loads from randomly occurring individual events, should be considered and described here.

3.3.39. A detailed description of possible off-site protective actions and any human interactions necessary to mitigate the impact of external hazards should be provided in chapter 13 of the safety analysis report. At the same time, the demonstration that there is adequate protection against the design basis hazard for each case should be provided in the applicable chapter of the safety analysis report.

3.3.40. General information concerning the different hazards taken into consideration in the design should be presented in this section. The detailed design information, including calculation and test results, should be included in chapters 4–12 of the safety analysis report.

Seismic design

3.3.41. The seismic design characteristics and specific design requirements applicable for the design of SSCs, including codes, standards, methodologies and basic assumptions, to be taken into account should be presented in this section (see SSR-2/1 (Rev. 1) [3]). A description of the design solutions for SSCs to ensure compliance with the requirements should be provided in chapters 4–12 of the safety analysis report. The information provided should include the following:

(a) Seismic design parameters;
(b) Design ground motion (including levels SL-1 and SL-2);
(c) The applicable seismic system analysis;
(d) Seismic analysis methods;
(e) The procedures used for analytical modelling;
(f) The interaction of structures with different safety classifications;
(g) Seismic instrumentation;
(h) Arrangements for control room operator notification.

Extreme weather conditions

3.3.42. This section should present the design basis weather conditions for the extreme meteorological hazards (as identified in chapter 2 of the safety analysis report), the codes and standards applicable for the design, the methodologies with basic assumptions, and any other specific design criteria regarding loads and load combinations that need to be taken into account. A description of the design measures for ensuring compliance with the safety objectives and the design requirements should be provided in chapters 4–12 of the safety analysis report.

Extreme hydrological conditions

3.3.43. This section should present the design basis external flooding or low water level conditions and hazards, as identified in chapter 2 of the safety analysis report. This section should also describe the codes and standards applicable for the design, the methodologies and basic assumptions used, and any other specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring compliance with the safety objectives and the requirements should be provided in chapters 4–12 of the safety analysis report.

3.3.44. This section should also describe the methods and procedures by which the static and dynamic effects of the design basis flood conditions identified in chapter 2 of the safety analysis report are applied to structures that are designated as providing protection against external flooding.

Aircraft crash

3.3.45. This section should specify and describe all the SSCs that are necessary to perform the functions required to attain and maintain a safe shutdown condition, or to mitigate the consequences, in the event of an aircraft crash. It should define the design basis aircraft crash characteristics, as described in chapter 2 of the safety analysis report, as well as the applicable design codes and standards, the assumptions, and any specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring the required safety performance and for demonstrating compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Missiles

3.3.46. The level of protection against all external missiles (other than aircraft) identified in chapter 2 of the safety analysis report should be included in this section of the safety analysis report. This section should specify the design basis missile hazard, provide the design basis missile data, identify the codes and standards used for the design of protective measures, and describe the methodologies and basic assumptions used as well as any specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring the required safety performance and demonstration of compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

External fires, explosions and toxic gases

3.3.47. This section should describe the protection against external fires, explosions and toxic gases originating from other industrial and transportation activities. The design basis external fire, explosion and toxic gas hazards identified in chapter 2 of the safety analysis report should be described, including the codes and standards applicable for the design, the methodologies and basic assumptions used, and any specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring the required safety performance and demonstration of compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Other external hazards

3.3.48. This section should describe the protection against any other external hazards considered in the design, covering each in a separate subsection. The design basis hazards should be described, including the codes and standards applicable for the design, the methodologies and basic assumptions used, and any specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring the required safety performance and demonstration of compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Protection against internal hazards

3.3.49. This section should provide a list of the internal hazards considered in the design. This section should also include a description of the quantitative design parameters of individual hazards; relevant design criteria, codes and standards; methods of assessment; and the general design measures provided to ensure that the essential SSCs important to safety are adequately protected against the detrimental effects of all the hazards considered in the plant design to ensure safe shutdown of the plant. Design requirements for internal hazards are established in para. 5.16 of SSR-2/1 (Rev. 1) [3], and further recommendations and guidance are provided in IAEA Safety Standards Series No. SSG-64, Protection against Internal Hazards in the Design of Nuclear Power Plants [26]. The list of internal hazards should include the following:

(a) Internal fires and explosions;
(b) Heavy load drops;
(c) Internal flooding;
(d) Pipe whip following pipe ruptures and dynamic effects associated with high energy pipe ruptures;
(e) Internal missiles, such as those originating from rotating structures;
(f) Failures of pressurized components, supports or other structures.

3.3.50. As noted in para. 3.3.38, consideration is required to be given to combinations of internal hazards (e.g. flooding due to an internal missile) or plausible combinations of external and internal hazards.

Internal fires, explosions and toxic gases

3.3.51. This section should summarize the protection against internal fires, explosions and toxic gases originating from on-site activities and technological failures. The design parameters, the loads and their potential effects, the protection measures, and the required human interactions should be specified and described, together with a demonstration that these provide adequate protection. A full description of, and justification for, the relevant countermeasures should be provided in chapter 9A of the safety analysis report. A description of design measures for ensuring the required safety level and compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Internal flooding

3.3.52. This section should summarize the protection against internal floods. The design requirements, the resulting loads and their implications, and the required human interactions should be specified and described, together with a demonstration that these measures provide adequate protection. This includes the identification of all potential flooding mechanisms as well as the protection and drainage measures necessary in relation to the particular structure, system or component. An analysis of the damage to SSCs should be included in this section. A description of design measures for ensuring the required safety level and compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Internal missiles

3.3.53. This section should describe provisions for the protection against internal missiles. The design requirements, the loads and their implications, and the required human interactions should be specified and described, together with a demonstration that these measures provide adequate protection. This includes the identification of all potential missile generating events, as well as the parameters of generated missiles, including turbine missiles and any other missiles inside or outside the containment. A description of design measures for ensuring the required safety level and compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

High energy line breaks

3.3.54. This section should describe the provisions for protection against high energy line breaks. The design requirements, the loads and their implications, and the required human interactions should be specified and described, together with a demonstration that these measures provide adequate protection. This includes the identification of all postulated failures of high energy pipelines, the dynamic effects of each pipe break, and the SSCs potentially affected. A description of design measures for ensuring the required safety level and compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

Other internal hazards

3.3.55. This section should describe the provisions for protection against any other internal hazards considered in the design, each covered in a separate subsection. The design basis hazards should be described, including the codes and standards applicable for the design, the methodologies and basic assumptions used, and any other specific design criteria regarding loads and load combinations that are taken into account. A description of design measures for ensuring the required safety level and compliance with the requirements should be provided in chapters 4–12 of the safety analysis report.

General design aspects for civil engineering works of safety classified buildings and civil engineering structures

3.3.56. This section of the safety analysis report should present relevant information on the design approaches to the civil engineering of buildings and structures, including their foundations. It should also briefly introduce the way in which margins have been provided for the construction of buildings and structures that are relevant to safety, including the seismic resistance of buildings and structures. Specific information on compliance with the design rules for civil engineering works and structures should be included in chapter 9B of the safety analysis report.

3.3.57. General information on civil engineering works and structures should be provided and should include the following items:

(a) Applicable codes, standards and other specifications;
(b) Loads and load combinations;
(c) Design and analysis procedures;
(d) Structural acceptance criteria;
(e) Materials, quality control and special construction techniques;
(f) Testing and in-service inspection requirements.

3.3.58. In addition to general design principles for structural and civil engineering, more specific information should be provided on the foundations, buried structures, buildings and civil structures. This section should focus on information relating to the foundations.

3.3.59. This section should specify the requirements for the containment building itself, including leaktightness, mechanical strength, pressure resistance and resistance to hazards. Specific information should be provided for concrete containments and for the steel and concrete internal structures of the containment. The major structures to be addressed should include the following:

(a) The reactor support system;
(b) The steam generator support system;
(c) The reactor coolant pump support system;
(d) The primary shield wall and the secondary shield walls of the reactor cavity;
(e) Other major internal structures, such as supports, refuelling cavity walls, the in-containment refuelling water storage tank, and the spent fuel intermediate storage pool, as well as the operating floor, intermediate floors and various platforms.

Detailed descriptions of the structures, including the general layout, sections and principal features of major internal structures, should be provided in chapter 9B of the safety analysis report.

3.3.60. The general information to be provided for the safety classified buildings, civil engineering structures, containment and containment internal structures listed should include the following:

(a) Applicable codes, standards and specifications;
(b) Loads and load combinations;
(c) Structural acceptance criteria;
(d) Testing and in-service inspection requirements;
(e) Treatment of design extension conditions, as appropriate.

3.3.61. Other buildings for which the design rules should be described include the following:

(a) Auxiliary buildings;
(b) The building containing the safety systems;
(c) The fuel storage building;

(d) Buildings with control locations (i.e. control room, supplementary control room, and other emergency response facilities and locations);
(e) Diesel generator buildings.

General design aspects for mechanical systems and components

3.3.62. Relevant information on the design principles and criteria and the codes and standards used in the design of mechanical components, and information on their physical separation, should be included in this section. Information should also be provided concerning the design loads and load combinations, specifying the appropriate design and service limits for components and supports.

3.3.63. The methods, assumptions, computer programs and experimental verification used in dynamic and static analyses to determine the structural and functional integrity of mechanical components, including a demonstration of their adequacy, should be presented. Information concerning the operational transients considered in the design and the resulting loads and load combinations, specifying the appropriate design and service limits for classified components and supports, should also be presented.

3.3.64. A complete list should be presented of the operational transients considered in the design and the fatigue and fracture analysis of all components of the reactor coolant system and the core support components, other supporting components, reactor internals and other systems that fulfil a safety function. The list should include the number of events for each transient; the number of load and stress cycles per event and for events in combination; and the number of transients assumed for the design life of the plant. This section should also describe the environmental conditions to which items important to safety will be exposed over the design life of the plant (e.g. coolant water chemistry).

3.3.65. This section should describe the requirements for ensuring the structural integrity of pressure-retaining components with their component supports and core support structures. This description should also incorporate information relating to component design and should include current design information and representative (i.e. bounding) information. Design information should also be given for components that are not themselves important to safety but are located in the vicinity of items important to safety. This information should be sufficient to demonstrate that the failure of these components would not adversely affect the function of the nearby items important to safety.

3.3.66. This section should describe the approach to and engineering design rules for the design and analyses of the piping system, including piping components and associated supports. The description should cover the criteria and procedures used in preparing the design specifications of the piping system, including load combinations, design data and other design inputs. Specific information on the design of piping from particular systems should be included in chapters 5, 6 and 9A of the safety analysis report.

General design aspects for instrumentation and control systems and components

3.3.67. Relevant information on the design principles and criteria and the codes and standards used in the design of instrumentation and control systems and components should be included in this section. Information should be provided regarding the following:

(a) The design basis;
(b) Performance;
(c) Reliability;
(d) Independence of provisions for the different plant states;
(e) Equipment qualification;
(f) Verification and validation;
(g) Application of the single failure criterion;
(h) Access to equipment;
(i) General information on the design principles applied with respect to nuclear security, including identification of the interfaces with safety;[7]
(j) Quality;
(k) Testing and testability;
(l) Maintainability;
(m) Identification of items important to safety;
(n) Common cause failure criteria.

3.3.68. This section should describe the design basis, identifying functional and non-functional requirements, including functions, conditions and criteria for the overall instrumentation and control and for each individual instrumentation and control system. The description should indicate how this information is used to categorize the functions and to assign them to systems of the appropriate safety class in accordance with SSG-30 [23].

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[7] This information will be used in accordance with national regulations and is typically set out in a separate document that contains sensitive information.

General design aspects for electrical systems and components

3.3.69. Relevant information on the design principles and criteria and the codes and standards used in the design of electrical systems and components should be included in this section. Information should be provided regarding the following:

(a) The design basis;
(b) Redundancy;
(c) Independence;
(d) Diversity;
(e) Controls and monitoring;
(f) Identification;
(g) Capacity and capability of systems for different plant states;
(h) External grid and related issues;
(i) Power quality.

3.3.70. This section should describe the design basis, identifying functional and non-functional requirements, including functions, conditions and criteria for the overall instrumentation and control and for each individual instrumentation and control system. The description should indicate how this information is used to categorize the functions and to assign them to systems of the appropriate safety class in accordance with SSG-30 [23].

Equipment qualification

3.3.71. The safety requirements for the qualification of items important to safety are established in Requirement 30 of SSR-2/1 (Rev. 1) [3]. This section should describe the scope of the qualification programme and the qualification procedures adopted to confirm that the plant items important to safety, including safety features for design extension conditions, are capable of meeting the design requirements and of remaining fit for purpose in the range of individual or combined environmental challenges identified for the situations under which they are expected to perform. The identified challenges should take into account all the stages and their duration in the lifetime of the plant.

3.3.72. This section should set out the way in which the equipment qualification programme takes account of all environmental conditions of the plant identified as being relevant and potentially disruptive, and other potentially disruptive influences, under which the SSCs are performing, including events associated with internal and external hazards. If acceptance criteria are used for the qualification of plant items by testing or analysis, these criteria should be described here.

3.3.73. This section should include information on the methods used to ensure that SSCs are suitable for their design duty and remain fit for purpose and continue to fulfil any required safety function claimed in the design justification (in particular those functions claimed in the safety analyses and presented in the corresponding chapter of the safety analysis report).

3.3.74. This section should describe the criteria used for qualification, including the following:

(a) The decision criteria for selecting a particular test or method of analysis;
(b) The considerations involved in defining conditions resulting from the applicable plant conditions, from post-accident environmental conditions, and from seismic and other relevant dynamic load input motion;
(c) The process used to demonstrate the adequacy of the qualification programme.

The criteria for electromagnetic qualification should also be presented, including the decision criteria for selecting a particular test or method of analysis, the considerations defining the electromagnetic impact, and the process for demonstrating the adequacy of the electromagnetic qualification programme.

3.3.75. A list of items important to safety, together with their qualification requirements and, once available, confirmation of their qualification, should be established and provided or referenced in this section of the safety analysis report.

In-service monitoring, tests, maintenance and inspections

3.3.76. This section should provide an overview of the regulations, codes and standards applicable to the areas of in-service monitoring, tests, maintenance and inspections. Specific design rules for each of the areas listed should be provided.

Compliance with national and international standards

3.3.77. This section should include a statement of the conformance of the plant design with the design principles and criteria established in national regulations and international standards, which themselves will allow compliance with the safety objectives adopted for the plant.

CHAPTER 4: REACTOR

3.4.1. This chapter of the safety analysis report should provide relevant information on the reactor to demonstrate its capability to fulfil relevant safety functions throughout the design life in all plant states. The reactor pressure vessel as a part of the reactor coolant system pressure boundary should be described separately in chapter 5 of the safety analysis report. The contents of chapter 4 of the safety analysis report should demonstrate compliance with Requirements 43–46 of SSR-2/1 (Rev. 1) [3]. Recommendations on meeting the safety requirements applicable to this chapter of the safety analysis report are provided in IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants [27]; the information included in this chapter should take account of those recommendations, as applicable.

Summary description

3.4.2. A summary description[8] should be provided of the mechanical, neutronic and thermohydraulic behaviour of the various reactor components, including the fuel, the reactor vessel internals, the reactivity control systems, and related instrumentation and control systems.

3.4.3. For each of the reactor components, a more detailed description should be provided, in accordance with Appendix II.

Fuel design

3.4.4. A description should be provided of the main elements of the fuel[9] (with account taken of Appendix II, as applicable), together with a justification of the selected design bases. The justification of the design bases of the fuel should include a description of the design limits for the fuel and the functional characteristics in terms of the desired performance under all plant states.

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[8] For this chapter and for other chapters of the safety analysis report, Appendix II provides guidance on describing the design of the nuclear power plant SSCs.

[9] In this Safety Guide, the term ‘fuel’ means arrays (assemblies or bundles) of fuel rods, including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters and fill gas; burnable poison rods, including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods.

Nuclear design

3.4.5. The following information should be provided in this section:

(a) The nuclear design bases, including nuclear design limits and reactivity control limits, such as limits on excess reactivity, fuel burnup, reactivity coefficients, neutron flux distribution, power distribution control and reactivity insertion rates;
(b) The nuclear characteristics of the lattice, including core physics parameters, fuel enrichment distributions in ^{235}U (and plutonium vector contents, if applicable), distribution and concentrations of burnable poison rods, burnup distribution, boron reactivity coefficient and boron concentrations, type of control rods and their locations, shutdown margin specification, and refuelling schemes;
(c) The analytical tools, methods and computer codes (together with information on code verification and validation, including uncertainties) used to calculate the neutronic characteristics of the core, including reactivity control characteristics;
(d) The additional nuclear safety parameters of the reactor core, such as radial and axial power peaking factors and the maximum linear heat generation rate;
(e) The neutronic stability of the core, including xenon stability, throughout an operating cycle, with consideration given to possible anomalies in the different modes of normal operation covered by the design basis;
(f) Special core configurations, such as a mixed core or mixed modes of normal operation.

Thermohydraulic design

3.4.6. This section should provide the following information:

(a) The thermohydraulic design bases for the reactor core and attendant structures, and the interface requirements for the thermohydraulic design of the reactor coolant system;
(b) The analytical tools, methods and computer codes (including their verification and validation, together with consideration of the uncertainties) used to calculate thermohydraulic parameters;
(c) Flow, pressure and temperature distributions, with specification of the limiting values and their comparison with the design limits;
(d) A demonstration of the thermohydraulic stability of the core.

Design of the reactor control, shutdown and monitoring systems

3.4.7. The reactor control, shutdown and monitoring systems should be described in this section of the safety analysis report. It should be demonstrated that these systems, including any essential auxiliary equipment and hydraulic systems, are designed and installed to provide the required functional performance and are properly isolated from other equipment. In addition, the design limits and the design evaluation of the reactor control, shutdown and monitoring systems should be described.

Evaluation of the combined performance of reactivity control systems

3.4.8. This section should describe the relevant situations in which two or more reactivity control systems are used during accidents and should provide an evaluation of the combined functional performance.

3.4.9. This section should also include failure analyses that demonstrate that the reactivity control systems are not susceptible to common cause failures. These analyses should consider failures originating within any of the reactivity control systems as well as those originating from other plant equipment and should be accompanied by comprehensive and logical supporting discussions.

Core components

3.4.10. This section of the safety analysis report should provide descriptions of the following:

(a) The systems of core components, defined as the general external details of the fuel, the structures into which the fuel has been assembled (e.g. fuel rods assembled into a fuel assembly or fuel bundle), related components necessary for fuel positioning and all supporting elements internal to the reactor, including any separate provisions for moderation and fuel location. Reference should be made to the other sections of the safety analysis report that cover related aspects of the reactor core as well as fuel handling and storage.

(b) The physical and chemical properties of the materials used for the core components, including the neutronic, thermohydraulic, structural and mechanical characteristics of the components.

(c) The expected response of core components to static and dynamic mechanical loads and the behaviour of these components with respect to design limits, together with a description of the effects of irradiation and corrosion on the ability of the core components to fulfil their safety functions adequately over the lifetime of the plant.

(d) Any significant subsystem component, including any separate provision for moderation and fuel location, with corresponding design drawings.

(e) The conclusions from a consideration of the effects of in-service maintenance programmes on the fulfilment of safety functions, including surveillance and inspection programmes to monitor the effects of irradiation and ageing on the core components.

CHAPTER 5: REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS

3.5.1. Chapter 5 of the safety analysis report should provide relevant information on the reactor coolant system and its associated systems, where possible in accordance with the scope and format described in Appendix II. The contents of this chapter should demonstrate compliance with Requirements 21, 23, 26 and 47–50 of SSR-2/1 (Rev. 1) [3]. Recommendations and guidance on the design of these systems are provided in IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [28].

3.5.2. Sufficient information should be provided to demonstrate that the reactor coolant system and associated systems will retain their required level of structural integrity in operational states and accident conditions (for those SSCs not directly affected by the accident). Information on the integrity of the reactor coolant pressure boundary should include the results of the detailed stress evaluations and of studies of engineering mechanics and fracture mechanics of all the components of the reactor coolant pressure boundary that are subjected to operational states (including shutdown operating modes) and postulated accident loads.

Summary description

3.5.3. This section should provide a summary description of the reactor coolant system and associated systems and their various components. It should indicate the independent and interrelated performance and safety functions of each component and should include an overview of important design and performance characteristics.

3.5.4. A list of all components of the reactor coolant system and associated systems should be provided, together with the corresponding applicable design codes. The specific detailed stress analyses for each of the major components should be directly referenced so as to enable further evaluations to be made, if necessary.

3.5.5. This section should contain a description of, and a justification for, the design features that have been implemented to ensure that the performance of the various components of the reactor coolant system and of the subsystems interfacing with the reactor coolant system meets the safety requirements for design. The description should include the reactor coolant piping or ducting, the main steam line isolation system, the isolation cooling system of the reactor core, the main steam line and feedwater piping, the pressurizer relief discharge system, and the residual heat removal system, including all components (e.g. pumps, valves, supports). For pressurized water reactors, this should also include the reactor coolant pumps, the steam generators and the pressurizer. For boiling water reactors, this should include the recirculation pumps and the boilers.

3.5.6. A schematic flow diagram of the reactor coolant system and associated systems denoting all major components, principal pressures, temperatures, flow rates and coolant volume under normal steady state, full power operating conditions should be provided. An elevation drawing of the piping and instrumentation of the reactor coolant system and associated systems showing the principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures should also be provided.

Materials

3.5.7. A justification of the choice of materials used for the components of the reactor coolant system and associated systems should be provided, specifically for those forming the primary pressure boundary. The information provided should describe the corresponding material specifications, including the following:

(a) Chemical, physical and mechanical properties;
(b) Resistance to corrosion;
(c) Consideration of the effects of irradiation (e.g. in terms of waste management and potential for occupational exposure);
(d) Dimensional stability, strength, toughness, crack tolerance and hardness.

3.5.8. The properties and required performance of seals, gaskets and fasteners in the pressure boundary should also be described. This section should address applicable degradation mechanisms and fabrication challenges, including stress corrosion cracking and sensitization of welds; it should describe the precautions implemented to protect against such degradation mechanisms and fabrication challenges, and the analysis performed to justify the adequacy of the chosen materials and processes.

Reactor coolant system and reactor coolant pressure boundary

3.5.9. This section should describe the measures implemented to ensure the integrity of the reactor coolant system throughout the lifetime of the plant, including those measures taken to prevent cold overpressurization. In addition, this section should provide information on the means of overpressure protection of the reactor coolant pressure boundary, including all pressure-relieving devices (e.g. isolation, safety and relief valves). The provisions for coolant leakage detection should also be described.

3.5.10. This section should also provide a description of the scope of application of the leak before break concept, or the break preclusion concept, and its implementation in the piping of the reactor coolant system. The description should include the means of monitoring and an analytical demonstration of what is necessary to ensure limitation of the break size in the reactor coolant system. This section should also describe the implications of these concepts for the design of other systems or components (e.g. reactor internals) and for the scope of the postulated initiating events covered by the safety analysis provided in chapter 15 of the safety analysis report.

Reactor vessel

3.5.11. The reactor vessel design should be described in this section in sufficient detail to demonstrate that all the materials, fabrication methods, inspection techniques and load combinations conform to applicable regulations and to industrial codes and standards. The design information should include the reactor vessel materials, the pressure–temperature limits and the integrity of the reactor vessel, including considerations of the effects of embrittlement. Information on the neutron flux distribution and expected neutron fluence on the walls of the reactor pressure vessel, derived from the core characteristics, should be included (see paras 3.4.5 and 3.4.10).

3.5.12. Information should also be provided on the provisions to ensure the protection of the reactor vessel against seismic loads and surrounding environmental conditions, including the effects of pressurized thermal shocks and the behaviour of reactor vessel penetrations.

Reactor coolant pumps or recirculation pumps

3.5.13. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the reactor coolant pumps (pressurized water reactors) or recirculation pumps (boiling water reactors) meets the safety requirements for design. The description should provide information on the hydraulic parameters that ensure adequate cooling of the fuel and adequate flow coastdown characteristics of the pumps in the event of a pump trip so as to avoid undesirable thermohydraulic conditions. The information should present the provisions made to preclude rotor overspeeding and to address cavitation and possible vibration of the reactor coolant pumps and associated structures in the event of a design basis loss of coolant accident. The description should also address the performance of pump seals, including their performance under prolonged station blackout conditions. The evaluation of pump and motor lubrication system failures (e.g. leaks of lubricant, loss of cooling) should also be included to help prevent the sticking of bearings in pumps and motors.

Primary heat exchangers (steam generators) in pressurized water reactors

3.5.14. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the steam generators meets the safety requirements for design. The description should include the internal structures of the steam generators and connections to feedwater and steam exits and drains, as well as access points for inspection and leak detection.

3.5.15. The description should provide information on the design limits for water chemistry, for the concentration of impurities and for levels of radioactive material in the secondary side of the steam generators during normal operation.

3.5.16. The potential effects of damage to the heat exchange tubes, and the design criteria to prevent this from occurring, should be specified, including the following:

(a) The operational states considered in the design of the steam generator tubes, and the accident conditions selected, together with the justification for this selection, to define the allowable stress intensity limits;
(b) The extent of tube wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined in para. 3.5.16(a) above, under the postulated condition of a design basis pipe break in the reactor

coolant pressure boundary or a break in the secondary piping during reactor operation.

Reactor coolant piping

3.5.17. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the reactor coolant piping meets the safety requirements for design. The description should include the design, fabrication and operational provisions to control those factors that contribute to stress corrosion cracking.

Reactor pressure control system

3.5.18. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the reactor pressure control system meets the safety requirements for design. In addition to the pressurizer systems (i.e. pressurizer heaters and sprays in pressurized water reactors), these design features should include the depressurization systems, such as the pressure relief tank or pool (in pressurized water reactors) or wet well (in boiling water reactors); the pressure relief and safety valves; and associated piping.

3.5.19. A description of the reactor depressurization systems used for design basis accidents and those used for design extension conditions should be provided, including a clear demonstration of the independence of the levels of defence in depth that reflects the relevance of these systems.

Reactor coolant system component supports and restraints

3.5.20. This section should provide a description of, and a justification for, the design features that have been implemented to ensure the adequacy and integrity of supports and restraints.

Reactor coolant system and connected system valves

3.5.21. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the valves interfacing with the reactor coolant system meets the safety requirements for design. This description should include safety and relief valves, valve discharge lines, and any associated equipment.

Access and equipment requirements for in-service inspection and maintenance

3.5.22. In this section, information should be provided on the system boundary that is subject to inspection. In particular, components and associated supports should be described, including all pressure vessels, piping, pumps, valves and bolting, with regard to the following:

(a) Accessibility, including radiation protection aspects, working conditions (e.g. temperature, hygrometry) and systems operability;
(b) Examination categories and methods;
(c) Inspection intervals;
(d) Provisions for evaluating the results of examinations, including evaluation methods for detected flaws and repair procedures for components that reveal defects;
(e) System pressure tests.

The programmes for in-service inspection and maintenance and their implementation milestones should be described, and reference should be made to any applicable standards.

Reactor auxiliary systems

3.5.23. This section should provide a description of, and a justification for, the design features that have been implemented to ensure that the performance of the various connected or associated systems interfacing with the reactor coolant system meets the safety requirements for design. The systems described in this section should be selected so as to avoid repetition of the information in other chapters of the safety analysis report, in particular in chapters 6, 9 and 10.

3.5.24. The associated systems that should be covered in this section include the following:

(a) The chemical and inventory control systems for the reactor coolant;
(b) The reactor coolant cleanup system;
(c) The residual heat removal system;
(d) The high point vents of the reactor coolant system;
(e) The heavy water collection system for pressurized heavy water reactors;
(f) The moderator system and its cooling system for pressurized heavy water reactors;

(g) The reactor core isolation cooling system for boiling water reactors;
(h) The isolation condenser system for boiling water reactors.

CHAPTER 6: ENGINEERED SAFETY FEATURES

3.6.1. Chapter 6 of the safety analysis report should present relevant information on the engineered safety features and associated systems. The engineered safety features to be covered in chapter 6 are those SSCs that are necessary to fulfil safety functions in the case of design basis accidents, design extension conditions (including design extension conditions with core melting) and some anticipated operational occurrences.

3.6.2. The description of the engineered safety features should demonstrate their capability to mitigate the consequences of accidents, to bring the nuclear power plant to a controlled state and, finally, to reach a safe state, in accordance with Requirements 51–58 and 65–67 of SSR-2/1 (Rev. 1) [3].

3.6.3. It is assumed that each group of systems covered in chapter 6 of the safety analysis report will separately address safety systems and safety features for design extension conditions, as appropriate, with a focus on adequate independence between the two corresponding levels of defence in depth.

3.6.4. Systems and provisions necessary for transferring heat to the ultimate heat sink (or to the diverse heat sink) should be described with special care, and their heat transfer function in cases of natural hazards exceeding the site design basis should be addressed.

3.6.5. The engineered safety features provided in different plant designs may vary. The engineered safety features explicitly mentioned in this Safety Guide are those that are typically used to limit the consequences of postulated accidents in light water cooled power reactors. These features should be treated as illustrative of the engineered safety features in general and of the kind of information that should be provided in this section of the safety analysis report.

3.6.6. The use of non-permanent equipment as part of accident management should be described in this chapter of the safety analysis report. The information provided should demonstrate that there are adequately robust design features to enable the reliable connection of non-permanent equipment, including connection during conditions induced by external hazards exceeding those of the design basis (see paras 6.28B, 6.45A and 6.68 of SSR-2/1 (Rev. 1) [3]).

3.6.7. For each of the engineered safety features, the detailed description of the design should, to the extent possible, include the items specified in Appendix II. In describing the materials used in the components of an engineered safety feature, interactions of the materials with fluids that could potentially impair the operation of the engineered safety feature should be taken into account. The description should cover the compatibility of materials used for engineered safety features with core coolant and containment spray solutions. All organic materials that exist in significant amounts within the containment building should be described, including plastics, lubricants, paints and coatings, electrical cable insulation, and asphalt.

Emergency core cooling systems and residual heat removal systems

3.6.8. This section should present relevant information on the emergency core cooling systems, residual heat removal systems and associated systems. The description should cover safety systems designed to cope with design basis accidents and safety features for design extension conditions, including design extension conditions with core melting. These systems can be related to the primary or secondary circuits or to the containment, depending on the reactor design (e.g. safety injection system, feedwater system, steam dump system, passive safety system). This section should provide relevant information on all the engineered safety features — both active and passive — in accordance with the general design aspects presented in chapter 3 of the safety analysis report in order to meet Requirement 52 of SSR-2/1 (Rev. 1) [3]. Further recommendations are provided in SSG-56 [28]. Relevant coolant storage tanks should also be described in this section. A description of the actuation logic (for protection systems) should be provided in chapter 7 of the safety analysis report.

3.6.9. This section should provide information on the emergency feedwater system (if not covered in chapter 10 of the safety analysis report) as an essential means of residual heat removal through the secondary side of the steam generators in the case of accident conditions in pressurized water reactors. The information provided should be linked to the general design aspects presented in chapter 3 of the safety analysis report and should demonstrate compliance with Requirement 51 of SSR-2/1 (Rev. 1) [3] and the recommendations provided in SSG-56 [28].

3.6.10. As with the emergency feedwater system, this section should describe the emergency steam dump system as another essential means for the removal of excessive or residual heat from the steam system under certain accident conditions (see Requirement 51 of SSR-2/1 (Rev. 1) [3] and the recommendations provided in SSG-56 [28]). Alternatively, the description of this system can be included in chapter 10 of the safety analysis report.

Emergency reactivity control system

3.6.11. This section should provide information on the means of ensuring reactor shutdown (e.g. by injecting concentrated boron) in addition to those provided by the standard reactivity control system.

Safety features for stabilization of the molten core

3.6.12. This section should provide relevant information on safety features to stabilize the molten core as a necessary means of molten core solidification — either inside the reactor pressure vessel or in a dedicated molten core localization system — as a necessary precondition for protecting the containment basemat and ensuring containment integrity in the long term.

Containment and associated systems

3.6.13. The description of the systems in this section should include both primary and secondary containment systems. This section should present relevant information on the containment and associated systems that are implemented to contain the effects of accidents and to prevent the loss of containment integrity in all plant states, including design extension conditions with core melting. This section should describe how the containment and associated systems meet Requirements 54–58 of SSR-2/1 (Rev. 1) [3] and comply with the recommendations provided in IAEA Safety Standards Series No. SSG-53, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants [29]. This section, in combination with chapter 15 of the safety analysis report, should provide a sufficient demonstration of containment integrity for all plant states and should provide the basis for the development of procedures, the specification of any necessary instrumentation, and of the necessary operator response and equipment response.

3.6.14. This section should describe both the concrete structures and the internal steel structures of the containment, including a demonstration of their performance. The containment systems to be covered in this section should include the following, as applicable:

(a) The containment heat removal systems or containment spray system and other active heat removal systems;

(b) The containment passive heat removal systems;
(c) The system for control of hydrogen and other combustible gases in the containment;
(d) The containment isolation system;
(e) The systems for protection of the containment against overpressure and underpressure;
(f) The containment annulus ventilation system;
(g) The containment ventilation system;
(h) The containment filtered venting system;
(i) The containment penetrations, airlocks, doors and hatches.

3.6.15. The maximum allowable leak rate for accident conditions should be specified in this section. In addition, the containment leakage testing system should be described. It should be demonstrated that the containment, containment penetrations and other containment isolation barriers allow for periodic leakage testing as part of the operational programmes. This section should provide sufficient basis for the development and implementation of an adequate testing programme for containment leakage (see Requirements 29 and 55 of SSR-2/1 (Rev. 1) [3] and SSG-53 [29]). The following tests should be considered, including information on the proposed schedule for performing pre-operational and periodic leakage rate tests as well as relevant special testing requirements:

(a) Containment integrated leak rate test;
(b) Containment penetration leak rate test;
(c) Containment isolation valve leak rate test.

Habitability systems

3.6.16. This section should present relevant information on habitability systems. Habitability systems are those engineered safety features that are provided to ensure that essential plant personnel can remain at their posts to take actions to operate the plant safely in operational states and to maintain acceptable conditions in the event of an accident. The relevant locations to be considered include control locations (i.e. control room, supplementary control room, and other emergency response facilities and locations), technical support centres and emergency centres. The description should include the available means of ensuring the habitability of such locations. Examples of these means are shielding, air filtration or purification systems, compressed air storage systems, and other provisions (e.g. adequate lighting) for the control of working conditions (see paras 3.9.12 and 3.9.18).

3.6.17. The habitability of control locations under design extension conditions with core melting should be addressed in this section of the safety analysis report. For remote sites, the description should include a demonstration of the habitability of these locations in the case of external hazards exceeding the design basis events combined with internal events.

Systems for the removal and control of fission products

3.6.18. This section should provide relevant information on the systems for the removal and control of fission products (if not already described as a part of the containment systems). The following specific information should be presented to demonstrate the performance capability of these systems:

(a) Considerations of the coolant pH and chemical conditioning in all necessary conditions of system operation;
(b) The effects on filter operability of postulated design basis loads due to fission products;
(c) The effects on filter operability of design basis release mechanisms for fission products.

Other engineered safety features

3.6.19. This section should present relevant information on any other engineered safety features implemented in the plant design that are not covered in any previous sections. Examples include the steam dump to the atmosphere and the backup cooling systems. The list of these systems to be described will depend on the type of plant under consideration. It should be decided whether certain systems (e.g. the auxiliary feedwater system) are described here or in chapter 9 of the safety analysis report, which deals with auxiliary systems in a much broader sense, or in chapter 10, which deals with steam and power conversion systems.

CHAPTER 7: INSTRUMENTATION AND CONTROL

Description of the instrumentation and control system

3.7.1. This chapter of the safety analysis report should provide relevant information on instrumentation and control systems, as described in Appendix II. In particular, this chapter should describe how Requirements 59–67 of SSR-2/1 (Rev. 1) [3] are met. Further guidance regarding the design of instrumentation and control systems is provided in IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [30].

3.7.2. This chapter of the safety analysis report should identify the instruments and the associated equipment necessary for operational states and for accident conditions. All the important instrumentation and control components — those important to safety and those not important to safety — should be described in this section.

3.7.3. This chapter of the safety analysis report should also describe the instrumentation and control systems and components that are qualified for their intended function, during their service life and for all plant states.

Design basis, overall architecture and functional allocation of the instrumentation and control system

3.7.4. This section should identify all instrumentation, control and supporting systems, including alarm, communication and display instrumentation, and should specify the functions allocated to each individual system. Furthermore, this section should describe the following:

(a) The overall architecture of the instrumentation and control system;
(b) The design basis for the instrumentation and control system;
(c) Provisions for normal operation and accident conditions;
(d) Safety classification of instrumentation and control systems and equipment;
(e) The strategies for defence in depth and for diversity;
(f) The identification of safety criteria.

General design considerations for instrumentation and control systems

3.7.5. This section should describe how the applicable design criteria are addressed, taking into account the importance of the system to safety, and should include the following:

(a) Quality of components and modules;
(b) Software quality, including its verification, validation and life cycle processes, as applicable, together with the quality of the related safety system;
(c) A description of how the performance requirements of all supported systems are met;

(d) Potential hazards to the system, including inadvertent actuations, and hazards relating to error recovery, self-testing and surveillance testing;
(e) Design criteria for access control, computer security and other aspects regarding nuclear security that might interfere with design criteria relating to safety;
(f) Redundancy and diversity requirements;
(g) Independence requirements;
(h) Fail-safe design of the protection systems;
(i) System calibration, testing and surveillance;
(j) Design of bypass and inoperable status indications;
(k) Prevention of a fault propagation path for environmental effects (e.g. high energy electrical faults, lightning) from one redundant portion of a system to another, or from another system to a safety system;
(l) Analysis of the application of the concept of defence in depth and diversity analysis for each potential failure mode, common cause failure (including software) and exposure of the system to internal and external hazards;
(m) The human–machine interface;
(n) Set points;
(o) Hardware and software classification;
(p) Equipment qualification;
(q) Replacement, upgrades and modifications to instrumentation and control systems.

The description of how the 'security by design' principle is applied on the basis of a computer security analysis is typically given in a separate document that contains sensitive information (see paras 2.29 and 3.13.29).

Control systems important to safety

3.7.6. This section should provide relevant information on the control system and demonstrate that Requirement 60 of SSR-2/1 (Rev. 1) [3] is met; that is:

> "**Appropriate and reliable control systems shall be provided at the nuclear power plant to maintain and limit the relevant process variables within the specified operational ranges.**"

Reactor protection system

3.7.7. This section should provide relevant information on the reactor protection system and demonstrate that Requirement 61 of SSR-2/1 (Rev. 1) [3] is met. In particular, information on the following specific aspects should be provided:

(a) The design bases for each individual reactor trip parameter, with reference to the postulated initiating events whose consequences the trip parameter is credited with mitigating.
(b) The specification of reactor trip system set points, time delays in system operation and uncertainties in measurement, and how these relate to the assumptions made in chapter 15 on safety analysis.
(c) Any interfaces with the actuation system for engineered safety features (including the use of shared signals and parameter measurement channels).
(d) Any interfaces with non-safety-related instrumentation, control or display systems, together with the provisions to ensure independence.
(e) The means employed to ensure the separation of redundant reactor trip system channels and the means by which coincidence signals are generated from redundant independent channels.
(f) Provisions for the manual actuation of the reactor trip system from the main control room, the supplementary control room and other emergency response facilities.
(g) In cases in which the actuation logic for the reactor trip is implemented by programmable digital means, a description of the development process that provides for disciplined specification and implementation of design requirements and the verification and validation activities planned to ensure that the final product is suitable for use. Interfaces with nuclear security provisions should be included as applicable (paras 2.29 and 3.13.29 should be taken into account).
(h) Monitoring, inspection, testing and maintenance of system and equipment.

Actuation systems for engineered safety features

3.7.8. This section should provide relevant information on the actuation systems for engineered safety features and demonstrate how Requirement 61 of SSR-2/1 (Rev. 1) [3] is met. In particular, information on the specific aspects listed in para. 3.7.7 regarding the reactor protection system, as applicable, should be provided here also.

3.7.9. In some plant designs, the actuation systems for reactor trip and the actuation system for engineered safety features are designed as one system. In such cases, it should be demonstrated how the independence of safety systems is ensured, and the strategies to protect against common cause failure within the safety systems should be specified.

Systems required for safe shutdown

3.7.10. This section should describe the instrumentation and control systems required to achieve and maintain a safe state (these systems are described in chapters 5, 9 and 10 of the safety analysis report). This includes instrumentation and control systems used to maintain the reactor core in a subcritical condition and to provide adequate core cooling to achieve and maintain both hot and cold shutdown. A list should be provided of the indications, controls, alarms and displays available in the control room and in the supplementary control room that are used by operating personnel to bring the plant to a safe state, to confirm that a safe state has been reached and is maintained, and to monitor the status of the plant and the trends in key plant parameters.

Information systems important to safety

3.7.11. This section should describe plant information systems important to safety. The information provided should include the following:

(a) A list of the parameters that are measured, the physical locations of the sensors, and the environmental qualification envelope, defined by the most severe operational states or accident conditions and by how long the reliable operation of the sensors is required.
(b) A specification of the parameters that are monitored by the plant computer displays in the control room, in the supplementary control room and in other emergency response facilities. The characteristics of any computer software (e.g. scan frequency, parameter validation and cross-channel sensor checking) used for filtering, analysis of trends, generation of alarms and long term storage of data should be described. If data processing and storage are performed by multiple computers, the means of achieving the synchronization of the different computer systems should also be described.

3.7.12. This section should also provide relevant information on any other diagnostic and instrumentation systems required for safety, for example any particular system needed for the management of severe accidents, leak detection systems, monitoring systems for vibrations and loose parts, and protective interlock systems that are credited in the safety analyses with preventing damage to safety related equipment and preventing accidents of certain types.

Interlock systems important to safety

3.7.13. This section should describe all other instrumentation systems that include interlock systems important to safety.

3.7.14. This section should describe relevant analyses and considerations of interlocks that prevent overpressurization of low pressure systems, interlocks to prevent overpressurization of the reactor coolant system during low temperature conditions, interlocks to isolate safety systems from non-safety systems and interlocks to preclude inadvertent interconnections between redundant or diverse safety systems for the purposes of testing or maintenance.

Diverse actuation system

3.7.15. This section should provide a description of the design of the diverse actuation system, including sensors, initiating circuits, bypasses, interlocks, priority actuation logic for automatic and manual control of plant equipment, operator interfaces, and support systems.

3.7.16. This section should provide an assessment of the level of diversity in digital instrumentation and control system architecture, a description of the independence of the safety functions, information on the application of the single failure criterion, a consideration of common cause failure, and the safety classification and qualification requirements. All plant states should be taken into account in the assessment.

Data communication systems

3.7.17. This section should describe all the data communication systems that are part of (or support) the other systems described in this chapter of the safety analysis report, addressing both safety and non-safety data communication systems.

3.7.18. The information provided should be sufficient to demonstrate that the data communication systems conform to relevant regulatory requirements and associated regulatory guidance and to recommendations in industry codes and standards applicable to data communication systems.

3.7.19. The means of and criteria for determining if a function has failed as a result of a communications failure should also be described.

Instrumentation and control in the main control room

3.7.20. This section should provide a description of the general philosophy followed in the design of the main control room and demonstrate that Requirement 65 of SSR-2/1 (Rev. 1) [3] is met.

3.7.21. This section should describe how the instrumentation and control systems allow the operating personnel in the control room to initiate or take manual control of each function necessary to control the plant and maintain safety.

3.7.22. This section should provide a description of the main control room layout, with emphasis on the presentation of information from the instrumentation and control in the main control room and the human–machine interface, including the following:

(a) Demonstration that there are sufficient displays in the control room to monitor all functions important to safety;
(b) The means by which the status of the plant is displayed;
(c) The means by which the safety status and trends of the key plant operating parameters are displayed;
(d) The safety classified indications and controls to implement emergency operating procedures and severe accident management guidelines.

3.7.23. This section should describe how the human–machine interface aspects of the design of the main control room conform to the human factors engineering programme described in chapter 18 of the safety analysis report.

3.7.24. The instrumentation and control relating to the habitability of the main control room, the supplementary control room and other emergency response facilities should also be described and should be consistent with the description of the corresponding systems in chapter 6 of the safety analysis report.

Instrumentation and control in supplementary control rooms

3.7.25. This section should provide an appropriate description of the supplementary control room functions and layout and should demonstrate that Requirement 66 of SSR-2/1 (Rev. 1) [3] is met.

3.7.26. This section should describe how the supplementary control room contains controls, indications, alarms and displays that are sufficient for the operator to bring the plant to a safe state, to confirm that a safe state has been reached and is maintained, and to monitor the status of the plant and the trends in key plant parameters.

3.7.27. This section should describe how the human–machine interface aspects of the design of the supplementary control room conform to the human factors engineering programme described in chapter 18 of the safety analysis report.

3.7.28. The means of physical and electrical isolation between the plant systems and the communication signals routed to the main control room and the supplementary control room should be described in detail to demonstrate that the supplementary control room is redundant and independent of the main control room.

3.7.29. The mechanisms for transferring priority control and communications from the main control room to the supplementary control room should be described so as to demonstrate how this transfer would occur under accident conditions.

Emergency response facilities

3.7.30. This section should describe the instrumentation and control in the emergency response facilitie (see paras 3.19.8 and 3.19.9) and should demonstrate that Requirement 67 of SSR-2/1 (Rev. 1) [3] is met. In particular, it should be shown that information about important plant parameters and the radiological conditions at the plant and in its surroundings, and a means of communication on the site and off the site, are provided to the emergency response facilities. This should include those facilities provided for plant staff to perform expected tasks for managing the response to an emergency under conditions generated by accidents and hazards, including certain control functions, if applicable.

Automatic control systems not important to safety

3.7.31. This section should describe the automatic control systems not important to safety. It should be demonstrated that postulated failures of these control systems will not degrade the operation of systems important to safety. It should also be demonstrated that the effects of a failure of an automatic control system will not create a condition that exceeds the acceptance criteria or assumptions established for design basis accidents.

Digital instrumentation and control systems

3.7.32. If digital instrumentation and control systems are used, this section should describe the overall scope of their application, including information on the following:

(a) The design qualification of digital systems, including software verification and validation;
(b) Protection against common cause failure;
(c) Functional requirements when implementing a digital protection system;
(d) Qualification and verification of predeveloped software;
(e) Software tools used to support the life cycle development of digital systems;
(f) Digital data communication.

The information provided in this section should demonstrate that Requirement 63 of SSR-2/1 (Rev. 1) [3] is met. Additionally, information to demonstrate that security measures for digital instrumentation and control systems [31] do not interfere with safety provisions should be provided (see 3.13.29).

Hazard analysis for instrumentation and control systems

3.7.33. This section should provide relevant information to demonstrate that the hazard analysis undertaken for instrumentation and control systems considered all plant states and modes of normal operation, including transitions between different modes of normal operation and the failure or non-availability of instrumentation and control systems.

CHAPTER 8: ELECTRICAL POWER

Description of the electrical power system

3.8.1. This chapter of the safety analysis report should provide relevant information on the electrical power systems. The information provided for individual electrical power systems should follow, to the extent applicable, the structure specified in Appendix II.

3.8.2. This chapter of the safety analysis report should describe how Requirement 68 of SSR-2/1 (Rev. 1) [3] on withstanding the loss of off-site power is met. Specific recommendations and guidance regarding the design of electrical power systems are provided in IAEA Safety Standards Series No. SSG-34, Design of Electrical Power Systems for Nuclear Power Plants [32].

3.8.3. This chapter should provide definitions, design features and classifications of the off-site power system, the on-site power system, the standby power system, and the alternate alternating current (AC) and direct current (DC) power systems.

3.8.4. The prioritization of the power supply from the power supply systems described in para. 3.8.3 to the non-safety loads and to the safety loads, during operational states and in accident conditions, should be described.

3.8.5. This chapter of the safety analysis report should also provide relevant information on how the safety power systems can be supplied (i.e. by either the preferred power supplies or the standby power sources). The description should include the alternate AC power system that supplies the safety power systems in design extension conditions.

General principles and design approach

3.8.6. In addition to the safety design criteria and rules and regulations, information on the following issues specific to electrical systems should be included:

(a) Postulated initiating events considered in the design, together with the functional requirements applicable to the electrical systems under the steady state conditions, short term operation conditions and transient conditions defined in the design basis;
(b) The impact of such events on all the on-site electrical power systems (AC and DC);
(c) The plant's capability to continue to fulfil safety functions and to remove decay heat from spent fuel for the period for which the plant is in a station blackout condition (loss of all AC power supplies);
(d) The design for reliability (redundancy, independence, diversity);
(e) The possibility of common cause failures that could render the safety power systems unavailable to fulfil their safety functions when called on, in the design, maintenance, testing and operation of the safety power systems and their support systems;
(f) The specific divisions of the electrical power systems in the plant, including the various system voltages and the designation of parts of the system that are considered essential;
(g) A demonstration of the functional adequacy of the electrical power systems important to safety (including breakers) and assurance that these

systems have adequate redundancy, physical separation, independence and testability, in conformance with the design criteria;

(h) A general description of the off-site power system, which is composed of the transmission system (grid), the switchyard connecting the plant with the grid and its interconnection to other grids, and the connection points to the on-site electrical system (or switchyard);

(i) The provisions for replacement and upgrades of and modifications to the electrical power systems.

Off-site power systems

3.8.7. This section should provide information relevant to the plant on the off-site electrical power systems. It should include a description of the off-site power systems, with emphasis on features for control and protection (breaker arrangements, manual and automatic disconnect switches) at the interconnection to the on-site power system.

3.8.8. This section should also describe the design requirements for the off-site power system (e.g. the switchyard design, the number of circuits to the on-site power system) — including the design requirements to support the safety function of the system — to provide sufficient reliability, capacity and capability.

3.8.9. This section should describe the design provisions used to protect the plant from off-site electrical disturbances and to maintain the power supply to in-plant auxiliaries. Information on grid reliability should also be provided, as should information on design provisions necessary to cope with frequent grid failures.

3.8.10. This section should describe the failure mode and effects analysis for off-site power system components. In addition, the results of a grid stability analysis (including stability after the main generator trip) should be provided.

On-site AC power systems

3.8.11. This section should provide relevant information on the AC power system at the plant and its main equipment. It should include a description of the on-site AC power systems, including the standby AC power systems (diesel or gas turbine driven systems), the generator configuration and the uninterruptible AC power system available for anticipated operational occurrences and accident conditions. Information on the selection of the following should also be included:

(a) Undervoltage (underfrequency and overvoltage) protection set points;

(b) Short circuit protection measures;
(c) Power quality limits;
(d) Equipment size, protection measures and means of coordination.

3.8.12. This section should describe the power requirements for each AC load in the plant, including the following:

(a) The steady state load and the startup kilovolt-amperes for motor loads;
(b) The nominal voltage and the allowable voltage drop (to achieve full functional capability within the required time period);
(c) The sequence and time necessary to achieve full functional capability for each load;
(d) The nominal frequency and the allowable frequency fluctuation;
(e) The number of divisions and the minimum number of divisions of engineered safety features to be energized simultaneously.

3.8.13. This section should describe the following:

(a) How the on-site AC power system is engineered to ensure the reliable delivery of emergency power to engineered safety features and uninterruptible AC power system loads.
(b) In the event of a loss of off-site power, how the standby AC power source is started and safety loads are sequenced to the safety buses without overloading the primary mover, and in time frames consistent with the assumptions presented in chapter 15 on safety analysis.
(c) In design basis accidents with a subsequent loss of off-site power, how the required safety loads can be sequenced onto the standby AC power source without overloading the primary mover and in time frames consistent with the assumptions presented in chapter 15 on safety analysis.
(d) How uninterruptible AC power is continuously provided to essential safety systems and instrumentation and control systems important to safety, irrespective of the availability of off-site AC power.
(e) How an alternate AC power supply is provided at the nuclear power plant, if the plant design depends on AC power to bring the plant to a controlled state following loss of off-site power and on-site safety standby power sources. It should also be described how the alternate AC power supply addresses diversity (e.g. that it is not susceptible to the events that caused the loss of on-site and off-site power sources) and has sufficient capacity to operate the systems necessary for coping with a station blackout, and how auxiliaries are qualified for their intended use.
(f) The provisions for the protection of AC power systems.

(g) The features to enable the safe use of non-permanent equipment to restore the necessary electrical power supply in design extension conditions with core melting (see para. 6.45A of SSR-2/1 (Rev. 1) [3]), demonstrating the equipment's adequacy and robustness.

On-site DC power systems

3.8.14. This section should provide relevant information on the DC power system. This includes a description of the characteristics, design features, ratings of breakers, transformers, batteries, switchgears, rectifiers and inverters that support the safe operation of the plant. The following information specific to DC power systems should be provided:

(a) An evaluation of the long term discharge capacity of the battery (the projected voltage decay as a function of time without charging when subjected to design loads);
(b) The major DC loads present (including the uninterruptible AC power system inverters and any DC loads not important to safety, such as the lubrication oil pumps for the turbine bearings);
(c) A description of the fire protection measures for the DC battery vault area and cable systems.

3.8.15. A justification of the power requirements should be provided for each plant DC load, including:

(a) The steady state load;
(b) Surge loads (including accident conditions);
(c) The load sequence;
(d) The nominal voltage;
(e) The allowable voltage drops (to achieve full functional capability within the required time period);
(f) The number of divisions;
(g) The provisions for protection of DC power systems.

3.8.16. This section should demonstrate the continuity of the DC power supply so that the monitoring of the key plant parameters and the completion of short term actions necessary for safety are maintained in the event of the loss of all the AC power sources. Information on the possible options to recharge batteries from alternate AC power sources should also be provided.

Electrical equipment, cables and raceways

3.8.17. This section should demonstrate that electrical equipment, cables and their raceways (including cable supports, wall and floor penetrations, and fire stops) are selected, rated and qualified for their service and for environmental conditions. Account should be taken of the cumulative effects of radiation exposure and thermal ageing expected over their service life. The seismic qualification, the electromagnetic interference qualification and the fire resistance of electrical equipment, buses, cable trays and their supports should also be described.

3.8.18. This section should identify at least four classes of cable, as follows:

(a) Instrumentation and control cables;
(b) Low voltage power cables (1 kV or less);
(c) Medium voltage power cables (greater than 1 kV to 35 kV);
(d) High voltage power cables (greater than 35 kV).

3.8.19. This section should describe the environmental qualification of cables and electrical penetrations that have to withstand conditions inside the containment during and after a loss of coolant accident, a main steam line break or other adverse environmental conditions, including severe accidents.[10]

Grounding, lightning protection and electromagnetic compatibility

3.8.20. A description of the provisions for electromagnetic compatibility of the nuclear power plant and its electrical and instrumentation and control systems should be provided. This section should also include a description of the grounding and lightning protection (both internal and external protection) system, including the components associated with the various grounding subsystems (e.g. station grounding, system grounding, equipment safety grounding, any special grounding for sensitive instrumentation and computer or low-signal control systems). Grounding and lightning protection plan drawings should also be included.

3.8.21. The industry codes and standards used in designing the subsystems should be identified, as should the bases for the related acceptance criteria. The analyses undertaken and any underlying assumptions used should be described

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[10] This is applicable only to the cables and electrical penetrations requiring environmental qualification for severe accidents.

to demonstrate that the acceptance criteria for the grounding subsystems will be successfully incorporated into the as-built plant.

CHAPTER 9: AUXILIARY SYSTEMS AND CIVIL STRUCTURES

3.9.1. Chapter 9 of the safety analysis report has two main parts. Part A of chapter 9 should provide information about the auxiliary systems not included in other chapters of the safety analysis report. In particular, chapter 9A should identify systems that are essential for the safe shutdown of the plant or for the protection of the public. For each system, the description should, to the extent possible, follow the structure given in Appendix II. The description of auxiliary systems should be sufficient to demonstrate that Requirements 69, 71–74, 76 and 80 of SSR-2/1 (Rev. 1) [3] are met. Specific recommendations on the design of auxiliary systems are provided in IAEA Safety Standards Series No. SSG-62, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants [33].

3.9.2. Part B of chapter 9 of the safety analysis report should provide information on the civil structures of the plant. This part should describe how the various civil structures in the plant comply with the general design requirements and other rules specified in chapter 3 of the safety analysis report. For each civil structure the description should, to the extent possible, follow the structure given in Appendix II and should demonstrate that the design of the civil structure follows general design rules using recognized engineering practices in accordance with Requirement 18 of SSR-2/1 (Rev. 1) [3].

3.9.3. Plant auxiliary systems and civil structures can vary among designs. The examples of subsystems provided below are not, therefore, intended to represent a complete list of systems to be described in this chapter of the safety analysis report. The structure of this chapter can be modified according to the specificities of the design, with account taken of the information provided in other chapters of the safety analysis report.

CHAPTER 9A: AUXILIARY SYSTEMS

Fuel storage and handling systems

3.9.4. This section should provide relevant information on the fuel handling and storage systems to demonstrate that the fuel is maintained in safe conditions at all times (see Requirement 80 of SSR-2/1 (Rev. 1) [3]). This information should include details of the proposed arrangements regarding subcriticality, shielding, handling, storage, cooling, spent fuel pool leakages and load drops, and the transfer and transport of nuclear fuel within the nuclear power plant. The following subsystems should be covered:

(a) The fresh fuel storage and handling system;
(b) The spent fuel storage and handling system;
(c) The spent fuel pool cooling and cleanup system;
(d) The handling systems for fuel cask loading.

3.9.5. With regard to fresh fuel, the information provided should include considerations such as packaging, handling, storage, criticality prevention, and fuel integrity monitoring and control.

3.9.6. With regard to reprocessed and irradiated fuel, the information provided should include considerations such as appropriate provisions for radiation protection, criticality prevention, fuel integrity control (including special provisions to deal with failed fuel), fuel chemistry, fuel cooling, and arrangements for fuel consignment and transport. Special attention should be devoted to the provisions for the 'practical elimination' of significant fuel degradation in the spent fuel pool and for uncontrolled radioactive releases.

3.9.7. The use of non-permanent equipment for the fulfilment of safety functions in respect of the spent fuel pool as part of accident management should be described in this section, including a demonstration that there are adequately robust design features to enable the reliable connection of non-permanent equipment, including under conditions induced by external hazards exceeding those of the design basis (see para. 6.68 of SSR-2/1 (Rev. 1) [3]).

Water systems

3.9.8. This section should provide relevant information on the water systems associated with the plant. In particular, it should provide information on the following:

(a) The service water system;
(b) The component cooling water system for reactor auxiliaries (intermediate cooling circuits);
(c) The essential chilled water system;
(d) The demineralized water make-up system;

(e) The ultimate heat sink system (including any diverse heat sink);
(f) The condensate storage and transfer system.

3.9.9. The robustness of the systems necessary for the transfer of residual heat to the ultimate heat sink system, and of the heat sink itself in the case of extreme external hazards, should be addressed in this section.

Process and post-accident sampling systems

3.9.10. This section should provide relevant information on the auxiliary systems associated with the reactor process system. It should include, for example, information on the process and post-accident sampling systems. The compressed air systems are dealt with in another section of this chapter, while the chemical control and volume control systems are covered in chapter 5 of the safety analysis report.

Air and gas systems

3.9.11. The systems that provide air for service and maintenance uses, including compressed air systems and service gas systems, should be described in this section. A description should also be provided of the capabilities to interconnect or isolate the instrumentation and control air system from the service air system if the design provides two such systems that can be interconnected.

Heating, ventilation and air-conditioning systems

3.9.12. This section should provide relevant information on the heating, ventilation, air-conditioning and cooling systems. The following heating, ventilation and air-conditioning subsystems should be covered:

(a) The heating, ventilation and air-conditioning systems in control locations (and other areas requiring habitability control);[11]
(b) The heating, ventilation and air-conditioning system in the spent fuel pool area;
(c) The heating, ventilation and air-conditioning systems in the auxiliary and radioactive waste areas;
(d) The heating, ventilation and air-conditioning system in the turbine building;

[11] These areas include the main control room, the supplementary control room, other emergency response facilities, and other areas or rooms hosting sensitive equipment (e.g. instrumentation and control equipment, electrical equipment, computers).

(e) The heating, ventilation and air-conditioning systems for engineered safety features;
(f) The chilled water system for heating, ventilation and air-conditioning.

Fire protection systems

3.9.13. This section should describe the provisions made to ensure that the plant design provides adequate fire protection. In particular, this section should provide relevant information to demonstrate that the design of the fire protection systems includes adequate provisions for defence in depth, considering the need for fire prevention, fire detection, fire warning, fire suppression, smoke control and fire containment. Consideration should be given to the selection of materials, the physical separation of redundant systems, resistance against external hazards (if considered to mitigate the consequences of external events) and the use of barriers to segregate redundant trains.

3.9.14. The extent to which the design provides adequate fire protection should be assessed. This section may refer to information provided in other sections of the safety analysis report (e.g. chapter 15 on safety analysis). Where appropriate, the provisions to ensure the safety of personnel in the event of a fire should also be described in this section.

Supporting systems for diesel generators or for gas turbine generators

3.9.15. The support systems for the diesel generators (or for the gas turbines) should be covered in this section (except for the AC systems, which are covered in chapter 8 of the safety analysis report). The design of supporting systems should be such as to ensure that the performance of these systems is consistent with the safety significance of the system or component that they serve in all plant states. The following subsystems for diesel generators or for gas turbine generators should be typically addressed in this section:

(a) The generator fuel oil storage and transfer system;
(b) The generator cooling water or cooling air system;
(c) The generator starting system;
(d) The generator lubrication system;
(e) The generator combustion air intake and exhaust system.

Overhead lifting equipment

3.9.16. The overhead lifting equipment (in particular, the reactor building crane and the fuel building crane) should be described in this section. The related rules and assumptions for design should also be described and justified. Special attention should be given to critical load handling operations that could have an effect on the fulfilment of safety functions. The information provided should demonstrate that Requirement 76 of SSR-2/1 (Rev. 1) [3] is fulfilled.

3.9.17. The information to be provided should include the following:

(a) The parameters defining the load that, if dropped, would cause the greatest damage;
(b) The areas of the plant where the load would be handled;
(c) The design of the overhead lifting equipment;
(d) The operating, maintenance and inspection procedures applied.

Miscellaneous auxiliary systems

3.9.18. This section should provide relevant information on any other plant auxiliary system whose operation might influence plant safety and that has not been covered in any other part of the safety analysis report. Examples of systems to be included in this section are:

(a) Communications systems, including diverse means of maintaining communication on the site and off the site;
(b) Lighting and emergency lighting systems;
(c) Equipment and floor drainage systems;
(d) Interfacing water systems (raw water reserves, demineralized water system, potable and sanitary water system);
(e) Chemistry systems;
(f) The storage system for non-permanent equipment used in design extension conditions.

CHAPTER 9B: CIVIL ENGINEERING WORKS AND STRUCTURES

3.9.19. Part B of chapter 9 of the safety analysis report should describe how the general design requirements specified in chapter 3 of the safety analysis report have been complied with in the design of specific structures in the nuclear power plant. Three groups of civil structures should be considered: the foundations, the reactor building and other civil structures. In describing the structures, a standardized format for the information provided (specified in Appendix II) should be followed to the extent possible.

3.9.20. The following information specific to civil engineering works and structures should be provided:

(a) Details of the range of anticipated structural loadings, together with the associated requirements for the buildings and structures, and the consideration given to hazards in the design.
(b) A description of the extent to which load–source interactions have been considered, with a confirmation of the ability of the buildings and structures to withstand the required load combinations while fulfilling their main safety functions.
(c) If a safety or seismic classification for buildings and structures has been used, the basis of the classification for the design option should be described. It should be demonstrated that the safety classification of buildings containing items important to safety is consistent with the classification of the SSCs that they contain. Further recommendations are provided in Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67 [24].
(d) If a structure is intended to provide additional functions separate from its structural function (e.g. functions of radiation shielding, separation and containment), the additional requirements identified for these functions should be specified and reference should be made to other sections of the safety analysis report, as appropriate.

Foundations and buried structures

3.9.21. In this section, information on the foundations should be provided, including diagrams containing plan and section views of the foundations, to define the primary structural aspects and elements relied on to perform the foundation function. The description should include the soil–structure interaction (see NS-G-3.6 [16]). Additionally, the type of foundation, its structural characteristics and the general arrangement of each foundation should be presented. In particular, foundations of steel or concrete containment should be described, as well as all seismically classified structures.

Reactor building

3.9.22. This section should describe the design features of the reactor building[12] provided to comply with Requirements 54–58 of SSR-2/1 (Rev. 1) [3]. Specific design features of the primary containment, such as its leaktightness, mechanical resistance, pressure-retaining capability and protection against hazards, should be covered. The concrete and steel internal structures of the containment should be described. If the design incorporates a secondary containment, it should also be described in this section of the safety analysis report. The information described in this section of the safety analysis report should be consistent with and complementary to the information provided in chapter 6 of the safety analysis report (see para. 3.6.13).

3.9.23. This section should also provide sufficient information to demonstrate the performance of the containment in all plant states and combinations of loads, in accordance with established acceptance criteria (see SSG-53 [29]).

Other structures

3.9.24. Other civil structures of the plant that are relevant to nuclear safety should be described in this section; this includes the control building, the auxiliary building, the ultimate heat sink structures and the emergency response facilities.

CHAPTER 10: STEAM AND POWER CONVERSION SYSTEMS

3.10.1. Chapter 10 of the safety analysis report should provide information on the design of plant steam and power conversion systems. The information provided should, to the extent possible, follow the structure specified in Appendix II and demonstrate how the system design meets Requirement 77 of SSR-2/1 (Rev. 1) [3]. The following information specific to steam and power conversion systems should also be provided:

(a) The performance requirements for the turbine generators in operational states.
(b) A description of the following:
    (i) The main steam line piping and the associated control valves;
    (ii) The main condensers;

[12] The reactor building is the building that shelters the primary containment and, if appropriate, the secondary containment.

(iii) The main condenser evacuation system;
(iv) The turbine generator system;
(v) The turbine gland sealing system;
(vi) The turbine bypass system;
(vii) The circulating water system;
(viii) The condensate cleanup system;
(ix) The condensate and feedwater system;
(x) The steam generator blowdown system (where applicable).

(c) The water chemistry programme, together with a description of the materials of the steam, feedwater and condenser systems.

(d) The consideration of flow accelerated corrosion in the design of the systems.

3.10.2. This chapter of the safety analysis report should emphasize those aspects of the design and operation of the steam and power conversion systems that affect the reactor and its safety features or contribute towards the control of radioactive material. The information provided should show the capability of the system to function without compromising (directly or indirectly) the safety of the plant, under both steady state and transient situations.

Role and general description

3.10.3. In this section, a summary description indicating the principal design features of the steam and power conversion systems should be provided. This description should include an overall system flow diagram and a summary table of the important design and performance characteristics (including a heat balance at rated power) and should indicate safety related system design features. The boundaries between the reactor coolant system and the main steam supply and feedwater systems should be specified.

Main steam supply system

3.10.4. This section should describe the main steam supply system and main steam line piping and should include piping and instrumentation diagrams showing system components, including interconnecting piping.

3.10.5. The descriptions should include sufficient detail to demonstrate the reliable fulfilment of safety functions, including fast and reliable isolation and steam relief. A demonstration that the separation of steam lines prevents leakage from one affecting another, and provides protection against an aircraft crash, should also be included (see para. 3.3.45).

3.10.6. For a boiling water reactor with a direct cycle design, the description of the main steam system should cover all components, from the outermost containment isolation valves to the turbine stop valves. It should also include connected piping of large diameters, up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation.

3.10.7. For pressurized water reactors, the description of the main steam system should extend from the connections to the secondary sides of the steam generators to the turbine stop valves. It should also include the containment isolation valves; the safety and relief valves; connected piping of large diameters, up to and including the first valves that are either normally closed or capable of automatic closure during all modes of normal operation; and the steam line to the auxiliary feedwater pump turbine, if applicable. The steam bypass and steam dump station to the atmosphere may also be described in this section (if not included in chapter 6 of the safety analysis report).

Feedwater systems

3.10.8. The main feedwater system and the auxiliary feedwater system should be described in this section, including the capability to supply adequate feedwater to the nuclear steam supply system, the criteria for isolation from the steam generator or from the reactor coolant system, and the environmental design requirements.

3.10.9. The description should include an analysis of the effects of component failure and of equipment malfunction on the reactor coolant system. It should also include an analysis of the detection and isolation provisions that are implemented to preclude radioactive releases to the environment in the event of a pipe leak or break or the degradation of the integrity of safety related equipment.

Turbine generator

3.10.10. The turbine generator system and associated equipment (including moisture separation and turbine overspeed protection), the use of extraction steam for feedwater heating, and control functions that could influence the operation of the reactor coolant system should all be described in this section. Piping and instrumentation diagrams and layout drawings should be provided to show the general arrangement of the turbine generator system and associated equipment with respect to safety related SSCs.

3.10.11. Information should be provided to demonstrate the structural integrity of turbine rotors and the protection against damage of safety related components due to a failure of a turbine rotor that produces a high energy missile.

3.10.12. This section should describe the equipment design and design bases of the turbine generator system, including the performance requirements under normal operation. It should also describe the following:

(a) The intended mode of normal operation (e.g. base load or load following);
(b) The functional limitations imposed by the design or the operational characteristics of the reactor coolant system (e.g. the rate at which the electrical load may be increased or decreased by means of reactor control rod motion or steam bypass);
(c) The design codes to be applied.

3.10.13. The information provided should include the seismic design criteria; the bases for the chosen criteria; and the safety, seismic and quality group classifications for the turbine generator system components, equipment and piping.

Turbine and condenser systems

3.10.14. In this section, the principal design features and subsystems associated with the operation of the turbine and the condenser should be described. These subsystems are design specific but they usually include the following:

(a) The main condenser.
(b) The condenser air extraction system (off-gas treatment in boiling water reactors).
(c) The circulating water system.
(d) The condensate system.
(e) The condensate cleanup system.
(f) The turbine auxiliary systems:
    (i) The turbine gland sealing system;
    (ii) The turbine bypass system to the condenser.
(g) The generator auxiliary systems.

Steam generator blowdown processing system

3.10.15. The steam generator blowdown processing system[13] and its design basis should be described in this section. This should include a description of its ability to maintain optimum secondary side water chemistry in the recirculating steam generators of pressurized water reactors during normal operation and during anticipated operational occurrences (e.g. main condenser in-leakage and primary-to-secondary leakage).

3.10.16. The design basis should include a consideration of the expected flows and the design flows in terms of the following aspects:

(a) All modes of normal operation (i.e. process and process bypass);
(b) All process design parameters and equipment design capacities;
(c) The expected temperatures and the design temperatures for temperature sensitive treatment processes (e.g. demineralization, reverse osmosis);
(d) The process instrumentation and control necessary to maintain operations within established parameter ranges.

Implementation of break preclusion for the main steam and feedwater lines

3.10.17. This section should describe the scope of the implementation of break preclusion in the main steam and feedwater lines. The aspects that impact plant safety (either direct effects on the fulfilment of the fundamental safety functions or indirect effects such as secondary damage to the plant systems, for example by pipe whip or extraordinary pressure loading) should be emphasized. If relevant, the description should also include how the 'leak before break' concept has been implemented.

CHAPTER 11: MANAGEMENT OF RADIOACTIVE WASTE

3.11.1. This chapter of the safety analysis report should describe the measures proposed for the safe management of radioactive waste of all types that will be generated throughout the lifetime of the plant as well as how these measures meet the relevant safety requirements. Relevant safety requirements include those regarding waste minimization (see para. 4.8 of SSR-2/1 (Rev. 1) [3]), treatment of radioactive waste (see Requirements 78 and 79 of SSR-2/1 (Rev. 1) [3]) and

[13] This is sometimes called the 'steam generator blowdown system' or the 'steam generator blowdown purification system'.

programmes for the management of radioactive waste (see Requirement 21 of SSR-2/2 (Rev. 1) [4]). Further requirements are provided in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [34]. In addition, recommendations of particular relevance to this Safety Guide are provided in IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [35], and further recommendations are given in IAEA Safety Standard Series No. SSG-40, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors [36].

3.11.2. This chapter of the safety analysis report should include a description of the following:

(a) The capabilities of the plant for pretreatment, treatment, conditioning and storage of liquid, gaseous and solid radioactive waste;
(b) The instrumentation used to monitor possible radioactive releases, both on the site and off the site.

The disposal of radioactive waste is expected to take place in a dedicated facility (a radioactive waste disposal facility) and is therefore not covered in this chapter of the safety analysis report. However, any waste acceptance criteria for such repositories should be taken into account in this chapter.

3.11.3. The radioactive waste described in this chapter of the safety analysis report is that generated during normal operation (i.e. in different operational activities, such as refuelling, purging, equipment downtime and maintenance). Any radioactive waste potentially generated during anticipated operational occurrences and accident conditions should be determined and described separately in chapter 15 of the safety analysis report.

3.11.4. The sections in this chapter of the safety analysis report should provide relevant information on the radioactive waste processing systems (i.e. pretreatment, treatment and conditioning systems) as well as the waste storage facilities located on the site. This information should include a description of the design features of the facilities for the pretreatment, treatment, conditioning and storage of solid, liquid and gaseous radioactive waste arising from all activities on the site throughout the lifetime of the plant. The conditioning of liquid and solid waste for future disposal should also be covered. The description should include the SSCs provided for these purposes and the instrumentation provided to monitor for possible leaks of radioactive waste. The scope and structure of the description of systems for the processing of radioactive waste should, to the extent possible, follow the structure specified in Appendix II.

Sources of waste

3.11.5. This section should include a description of the main sources of solid, liquid and gaseous radioactive waste and the estimated rate at which such waste will be generated. This section should also describe the expected liquid and gaseous radioactive releases under normal operation, in compliance with the design requirements.

3.11.6. The assessment of gaseous and liquid releases resulting from accident conditions is treated in chapter 15 of the safety analysis report, although the results of such assessments may also be described here and used as input.

3.11.7. This section should provide information on the quantities of waste and the rates of accumulation, as well as the conditions and forms of radioactive waste resulting from normal operation and the methods and technical means of processing, storage and transport of such waste.

3.11.8. This section should describe the specific options considered for the safe predisposal management of waste. The consideration of waste should cover all stages of waste management over the lifetime of the plant.

3.11.9. Measures to minimize the generation and accumulation of waste at all stages of the lifetime of the plant should be described. They should include measures taken to reduce the waste arising to a level that is as low as practicable. These measures are required to minimize both the volume and the activity of the waste (see para. 4.8 of SSR-2/1 (Rev. 1) [3]) and should be implemented in such a way as to meet any specific criteria, such as waste acceptance criteria, associated with the design of the waste storage and disposal facility.

Systems for management of liquid radioactive waste

3.11.10. This section should describe the capabilities of the plant for pretreatment, treatment, conditioning and storage of liquid radioactive waste generated during operation and resulting from accident conditions.

3.11.11. The information provided in this section should include descriptions of the following activities and measures that are associated with radioactive liquid waste generated at all stages of the lifetime of the plant:

(a) Control and containment of waste, including proposals to categorize and separate it, as necessary.
(b) Handling of waste, including provisions for its safe handling while transferring, moving or transporting it from the point of origin to the specified storage point. The possible need to retrieve waste at some time in the future, including during the decommissioning stage, should also be considered.
(c) Processing of waste in accordance with established procedures, with account taken of the interdependencies among all steps in the management of radioactive waste, including the anticipated disposal option. In assessing different options, consideration should be given to establishing the most suitable option that, to the extent possible, does not foreclose alternative options, in the event that the preferred waste disposal options change over the lifetime of the plant. The possible need for specialized systems to deal with issues arising from processing (e.g. evaporating, conditioning), such as volatility, chemical stability, reactivity and criticality, should be addressed, and any such systems should be described.
(d) Storage of waste, including information on the quantities, types and volumes of waste. The need to categorize and separate waste within the provisions for storage should be considered. The possible need for specialized systems to deal with issues of storage, such as cooling, containment, volatility, chemical stability, reactivity and criticality, should also be addressed, and any such systems should be described.

3.11.12. This section should include an assessment of liquid discharges during operational states. The assessment of radioactive releases in accident conditions and the resulting radiological consequences should be included in chapter 15 of the safety analysis report.

3.11.13. This section should also address the possible means of dealing with potentially large volumes of contaminated water generated under accident conditions.

Systems for management of gaseous radioactive waste

3.11.14. This section should describe the capabilities of the plant for pretreatment, treatment, conditioning and storage of gaseous radioactive waste generated during normal operation.

3.11.15. This section should also include an assessment of gaseous discharges during normal operation. The assessment of radioactive releases in accident conditions and the resulting radiological consequences should be included in chapter 15 of the safety analysis report.

Systems for management of solid radioactive waste

3.11.16. In this section, the term ‘system for management of solid waste’ refers to a permanently installed system. This section should describe the capabilities of the plant for pretreatment, treatment, conditioning and storage (prior to shipment) of wet and dry solid radioactive waste generated during normal operation.

3.11.17. Similarly, as in the case of liquid wastes, information provided for solid waste should cover their control, handling, processing and storage. This section should also contain information on the preparations for the safe transport of radioactive waste to another facility for storage or disposal, confirming that the requirements established in IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material, 2018 Edition [37], are met.

Process and effluent radiological monitoring and sampling systems, including on-site and off-site monitoring

3.11.18. This section should describe the systems and equipment that monitor and sample the process and effluent streams in order to measure and control the discharge of radioactive materials generated in operational states and accident conditions. This section should also demonstrate that the means of on-site radiation monitoring comply with paras 6.77–6.82 of SSR-2/1 (Rev. 1) [3] and those of off-site monitoring comply with para. 6.84 of SSR-2/1 (Rev. 1) [3].

CHAPTER 12: RADIATION PROTECTION

3.12.1. This chapter of the safety analysis report should deal specifically with the occupational exposure of workers in the nuclear power plant. Public exposure for all plant states, including the determination of doses to the public during normal operation, is addressed separately in chapters 15 and 20 of the safety analysis report.

3.12.2. This chapter of the safety analysis report should provide information on the policy, strategy, methods and provisions for radiation protection. The expected occupational exposures during operational states, and the measures taken to avoid and restrict exposures, should also be described.

3.12.3. The potential exposure of workers in the nuclear power plant under accident conditions, including design extension conditions with core melting, should be addressed, and the means and other measures taken to minimize such exposures should be described.

3.12.4. The information provided in this chapter of the safety analysis report should either describe the ways in which adequate provisions for radiation protection have been incorporated into the design, or refer to other sections of the safety analysis report where this information can be obtained.

3.12.5. This chapter of the safety analysis report should demonstrate how the basic radiation protection measures of time, distance and shielding have been considered. It should also demonstrate that appropriate design and operational arrangements have been made to reduce the amount of unnecessary radiation sources.

3.12.6. The information provided in this chapter of the safety analysis report should demonstrate compliance with IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [38]; with paras 2.6 and 2.7 and with Requirement 81 of SSR-2/1 (Rev. 1) [3]; and with Requirement 20 of SSR-2/2 (Rev. 1) [4]. Further recommendations and guidance are provided in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [39].

Optimization of protection and safety

3.12.7. This section should provide a description of the design provisions that are implemented and the operating organization’s policy for the optimization of protection and safety, both in operational states and in accident conditions, for the entire lifetime of the plant, including decommissioning. This policy should be consistent with the general design requirements in chapter 3 of the safety analysis report.

3.12.8. The specific measures taken to optimize protection and safety should be described. This section should also describe the estimated occupancy of designated radiation areas during normal operation and in anticipated operational occurrences. The need for workers to be present in areas where radiation levels are high should be justified, and the working hours in such areas should be limited by means of careful planning to restrict occupational exposures.

Sources of radiation

3.12.9. This section should provide a description of all on-site sources of radiation in operational states (including outages for inspections, maintenance and refuelling) as well as in accident conditions. The sources should include the following:

(a) Contained and immobile sources of radioactive material, such as the following:
   (i) The reactor core;
   (ii) The reactor vessel;
   (iii) The reactor internals and control rods;
   (iv) The reactor coolant;
   (v) The chemical and volume control system;
   (vi) The spent fuel pool cooling system;
   (vii) The liquid, gaseous and solid radioactive waste systems (described consistently with chapter 11);
   (viii) The residual heat removal systems;
   (ix) Spent fuel;
   (x) Other activated components (e.g. the biological shield).
(b) Sources of airborne radioactive material, such as the following:
   (i) Leakages from systems and equipment for the transport of radioactive fluids;
   (ii) Activation of air;
   (iii) Gaseous leakages from the distribution of coolant from the spent fuel pool (affecting the containment atmosphere, the fuel building atmosphere and the auxiliary building atmosphere).

3.12.10. Special source terms should be set out for accident conditions, including design extension conditions with core melting. The quantitative characteristics, such as mass of fuel or volume of coolant inventory, of different radiation sources should be described.

3.12.11. This section should also describe the possible pathways for occupational exposure associated with the radiation sources in all operational states as well as in accident conditions.

Design features for radiation protection

3.12.12. This section should describe the design features of the equipment and the facility that provide for radiation protection. This should include information on the various means implemented to achieve the following:

(a) Minimization of the source term;
(b) Minimization of the total working time in a designated radiation area;
(c) Reduction of the radiation level in an area or around any equipment or component;
(d) Reduction of the generation of activated corrosion products and minimizing their transport and deposition.

3.12.13. The description of the means of reducing occupational exposure should include the following:

(a) Minimizing contamination by choosing corrosion resistant materials, using an adequate water chemistry regime, enhancing the purifying capacity of the primary coolant and decontaminating the facility;
(b) Using radiation shielding, prior mock-up training, remote operation and other actions to reduce external exposure;
(c) Reducing internal exposure by isolation, ventilation, decontamination and use of protective clothing and respiratory protective equipment;
(d) Categorizing plant areas (zones) in accordance with the radiation level and the contamination level, and restricting access to controlled areas;
(e) Categorizing plant personnel in accordance with their working conditions and carrying out corresponding measures for the control and supervision of the work;
(f) Monitoring individuals and working areas;
(g) Using warning signs to control access and to avoid inadvertent access and unnecessary exposures.

3.12.14. This section should describe how the principles of radiation protection are applied in the design, taking into account Requirement 1 of GSR Part 3 [38], including a description of the means implemented to ensure the following:

(a) No person receives doses of radiation in excess of the dose limits as a result of normal plant operation.
(b) Occupational exposures in all plant states are as low as reasonably achievable.
(c) Dose constraints are used to avoid inequities in the dose distributions.
(d) Measures are taken to protect workers from receiving doses near the dose limits year by year.
(e) All practicable steps are taken to avoid or minimize exposures due to accidents with radiological consequences (including an analysis of potential accidents and the response and any protective or remedial actions taken).
(f) All practicable steps are taken to mitigate the radiological consequences of any accident.

3.12.15. This section should provide information on radiation monitoring in respect of all significant radiation sources and in all activities throughout the lifetime of the plant (i.e. in addition to the effluent monitoring described in para. 3.11.18). This section should demonstrate that the arrangements for individual monitoring and workplace monitoring meet Requirement 82 of SSR-2/1 (Rev. 1) [3].

3.12.16. This section should contain a description of the stationary instrumentation for monitoring of radiation levels and for continuous monitoring of airborne radioactive material. In addition, it should provide the criteria for the selection and placement of this instrumentation and should address the design provisions for the decontamination of equipment, if necessary.

3.12.17. The means of monitoring and decontamination of personnel, including both fixed and portable instruments for measuring surface contamination, should be described. This should include adequate provisions for monitoring during operational states, design basis accidents and design extension conditions.

Dose constraints and dose assessment

3.12.18. The dose constraints established for workers in each plant state should be stated here (see also para. 3.3.7). This section should demonstrate that these dose constraints are achievable in operational states and in accident conditions. An assessment of the potential effective doses and the potential equivalent doses from different sources of radiation and for various work activities should be presented.

3.12.19. Dose assessment as described in this section should be based on individual monitoring during plant operation, on operational experience from similar plants or on appropriate computational models. Data from similar plants and descriptions of computational models should be provided in the safety analysis report or should be adequately referred to.

Radiation protection programme

3.12.20. This section should describe (consistently with the operational programmes described in chapter 13 of the safety analysis report) the administrative measures, equipment, instrumentation, facilities and procedures for the radiation protection programme, which should be designed to meet Requirement 24 of GSR Part 3 [38]. It should be demonstrated that the radiation protection programme for the plant is based on a prior risk assessment that takes into account the location and magnitude of all radiation hazards and covers the following:

(a) The assignment of responsibilities for protection and safety to different management levels;
(b) The designation and functions of qualified experts;
(c) The integration of occupational radiation protection with other areas of health and safety, such as industrial hygiene, industrial safety and fire safety;
(d) The measures necessary to optimize protection and safety;
(e) The classification of working areas and access control;
(f) The issuing of radiation protection procedures, local rules and other relevant documents to personnel, and supervision of the work;
(g) The monitoring of individuals and the workplace, keeping in the plant the records of investigations of radiation levels and contamination, the results of radiation monitoring and other relevant information;
(h) Limitation of the number of personnel working in controlled areas, and the planning and managing of such work and the corresponding work permits;
(i) The selection and use of protective clothing and respiratory protective equipment;
(j) The shielding of facilities and equipment;
(k) The establishment and maintenance of records of occupational exposure and the health surveillance of workers, in accordance with Requirement 25 of GSR Part 3 [38];

(l) Reduction of the radiation sources and the source term, in accordance with paras 3.12.9 and 3.12.12;
(m) The training programme for workers, including retraining, and procedures for reviewing training and qualifications;
(n) Investigation and reporting of any radiation accidents, and the taking of corrective actions to prevent a recurrence of such an accident;
(o) Arrangements for emergency preparedness and response (see paras 3.19.1–3.19.12).

CHAPTER 13: CONDUCT OF OPERATIONS

3.13.1. This chapter of the safety analysis report should describe how the operating organization fulfils its prime responsibility for safety in the operation of a nuclear power plant in accordance with the requirements established in SSR-2/2 (Rev. 1) [4]. More specifically, the chapter should address the following:

(a) Important operational issues that are relevant to safety;
(b) The approaches adopted by the operating organization to address these issues by implementing relevant operational programmes;
(c) The provisions made by the operating organization to establish and maintain an adequate number of staff with the necessary technical competence and skills, and to provide the operating procedures to be followed to ensure protection and safety.

3.13.2. The level of detail provided in this chapter of the safety analysis report may differ significantly between different stages of the safety analysis report; the most complete information should be provided in the preliminary safety analysis report or the final safety analysis report.

Organizational structure of the operating organization

3.13.3. This section should provide a description of the structure of the operating organization and specify the functions, roles and responsibilities of the different components within it. The organization and responsibilities of review bodies (e.g. safety committees, advisory panels) should also be described. The description of the organizational structure should demonstrate that all the management functions for the safe operation of the nuclear power plant, such as policy making functions, operating functions, supporting functions and review functions, are adequately addressed. Further guidance is provided in IAEA

Safety Standards Series No. SSG-72, The Operating Organization for Nuclear Power Plants [40].

3.13.4. The description should cover the functions and responsibilities of individual organizational units and the process for the qualification of operating personnel, and should include activities such as design, manufacturing, construction, commissioning, operation, plant configuration control and decommissioning.

3.13.5. This section should also identify qualification requirements for key personnel.

Training

3.13.6. This section should provide information to demonstrate that the general qualification and training programme for plant staff is adequate to achieve and maintain the required level of professional competence throughout the lifetime of the plant. The information provided should include the initial qualification requirements, the staff training programme, refresher training and retraining, and the documentation system. The training programme and facilities, including simulator facilities should be briefly described and should reflect the status, characteristics and behaviour of the plant units. Further recommendations are provided in IAEA Safety Standards Series No. SSG-75, Recruitment, Qualification and Training of Personnel for Nuclear Power Plants [41].

3.13.7. This section should describe how a systematic approach to training is to be adopted, including reviews and updates based on operational experience and research results. The training programme should be based on an analysis of the responsibilities and tasks involved in the work, and should apply to all personnel, including managers.

3.13.8. Where the licensing regime includes provision for the licensing of operators and for personnel in other roles or positions, this section should describe the system that will be implemented and explain the provisions that will be put in place to comply with the licensing requirements.

Implementation of the operational safety programme

Conduct of operations

3.13.9. Operational safety programmes are specific programmes performed to ensure the adequate state of the plant with regard to relevant requirements for safe operation. This section of the safety analysis report should either describe such programmes or indicate the plans that are in place for their implementation in future stages of the lifetime of the nuclear power plant.

Maintenance, surveillance, inspection and testing

3.13.10. This section of the safety analysis report should provide a description of, and a justification for, the arrangements to be applied to identify, control, plan, execute and review the maintenance, surveillance, inspection and testing practices that influence reliability and affect nuclear safety.

3.13.11. The surveillance programmes should be described and should include the predictive, preventive and corrective maintenance activities that are required to be conducted (in accordance with Requirement 31 of SSR-2/2 (Rev. 1) [4]) to control the potential degradation of SSCs and to prevent failures. In addition, it should be demonstrated that the surveillance programme is adequately specified to ensure compliance with the OLCs for the plant.

3.13.12. This section should also describe the approaches and methods used to demonstrate the appropriateness of the plant inspections, including in-service inspections. Emphasis should be placed on the adequacy of the in-service inspections of the integrity of the primary and secondary coolant systems, owing to their importance to safety and the severity of the possible consequences of failure.

3.13.13. This section should describe the different types of testing that can affect the safety functions of a nuclear power plant and how it is ensured that testing is initiated, carried out and confirmed within the timescales allowed.

Core management and fuel handling

3.13.14. This section should describe how the necessary arrangements are made for operational activities associated with core management and fuel handling to ensure the safe use of the fuel in the reactor and safety in its transport and storage on the site. It should be shown that, for each refuelling batch, tests or simulations are performed to confirm that the core performance meets the safety requirements, mainly Requirement 43 of SSR-2/1 (Rev. 1) [3]. Recommendations are provided in IAEA Safety Standards Series No. SSG-73, Core Management and Fuel Handling for Nuclear Power Plants [42].

3.13.15. It should be described how the core conditions are monitored in order to remain within operational limits. In addition, it should be shown that appropriate methods have been established for dealing with defects in fuel rods or control rods, so as to minimize the amounts of fission products and activation products in the primary coolant or in gaseous effluents during normal operation.

Ageing management and long term operation

3.13.16. This section should describe all the parts of the plant that can be affected by ageing and should present the proposals made for addressing any ageing issues that have been identified, according to the safety relevance of the SSCs. The description should cover appropriate material monitoring and sampling programmes necessary to verify the ability of equipment and SSCs to fulfil their safety functions throughout the lifetime of the plant. Appropriate consideration should be given to the feedback of operating experience (see Requirement 24 of SSR-2/2 [4] and para. 3.13.20 of this Safety Guide) with respect to ageing. Recommendations are provided in IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [43].

3.13.17. The long term operation programme focused on ageing management should be described, if applicable. The description should cover the additional measures necessary to verify the capability of SSCs to fulfil their safety functions and to meet their qualification requirements during the period of long term operation.

Control of modifications

3.13.18. This section should describe the proposed method of designing, planning, executing, testing and documenting the modifications to the plant throughout its lifetime. This should take account of the safety significance of the proposed modifications to allow them to be graded and referred to the regulatory body, as necessary. Recommendations and guidance regarding plant modifications are provided in SSG-71 [12].

3.13.19. It should be confirmed in this section that the modification control process covers all safety significant changes (including permanent and temporary changes) made to SSCs, OLCs, plant procedures and process software.

Programme for the feedback of operating experience

3.13.20. This section should describe the programme that is to be implemented for the feedback of operating experience. The description should include the measures to ensure that operational events and incidents taking place at the plant and at other relevant nuclear power plants are identified, recorded, notified, investigated internally and used to incorporate, when appropriate, lessons for the operation of the plant (see Requirement 24 of SSR-2/2 (Rev. 1) [4]). The programme should include a consideration of the technical and organizational aspects and of the human factors. More detailed recommendations are provided in IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [44].

Documents and records

3.13.21. This section should provide information on the management system provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and deleting documents, records and reports relevant for the operation of the plant over its lifetime. The description should specify associated retention times in accordance with the level of importance in terms of plant licensing, operation and decommissioning. In particular, this should include the provisions for documenting the management of plant configuration, as well as the management of waste and the decommissioning of the plant (see SSG-72 [40]).

Outages

3.13.22. This section should provide a description of the relevant arrangements for conducting periodic shutdowns of the reactor. A description of how the plant configuration is maintained in accordance with OLCs and the safety analysis report should also be provided in this section. Attention should be paid to the measures that need to be taken to ensure radiation protection and safety in specific circumstances during the outages. Such circumstances include the organization and planning of multiple activities and personnel from different fields and services while dealing with time pressures, and the management of unforeseen events. The feedback of operating experience and how it has been analysed and incorporated to improve the management of outages should also be described.

Plant procedures and guidelines

Administrative procedures

3.13.23. This section should describe all the relevant documents that will be used by plant staff to ensure that procedures and guidelines for normal operation, anticipated operational occurrences and accident conditions are followed in the intended manner. It is not expected that detailed written procedures will be included. However, depending on the stage of the project, this section of the safety analysis report should either describe the preliminary arrangements and schedules for the preparation of such procedures and guidelines or should provide a brief description of the nature and content of the procedures and guidelines. The categories of procedures and guidelines that should be covered are described below.

Operating procedures

3.13.24. This section should provide a description of the structure of the plant operating procedures. The information presented should be sufficient to demonstrate that the operating procedures are (or will be) developed to ensure that the plant is operated within the OLCs. The description should include the operating procedures for normal operation, providing instructions for the safe conduct of all operating modes, such as startup, power operation, shutting down, cooldown, shutdown, load changes, maintenance, testing, process monitoring and refuelling.

Procedures and guidelines for operating the plant during accidents

3.13.25. This section should provide a description of the procedures that will be used by the operating organization in anticipated operational occurrences, in accident conditions and in other accident scenarios. Event based approaches and symptom based approaches can be used; a justification of the approach that has been selected should be provided. The operator actions required to diagnose and to deal with accident conditions should be covered appropriately.

3.13.26. The approach used for verification and validation of the procedures should be presented, including, where applicable, human factors. The description should demonstrate that the procedures are applicable to the representative set of scenarios (anticipated operational occurrences, accident conditions and scenarios not covered by safety analyses regardless of their probability of occurrence). Links to the results of the safety analysis presented in chapter 15 of the safety analysis report or to results from other analyses performed should also be included, as appropriate. More detailed recommendations on the development and implementation of emergency operating procedures are provided in SSG-54 [13].

3.13.27. This section should provide a description of the approach to accident management. The corresponding accident management procedures or guidelines developed to prevent the progression of accidents, including accidents more severe than design basis accidents, and to mitigate their consequences if they do occur, should be presented. The information provided should make reference to the overall accident management programme at the plant, if appropriate. Recommendations on the development and implementation of accident management procedures or guidelines are provided in SSG-54 [13].

3.13.28. In relevant cases, such as multiple unit events, contingencies for an alternative water and alternative electrical power supply, as well as for a degraded regional infrastructure, should be addressed. The description should confirm that severe accident management guidelines have been developed in a systematic way, with account taken of the following:

(a) The results from the severe accidents analysis for the plant;
(b) The identified vulnerabilities of the plant to such accidents;
(c) The strategies selected to deal with these vulnerabilities;
(d) The availability of the means of interconnection between units at a multiple unit site.

Nuclear safety and nuclear security interfaces

3.13.29. Nuclear security issues are usually dealt with separately, and the related documents are withheld from public disclosure. Although the plans for the physical protection of the facility (see Refs [31, 45]) are described in a separate, confidential application (or part of the application), this section of the safety analysis report should recognize the existence of such plans.

3.13.30. This section should indicate how the operating organization ensures that safety requirements and security requirements are managed in accordance with Requirement 17 of SSR-2/2 (Rev. 1) [4], that is, how safety measures and nuclear security measures are designed and applied in an integrated manner and as far as possible in a complementary manner, so that nuclear security measures do not compromise safety and safety measures do not compromise nuclear security. This includes the establishment of an effective system to address safety and nuclear security aspects in a coordinated manner and involving all interested parties, together with the identification of specific provisions important for the integration of safety and nuclear security.

CHAPTER 14: PLANT CONSTRUCTION AND COMMISSIONING

3.14.1. Chapter 14 of the safety analysis report should include a demonstration that the nuclear power plant will be suitable for service prior to entering the construction stage, in accordance with Requirement 11 of SSR-2/1 (Rev. 1) [3] and paras 6.14 and 6.15 of SSR-2/2 (Rev. 1) [4].

3.14.2. Chapter 14 of the safety analysis report should also include a demonstration that the nuclear power plant will be suitable for service prior to entering the operational stage, in accordance with paras 6.4, 6.14 and 6.15 of SSR-2/2 (Rev. 1) [4]. This chapter should describe the commissioning programme (see Requirement 25 of SSR-2/2 (Rev. 1) [4]) intended to verify and validate the plant's performance against the design prior to the operation of the plant.

3.14.3. The relationship between the plant safety demonstration and the commissioning programme should be explained. The commissioning programme should, among other things, confirm that separate plant items important to safety will perform within their specifications and ensure that the safety functions can be reliably fulfilled.

3.14.4. As part of the commissioning programme, chapter 14 of the safety analysis report should also demonstrate that operating procedures are verified and validated in accordance with para. 6.9 of SSR-2/2 (Rev. 1) [4] and that this verification and validation will be conducted with the participation of future operating personnel.

3.14.5. This chapter of the safety analysis report should also present the details of the commissioning organization — including the relevant interfaces between design organizations, construction organizations and operating organizations during the commissioning period — and any provisions for additional personnel and their interactions with the commissioning organization.

3.14.6. This chapter of the safety analysis report should also describe how qualified operating personnel at all levels will be adequately trained and directly involved in the commissioning process. The processes established for the operating organization to develop and approve test procedures, to control the performance of tests and to review and approve test results should be described in detail. This should include the actions to be taken when the outcomes of the tests do not fully meet the design requirements.

Specific information to be included in the safety analysis report prior to construction

3.14.7. The specific information provided in the safety analysis report prior to plant construction should include the following:

(a) A description of the construction programme, including the major stages and milestones;
(b) A description of the main organizations and contractors that will manage, supervise or execute the construction;
(c) The plans for the utilization of information from (recent) plant construction experiences;
(d) A description of the arrangements to ensure quality of the construction and compliance with regulatory requirements and associated regulatory guidance;
(e) A description of the arrangements to ensure that the as-built plant conforms to the information provided in the safety analysis report and arrangements to feed back any site adaptations for subsequent updates of the safety analysis report;
(f) A description of the operating organization's activities and arrangements to supervise the construction on the site and, when relevant, off the site;
(g) A description of the major stages of the initial test programme and discussion of the overall test[14] objectives and general prerequisites for each major stage of the test programme;
(h) A description of the pre-operational stage and/or commissioning planned for each new, unique or special design feature, including a specification of the test method and test objectives;
(i) The plans for how the applicable regulatory requirements and associated regulatory guidance will be followed in the development and conduct of the initial test programme and in the development of the inspection schedule prior to initial fuel loading;
(j) The plans for the utilization of information from plant operating experience to establish where special emphasis might be warranted in the test programme;

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[14] At the construction stage, prior to non-nuclear commissioning, and for each SSC, tests include vendor inspections, welding inspections, leaktightness tests and pressurized tests for the pressure boundary, and fuel assembly inspections at the fuel fabrication facility and at the nuclear power plant.

(k) A description of the overall schedule, relative to the expected initial fuel loading, for developing and conducting the major stages of the test programme;
(l) The plans pertaining to the trial use of plant operating procedures and emergency procedures during the initial test programme;
(m) The general plans for the assignment of additional personnel to supplement plant operating personnel and technical staff during each major stage of the test programme.

Specific information to be included in the safety analysis report prior to commissioning

3.14.8. The specific information provided in the safety analysis report prior to plant commissioning should include (updated) information on the following:

(a) A description of the major stages of the commissioning programme and the specific objectives to be achieved for each major stage, including the following:
    (i) Non-nuclear testing, comprising individual pre-operational tests, overall pre-operational systems tests, structural integrity tests and integrated leakage tests for the containment and for the primary system and secondary system;
    (ii) Nuclear testing, comprising initial fuel loading, subcritical tests, initial criticality tests, low power tests and power ascension tests.
(b) A description of the organizational units and any external organizations or other personnel that will manage, supervise or execute any stages of the commissioning programme.
(c) A description of the system that will be used to develop, review and approve individual commissioning procedures by the operating organization, including the organizational units or personnel that are involved and their responsibilities.
(d) A description of the administrative controls that will govern the conduct of each major stage of the commissioning programme.
(e) The measures to be established for the review, evaluation and approval by the operating organization of commissioning results for each major stage of the programme.
(f) Baseline data for equipment and systems for future reference.
(g) The requirements pertaining to the management and disposal of records relating to commissioning procedures and test data following completion of the commissioning programme.

(h) The list of regulatory requirements and associated regulatory guides applicable to the initial commissioning programmes that will be used or a description of the alternative methods that will be used along with a justification for their use.
(i) The programme for utilizing information from plant operating experience in the development of the initial commissioning programme, including identification of the participating organizations in the programme, and a summary description of their qualifications.
(j) The schedule for the development of plant procedures as well as a description of how, and to what extent, the plant operating procedures and emergency operating procedures will be used and tested during the initial commissioning programme.
(k) A description of the procedures that will guide the initial fuel loading and the initial criticality, including the protection and safety measures to be established for safe operation.
(l) The schedule, relative to the initial fuel loading, for conducting each major stage of the commissioning programme, including the complete inspection schedule.
(m) Brief descriptions of all the commissioning tests that will be conducted during the initial commissioning programme, with emphasis on safety systems and safety features that are relied on for the following:
    (i) The safe shutdown and cool down of the plant in operational states and accident conditions;
    (ii) Conformance with OLCs that will be established by the technical specifications;
    (iii) The prevention or mitigation of the consequences of anticipated operational occurrences and accident conditions.
(n) A summary of the individual programmes implemented in each of the main stages of the commissioning programme, including an assessment of the achievement of test objectives.

CHAPTER 15: SAFETY ANALYSIS

3.15.1. Chapter 15 of the safety analysis report should provide a description of the safety analyses performed to assess the safety of the plant in normal operation and in response to postulated initiating events and accident scenarios on the basis of established acceptance criteria. These analyses include deterministic safety analyses of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, including considerations relating to the event sequences to be ‘practically eliminated’, as well as the probabilistic safety assessment. Analyses to justify specific operator actions can also be included in this chapter of the safety analysis report. The results of these analyses are typically used as a basis for the development of the plant operating procedures and guidelines.

3.15.2. The description of the analyses and the associated assumptions provided in this chapter of the safety analysis report may be supported by reference material, where necessary. The level of detail provided in this chapter should increase as the nuclear power plant project develops from the siting stage through the construction stage to the commissioning and operation stages.

3.15.3. The scope of information provided in chapter 15 of the safety analysis report should reflect the requirements on safety analysis relevant for nuclear power plant design, in particular Requirements 16, 17, 19, 20 and 42 of SSR-2/1 (Rev. 1) [3] and Requirements 14–21 of GSR Part 4 (Rev. 1) [2]. Recommendations and guidance on deterministic safety analysis are provided in IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [46]; recommendations on probabilistic safety assessment are provided in IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [47], and IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [48].

3.15.4. The information provided in this chapter of the safety analysis report should be sufficient to justify and confirm the design basis for items important to safety and to ensure that the overall plant design is capable of meeting the established acceptance criteria, in particular the dose limits and the authorized limits for radioactive releases associated with each plant state, and that the consequences of accidents are as low as reasonably achievable.

3.15.5. The level of detail in chapter 15 of the safety analysis report should provide sufficient information to enable an independent verification of the safety analyses, as required by Requirement 21 of GSR Part 4 (Rev. 1) [2].

3.15.6. The safety analyses should, to the extent possible, be comprehensively presented in this chapter of the safety analysis report. However, certain analyses may be placed in other chapters of the safety analysis report (e.g. the analysis of loads and the consequences of internal and external hazards, the analyses of the structural capacities of different SSCs).

General considerations

3.15.7. This section should provide an introduction to the chapter on safety analysis, covering both deterministic and probabilistic analyses. This should include a description of the scope of the safety analysis and the approach adopted (i.e. conservative or realistic, as appropriate) for each plant state, from normal operation to design extension conditions with core melting.

3.15.8. This section should also explain how any previously identified generic issues and relevant operating experience have been used to enhance the quality of the safety analysis, as indicated in, for example, paras 4.7, 4.27 and 4.52 and Requirement 19 of GSR Part 4 (Rev. 1) [2].

3.15.9. Any applicable reference documents on the methodology used in the safety analysis should be introduced here. Owing to the complexity of this chapter of the safety analysis report, it is also appropriate to include a description of the structure of the whole chapter in this section.

Identification, categorization and grouping of postulated initiating events and accident scenarios

3.15.10. The approach used to identify postulated initiating events and accident scenarios for both deterministic and probabilistic analyses should be described in this section. This may include the use of analytical methods such as screening of defence in depth[15], master logic diagrams, hazard and operability analysis, and failure mode and effects analysis (see SSG-2 (Rev. 1) [46]).

3.15.11. It should be confirmed in this section that the identification of postulated initiating events and accident scenarios to be analysed has been performed in a systematic way and has led to the development of a comprehensive list of events.

3.15.12. In presenting the events, they should be subdivided into categories in accordance with their anticipated frequencies and grouped according their type (i.e. taking into account their effect on the plant). The purpose of this categorization is as follows:

(a) To justify the basis for the range of events under consideration;

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[15] The expression 'screening of defence in depth' means the systematic identification of the mechanisms that could affect the performance of safety functions and thus prevent the achievement of safety objectives at different levels of defence in depth (see Ref. [49]).

(b) To reduce the number of initiating events that require detailed analysis to a subset based on the bounding cases in each of the various event groups credited in the safety analyses in order to avoid repeating a detailed analysis for events with very similar system performance (e.g. in terms of timing, the plant systems response and radiological release fractions);
(c) To allow appropriate acceptance criteria for the safety analyses to be applied to different event groups or categories.

3.15.13. The basis for the categorization and grouping of postulated initiating events should be described and justified. In addition to normal operation, the list of scenarios to be addressed in the safety analysis report should cover anticipated operational occurrences, design basis accidents, design extension conditions without significant fuel degradation and design extension conditions with core melting. Postulated initiating events taking place in all modes of normal operation (from shutdown to low power to full power operation) should be covered, including potential events that could occur during commissioning and testing of the nuclear power plant. Since design extension conditions typically develop owing to additional multiple failures, such multiple failures that are considered to be plausible should be presented in this section.

3.15.14. The resulting list of plant specific events and accident scenarios of all types (both internal and external to the plant) should be presented in this section for all modes of normal operation (including operation at power or during shutdown and refuelling) and for other relevant plant conditions that will be analysed (e.g. manual or automatic plant control).

3.15.15. Where appropriate, interactions between the electrical grid and the plant, and interactions between different reactor units on the same site, should be considered as sources of initiating events and should be described in this section.

3.15.16. Failures that are considered as initiated in plant systems other than the reactor coolant system, such as the containers or stores for fresh or irradiated fuel and storage tanks for radioactive gaseous or liquid wastes, should also be described here.

3.15.17. Where appropriate (for consideration as sources of initiating events), the interactions between the reactor core and the spent fuel pool, as well as their mutual impact, should also be identified.

3.15.18. It should also be described how relevant internal and external hazards, of both natural and human induced origin, have been considered in the determination of postulated initiating events.

3.15.19. This section should, with reference to specific analyses presented in this safety analysis report, also list the conditions that could lead to an early radioactive release or a large radioactive release and thus need to be ‘practically eliminated’, as required by para. 5.31 of SSR-2/1 (Rev. 1) [3].

Safety objectives and acceptance criteria

3.15.20. This section should describe how specific safety analyses refer to the safety principles and objectives and the general acceptance criteria introduced in chapter 3 of the safety analysis report on the general approaches to the design of SSCs.

3.15.21. The radiological acceptance criteria relating to radiological consequences and the technical acceptance criteria relating to the integrity of barriers should be specified in this section for different categories of events and types of analysis. The information on acceptance criteria given in this section should be consistent with the more general information provided in chapter 3 of the safety analysis report.

3.15.22. If probabilistic values such as core damage frequency or large release frequency are established as acceptance criteria or safety objectives, the specific values used should also be provided in this section.

3.15.23. The selection of the acceptance criteria for individual postulated initiating events and for accident scenarios should be described in this section. The scope and conditions of applicability of each specific criterion should be clearly specified.

Human actions

3.15.24. This section should describe the approaches adopted to take into account human actions in the plant and the methods selected to model these actions in both deterministic and probabilistic analyses (see Requirement 11 of GSR Part 4 (Rev. 1) [2]). Any differences in the approach to considering human actions between the deterministic and probabilistic analyses should be described.

3.15.25. It should also be confirmed that credited human actions can be accomplished with the authorized minimum shift complement, in particular in scenarios involving external hazards affecting multiple unit plants.

Deterministic safety analyses

General description of the approach

3.15.26. This section should describe how sufficient margins have been demonstrated using a deterministic safety analysis in which acceptable approaches (i.e. conservative, best estimate or realistic; see SSG-2 (Rev. 1) [46]) have been applied, and how in the case of best estimate analysis the uncertainties in both the computer codes and the input data have been taken into account.

3.15.27. The computer codes used for the deterministic analyses should be briefly described. The version number of each computer code used should be specified with reference to the relevant supporting documentation. If a set of codes is used, the method used for combining or for coupling these codes should be described.

3.15.28. This section should include a brief demonstration of the applicability of the computer code to the particular analysis. In particular, a summary of the scope of verification and validation of the computer codes should be presented, with references to more detailed reports.

3.15.29. The plant models (including nodalization schemes) used for the deterministic analyses, as well as the assumptions made concerning plant parameters, the operability of systems and the operating organization's actions (if any), should be described in this section. The key validations of the plant model (including an assessment of the convergence of nodalization and physical models) should also be summarized. Sufficient information on the plant data used for the development of the plant models should be provided to enable independent verification of the safety analysis (see Requirement 21 of GSR Part 4 (Rev. 1) [2]).

3.15.30. The main simplifications made in developing the plant models should be described and justified. The set of assumptions used in the deterministic safety analyses performed for different types of scenario should also be described in this section.

3.15.31. Any additional guidelines (e.g. on the choice of operating states of systems or support systems, conservative time delays, and operator actions) for the development of the plant models should be described or referred to in this section.

Analysis of normal operation

3.15.32. This section should demonstrate that normal operation can be carried out safely, and hence it should confirm the following:

(a) Radiation doses to members of the public due to planned discharges or releases of radioactive material from the plant are below the dose limits and kept as low as reasonably achievable, as required by para. 2.6 of SSR-2/1 (Rev. 1) [3].
(b) Plant parameters in normal operation are maintained within the boundaries specified by the relevant OLCs, and a reactor trip or initiation of the control and limitation systems and safety systems would be avoided.

3.15.33. All possible regimes of normal operation should be covered in this description, with particular attention to transient operational regimes, such as changes in reactor power, reactor shutdown from power operation, reactor cooling down, mid-loop operation, handling of irradiated fuel, and off-loading and transfer of irradiated fuel from the reactor to the spent fuel pool.

Analysis of anticipated operational occurrences and design basis accidents

3.15.34. This section should provide the assumptions used and the results obtained from the analyses of postulated initiating events belonging to the categories of anticipated operational occurrences and design basis accidents. This section should contain sufficient information to confirm the adequacy of the design of the nuclear power plant systems and components, and of the envisaged operator actions, by demonstrating compliance with the associated acceptance criteria.

3.15.35. This section of the safety analysis report may be further subdivided into different sections for anticipated operational occurrences and for design basis accidents.

3.15.36. The analyses presented in this section should cover events taking place in the reactor coolant system during normal operation, including low power and shutdown modes. The analyses of events associated with spent fuel pools and radioactive waste management systems are covered in separate sections of chapter 15 of the safety analysis report.

3.15.37. For each group of postulated initiating events, it may be sufficient to present analyses for a limited number of bounding scenarios that represent a bounding response for a group of events. The basis for selection of these bounding events should be described and the resulting selection justified.

3.15.38. The plant parameters important to the outcome of the safety analysis should be presented, including, as a minimum, all parameters important for the assessment of compliance with the selected acceptance criteria.

3.15.39. The response of plant systems to the postulated initiating events, including the operating conditions in which a system is actuated, and the associated time delays and capacity after actuation, should be presented. It should also be demonstrated that the response is consistent with the overall functional requirements for the system as described in the relevant chapter of the safety analysis report on the individual plant systems.

3.15.40. In this section it should be demonstrated that all the relevant acceptance criteria for a particular postulated initiating event are met; the results from as many specific analyses as necessary should be included in the safety analysis report.

3.15.41. For each individual group of postulated initiating events analysed, a separate subsection should be included, providing the following information:

(a) Postulated initiating event to be analysed: A description of the postulated initiating event, the category to which it belongs and the applicable acceptance criteria to be met. The selection of a bounding case with a justification for this selection should be described.
(b) Tools and methodology: A description of the computer codes and models used for the analysis.
(c) Plant parameters: The specific values of important plant parameters and initial conditions used in the analysis, with an indication of the reference (nominal) values and the uncertainties associated with the parameters. An explanation should be provided of how these values have been chosen and the degree to which they are conservative for the specific postulated initiating event or scenario being analysed. In cases in which an approach involving the quantification of uncertainties is selected, the ranges and probability distribution of parameters should be specified and justified.
(d) Availability of systems (e.g. control and limitation systems, active and passive safety systems) and operator actions: A detailed description of the plant operating configuration prior to the occurrence of the postulated initiating event. This description should include information on the

availability of systems (including consideration of the worst single failure in safety systems) and operator actions that are credited in the analysis. Any assumptions regarding the availability of systems and operator actions should be consistent with established conservative assumptions regarding the operability of different plant systems in accordance with the rules for conservative safety analysis, described in SSG-2 (Rev. 1) [46].

(e) Analysis assumptions and treatment of uncertainties: Information on any additional failures in nuclear power plant systems and components postulated to occur in the specific accident scenario and any other conservative assumptions.

(f) Plant response assessment: A description of the modelled plant behaviour, highlighting the timing of the main events (i.e. initial event, any subsequent failures, times at which various safety groups are actuated and the time at which a safe, long term, stable state is achieved). Individual system actuation times, including the reactor trip time and the time of operator intervention, should be provided. Key parameters should be presented as functions of time during the event. The parameters should be selected so that a complete picture of the event's progression can be obtained within the context of the acceptance criteria being considered. Any abrupt or otherwise unexpected changes of parameters should be explained. The results should present the relevant plant parameters and a comparison with the acceptance criteria, with a final statement on the acceptability of the result. The status of physical barriers and the fulfilment of the safety functions should also be described.

(g) Assessment of radiological consequences: The results of the assessment of the radiological consequences of a given event, if applicable. The key results should be compared with the radiological acceptance criteria. The analysis of radiological consequences can be presented together with other results in a common section for each relevant postulated initiating event analysed, or it can be placed in a separate section together with all the design basis accident analyses that show radiological consequences, with an appropriate selection of bounding cases for different categories of events.

(h) Sensitivity studies and uncertainty analyses: The sensitivity studies and uncertainty analyses that have been performed (when necessary, as described in SSG-2 (Rev. 1) [46]) should be presented to demonstrate the robustness of the results and to support the conclusions of the accident analyses.

(i) Assessment of generated radioactive waste: Whenever relevant, the amount and composition of radioactive waste generated in the management of a given event should be described (see para. 3.11.3).

3.15.42. To support the demonstration of the independence between levels of defence in depth and, in particular, the robustness of the design in anticipated operational occurrences, the safety analysis report should also include a realistic analysis of certain anticipated operational occurrences. The main objective should be to demonstrate that the plant systems (in particular control and limitation systems) can prevent anticipated operational occurrences from evolving into accident conditions and that the plant can return to normal operation following an anticipated operational occurrence. Detailed guidance for performing a conservative and realistic analysis of anticipated operational occurrences is provided in SSG-2 (Rev. 1) [46].

Analysis of design extension conditions without significant fuel degradation

3.15.43. This section should present the assumptions used and the results obtained from the analyses of design extension conditions without significant fuel degradation for accidents taking place in the reactor coolant system. The analyses presented in this section should demonstrate with an adequate level of confidence that core melting can be prevented and that there are adequate margins to avoid cliff edge effects.

3.15.44. The scope and content of the information provided should be similar to that described above for design basis accidents, with account taken of the main differences in approaches to safety analysis, in particular the use of a best estimate approach, as described in SSG-2 (Rev. 1) [46].

Analysis of design extension conditions with core melting

3.15.45. This section should present the assumptions used and the results obtained from the analyses of design extension conditions with core melting with subsequent releases of radioactive material to the containment. The analyses presented in this section should identify the most severe plant parameters resulting from the core melt sequences and demonstrate the following:

(a) The plant can be brought into a state where the functions of the containment can be maintained in the long term.
(b) The plant SSCs (e.g. the containment design) are capable of avoiding an early radioactive release or a large radioactive release, including containment bypass.
(c) Compliance with the acceptance criteria is achieved by features implemented in the design and by the implementation of severe accident management guidelines.

(d) The possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated'[16].

3.15.46. The scope and content of the information provided for this category of design extension conditions should be similar to that described above for design basis accidents, with account taken of the main differences in approaches to safety analysis, as described in SSG-2 (Rev. 1) [46].

3.15.47. This section should include a description of the physical and chemical processes and phenomena (both in-vessel and ex-vessel) that might occur during the progression of design extension conditions with core melting and how these phenomena affect the performance of the containment.

3.15.48. The information provided should address the impact of the most challenging conditions and demonstrate that the established acceptance criteria are met.

Analysis of postulated initiating events and accident scenarios associated with the spent fuel pool

3.15.49. This section should present the safety analysis performed for postulated initiating events specifically initiated in the spent fuel pool. Specific operating modes considered relating to fuel handling (e.g. emergency core unloading) should also be addressed. It should be demonstrated that the relevant acceptance criteria (usually more restrictive than the criteria relating to events initiated in the reactor coolant system) regarding maintaining subcriticality, heat removal, structural integrity, shielding and the confinement of radioactive gases released from irradiated fuel in the spent fuel pool are complied with. The information presented should contribute to the confirmation that accidents with significant fuel degradation in the spent fuel pool have been 'practically eliminated'.

3.15.50. The scope and content of the information provided should be similar to that described above for design basis accidents and for design extension conditions without significant fuel degradation, with account taken of differences in the systems involved, the large thermal inertia of the spent fuel pool, more stringent acceptance criteria and specific pathways for releases of radioactive material.

[16] Conditions that are considered 'practically eliminated' are not part of design extension conditions (see SSG-2 (Rev. 1) [46]).

Analysis of radioactive releases from a subsystem or component

3.15.51. This section should present the safety analysis performed for postulated initiating events caused by the release of radioactive material from a subsystem or component (typically from systems for the treatment or storage of radioactive waste), from minor leakage from a radioactive waste system to the overheating of, or damage to, used fuel in transit or storage, or a large break in a gaseous or liquid waste treatment system.

3.15.52. The scope and content of the information provided should be similar to that described above for design basis accidents, with account taken that the main focus of the analysis is on the dispersion of radioactive material in the environment rather than on the analysis of processes inside the nuclear power plant.

Analysis of internal and external hazards

3.15.53. The analysis of all relevant site specific internal and external hazards (if not already covered in other chapters of the safety analysis report) should be presented in this section for the hazards specified in chapter 3.

3.15.54. The information provided on the analysis of hazards should show (if not already covered in other chapters of the safety analysis report) that a hazard can be screened out owing to its negligible likelihood, that the nuclear power plant design is robust enough to prevent the associated load from developing into an initiating event, or that the hazard causes an initiating event (or a combination of them) already considered in the analysis of postulated initiating events.

3.15.55. The information provided on the analyses should be subdivided into hazards initiated inside the nuclear power plant (internal hazards), external hazards caused by natural events and external hazards initiated by human activities, and should include the engineering tools used for each kind of hazard.

3.15.56. The analysis of hazards presented in this section should, in general, cover design basis hazards. For external hazards of natural origin, the analysis should also cover hazards exceeding those considered for the design basis and should verify that there are adequate margins to avoid cliff edge effects leading to an early radioactive release or a large radioactive release (see para. 5.21A of SSR-2/1 (Rev. 1) [3]).

Probabilistic safety assessment

3.15.57. This section should summarize the scope of the complete probabilistic safety assessment study, the methods used and the results obtained, covering both Level 1 and Level 2 studies, including a consideration of events in the spent fuel pool and associated hazards, as applicable. The complete probabilistic safety assessment study should be made available to the regulatory body as a separate report, if required.

General approach to probabilistic safety assessment

3.15.58. This section should describe and justify the scope of the probabilistic safety assessment. The methodology and computer codes that have been used should also be described. Sources of important input data should be introduced with a justification of their use. If any quantitative probabilistic safety criteria or goals have been used in the development of the plant design, these should also be referred to in this section.

3.15.59. The basic data used for the assessment, together with their associated uncertainties, should be provided, including an assessment of the frequency of initiating events, component reliability, common cause failure probabilities and human error probabilities.

Results of the Level 1 probabilistic safety assessment

3.15.60. The methods used and the results of the Level 1 probabilistic safety assessment (see SSG-3 [47]) should be summarized in this section. This should include the probabilistic safety assessment modelling, including accident sequence and system modelling, human reliability analysis, dependence analysis and classification of accident sequences into plant damage states.

3.15.61. The results of the Level 1 probabilistic safety assessment and their associated uncertainty should also be provided, including an analysis of the most important contributors to the frequency of fuel damage for all the plant modes of operation and for all internal and external events included in the scope of the probabilistic safety assessment.

Results of the Level 2 probabilistic safety assessment

3.15.62. The methods used and the results of the Level 2 probabilistic safety assessment (see SSG-4 [48]) should be summarized in this section, focusing on the expected magnitude (i.e. the source term) and the frequency of radioactive releases to the environment as a consequence of core melting, together with a suitable uncertainty analysis.

3.15.63. The results of the plant damage state analysis, which provides a structured interface between the Level 1 and Level 2 probabilistic safety assessments, should be presented. It should be described how the plant damage state is used as an input to the containment behaviour analysis performed by means of a containment event tree model.

3.15.64. The main results of the containment performance analyses (i.e. from the containment event trees evaluation) and the source term evaluations should be summarized in this section.

Probabilistic safety assessment insights and applications

3.15.65. A summary of the results of the probabilistic analyses should be described in this part of the safety analysis report. An assessment of compliance with established probabilistic acceptance criteria or goals, if relevant, should be made. The results should be presented in such a manner that they clearly convey the quantitative risk measures and the aspects of the plant design that are the most important contributors to these risk measures. The intended use of probabilistic safety assessment to support the plant design and future plant operation should also be described.

3.15.66. The insights provided by the probabilistic safety assessment with respect to achieving a balanced design (see para. 5.76(a) of SSR-2/1 (Rev. 1) [3]), prevention of 'cliff edge effects' (see para. 5.76(b) of SSR-2/1 (Rev. 1) [3]) and demonstration that plant event sequences that would lead to an early radioactive release or a large radioactive release can be considered 'practically eliminated', should be summarized.

Summary of results of the safety analyses

3.15.67. This section should provide a summary of the overall results of the safety analyses, for each of the categories of events and covering both deterministic analysis and probabilistic analysis.

3.15.68. This section should confirm that the requirements for safety analysis relevant to nuclear power plant design (i.e. mainly those established in SSR-2/1 (Rev. 1) [3] and GSR Part 4 (Rev. 1) [2]) have been met in every respect, providing justification if those requirements have been revised, or have been applied with changes as a result of further considerations. In the latter cases, any compensatory measures taken to meet the revised safety requirements should be specified.

CHAPTER 16: OPERATIONAL LIMITS AND CONDITIONS FOR SAFE OPERATION

3.16.1. Chapter 16 of the safety analysis report should describe the plant OLCs. It should demonstrate that these OLCs will ensure compliance with Requirement 6 of SSR-2/1 (Rev. 1) [3] and that they include all the required components described in para. 5.44 of SSR-2/1 (Rev. 1) [3].

3.16.2. Chapter 16 of the safety analysis report should also document that the OLCs have been established in accordance with Requirement 6 and para. 4.6 of SSR-2/2 (Rev. 1) [4]. In particular, it should confirm that the OLCs are consistent with the design and with the relevant safety analyses; that proper measures are taken to ensure operation in compliance with OLCs; that the staff are properly trained to be familiar with the OLCs; that deviation from OLCs are evaluated, documented and reported as required; and that OLCs are regularly reviewed and revised.

3.16.3. The OLCs form an important part of the basis on which the operating organization is authorized to operate the plant; further guidance is provided in IAEA Safety Standards Series No. SSG-70, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants [50]. The OLCs should be presented either as part of the safety analysis report in this chapter or as a separate document that is referenced in the safety analysis report.

Scope and application

3.16.4. This section should describe the scope and range of applicability of the OLCs. The OLCs are typically presented in the form of the following:

(a) Safety limits;
(b) Safety systems settings;
(c) Limits and conditions for normal operation;
(d) Surveillance and testing requirements;
(e) Action statements for deviations from normal operation.

These OLCs are formally derived from the limiting plant configuration, with account taken of all plant states, and from the acceptable range of operating parameters justified in relevant chapters of the safety analysis report, in particular chapter 15. This is to ensure that the operation of the plant will at all times be within the safe operating regime established for the plant.

Bases for development

3.16.5. In this section it should be demonstrated how the OLCs have been developed. In particular, it should be confirmed that the OLCs are based on the safety analyses of the plant and its environment in accordance with the provisions made in the design. The justification for each of the OLCs should include any relevant background information. Amendments to OLCs should be incorporated, as necessary, as a result of testing carried out during commissioning or modifications performed on the plant during operation.

Safety limits

3.16.6. The detailed OLCs for safe operation should be included in this section, with limiting values of important parameters and operability conditions of systems and components.

Requirements for limits and conditions for normal operation, surveillance and testing

3.16.7. The requirements for surveillance, maintenance and repair to ensure that the important parameters for normal operation remain within acceptable limits and that systems and components are operable should be specified and described in this section. Where appropriate, such requirements should be justified with account taken of insights from a probabilistic safety assessment. The actions to be taken if the OLCs are not fulfilled should also be clearly described.

Administrative requirements

3.16.8. In some cases, essential administrative aspects, such as the minimum shift composition and the frequency of internal reviews, may also be covered by the OLCs. The reporting requirements for operational events and the administrative requirements, together with a demonstration of how these requirements are met, should be described in this section.

CHAPTER 17: MANAGEMENT FOR SAFETY

3.17.1. Chapter 17 of the safety analysis report should describe the overall management of all safety related activities to ensure compliance with Principle 3 of SF-1 [21] regarding leadership and management for safety. The information provided in this chapter should cover establishing, assessing, sustaining and continuously improving effective leadership and management for safety. The information provided should be sufficient to enable the verification of compliance with IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [51].

3.17.2. The description of the management system that is given in the safety analysis report for each stage of the plant lifetime (from siting to decommissioning) should reflect the differences in scope and focus of the management system that occur in the different stages of the plant lifetime, as described in appendices III to VIII of IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [52].

3.17.3. The information provided in this chapter of the safety analysis report should demonstrate that the responsibilities of the operating organization have been established in accordance with Requirements 1–3 of SSR-2/1 (Rev. 1) [3] (in relation to the management of safety in design) and Requirements 1, 5, 8 and 9 of SSR-2/2 (Rev. 1) [4] (in relation to the management of operational safety). Recommendations and guidance on meeting these requirements are provided in IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [53], and GS-G-3.5 [52].

3.17.4. Chapter 17 of the safety analysis report should describe the different management processes aimed at ensuring the overall objectives for safety, and how these processes are established, controlled, monitored and reviewed, while ensuring that safety is given the highest priority.

General characteristics of the management system

3.17.5. This section should describe how goals, strategies, plans and objectives are established by the organization, consistent with the organization's safety policy.

3.17.6. This section should provide an overall description of the management system, starting from the high level objectives, together with an explanation of how the management system is addressed in different levels of plant documentation.

3.17.7. This section should also describe how the management system ensures effective coordination between the site management, the corporate structure, technical support organizations and other organizational units of the operating organization. The description should explain how effective management control of the design and operation will be achieved so as to promote safety.

3.17.8. This section should describe how the management system integrates its elements — including safety, health, environmental, security, quality, human and organizational factor, societal, and economic elements — so that safety is not compromised, in accordance with Requirement 6 of GSR Part 2 [51].

Specific elements of the management system

3.17.9. This section should describe the overall accountability for the management system and the assignment of individuals for the coordination, development, application and maintenance of the management system.

3.17.10. This section should describe how processes and activities will be developed and effectively managed to achieve the organization's goals without compromising safety, in accordance with Requirement 10 of GSR Part 2 [51].

3.17.11. This section should also describe how other relevant factors of the management system, such as the application of the graded approach and the management of resources, will be addressed in the management system in accordance with Requirements 6, 7 and 9 of GSR Part 2 [51].

Quality management

3.17.12. This section should specifically describe those processes of the management system that are intended to ensure the quality of safety classified SSCs as applicable in different stages of the lifetime of the nuclear power plant (see SSG-30 [23]).

Measurement, assessment and improvement of the management system

3.17.13. This section should describe how the effectiveness of the management system will be monitored and assessed, including all processes and arrangements made to ensure continuous improvement, in accordance with Requirement 13 of GSR Part 2 [51]. The description of the arrangements should include internal and external audits performed periodically and other types of independent evaluation.

Fostering a culture for safety

3.17.14. This section should describe how the management system establishes the framework to foster and sustain a culture for safety, in accordance with Requirement 12 of GSR Part 2 [51], with due consideration of the attributes of a strong safety culture given in GS-G-3.5 [52].

3.17.15. This section should describe how senior management plans to regularly undertake assessments of leadership for safety and of safety culture in its own organization and to ensure that self-assessment of leadership for safety and of safety culture includes assessment at all organizational levels and for all functions in the organization, in accordance with Requirement 14 of GSR Part 2 [51]. This section should also describe how senior management plans to ensure that self-assessment makes use of recognized experts in the assessment of leadership and of safety culture and that independent assessment of leadership and of safety culture is conducted for enhancement of the organizational culture for safety.

3.17.16. This section should also include a description of how senior management plans to use the results of the assessment of the management system in the enhancement of the organizational culture for safety.

CHAPTER 18: HUMAN FACTORS ENGINEERING

3.18.1. Chapter 18 of the safety analysis report should describe the human factors engineering programme and its application to the plant design to meet Requirement 32 of SSR-2/1 (Rev. 1) [3]; further guidance is provided in IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [54]. This programme applies to all operational states and accident conditions and to all plant locations where such interactions are anticipated. The human factors engineering considerations presented in the safety analysis report should, as a minimum, cover the following:

(a) The arrangements for the management of the human factors engineering programme, including the allocations of authority and oversight in the design process;
(b) The human factors analysis methods that are applied;
(c) The assumptions used in the choice of human–machine interface design, with account taken of human factors engineering;

(d) Human factors verification and validation, including the identification and resolution of human factors engineering issues that are identified during the design stage and the assumptions made during analyses;
(e) A description of how human–machine interface design has been implemented in the overall plant design;
(f) A description of the strategy for monitoring human performance for safety critical tasks.

3.18.2. This chapter of the safety analysis report should provide information on how human capabilities and limitations were taken into account in the design of the nuclear power plant to support the performance of tasks by plant personnel.

3.18.3. Although this chapter of the safety analysis report should comprehensively cover the issues associated with human factors, such factors should also be considered in other chapters of the safety analysis report, including those relevant for siting (chapter 2), instrumentation and control (chapter 7), radiation protection (chapter 12), operation (chapter 13), safety analysis (chapter 15), management systems (chapter 17), emergency preparedness and response (chapter 19) and decommissioning (chapter 21).

Management of the human factors engineering programme

3.18.4. This section should outline the processes in the human factors engineering programme (i.e. analyses, design of the human–machine interfaces, and evaluations such as verification and validation) and the inputs and outputs for these processes.

3.18.5. This section should describe the following:

(a) The integration of human factors engineering with other plant design or modification activities;
(b) The coordination required between responsible personnel and project and design authorities, and between different disciplines, to perform human factors engineering activities;
(c) The process for communicating the outputs of analyses to the responsible engineering disciplines and for ensuring that the outputs have been addressed;
(d) The organization and competencies necessary for integrating human factors engineering into the design;
(e) The framework for documenting and tracking human factors engineering issues that are identified by the human factors engineering processes;

(f) The responsibilities and authorities within the human factors engineering team regarding the integration of human factors engineering aspects into the design.

Human factors engineering analysis

Review of operating experience

3.18.6. This section should describe the review of operating experience, how it was used to identify and analyse human factors engineering issues relating to safety, and how this was documented.

Function analysis and function allocation

3.18.7. This section should describe the function analysis for all plant states to demonstrate that the functions necessary to accomplish safe operation are sufficiently well defined and properly analysed.

3.18.8. This section should describe the allocation of functions for all plant states to demonstrate that the functions necessary to accomplish safe operation are sufficiently well defined and properly analysed.

Task analysis

3.18.9. This section should describe the approach to task analysis for groups of operating personnel relevant to the task being analysed (e.g. operators of the reactor, operators of the turbines, shift supervisors, field operators, safety engineers, operation and maintenance staff). The tasks described should cover all plant states.

3.18.10. This section should describe specific tasks that are necessary for the fulfilment of a safety function in different locations (e.g. the main control room, the supplementary control room, local control stations, emergency response facilities or locations), identified for all plant states and for all plant operation modes. The section should also consider all relevant groups of operating personnel, including those listed in para. 3.18.9.

3.18.11. The scope of the task analysis should be described in this section, including how representative important human tasks (i.e. maintenance, testing, inspection and surveillance) were selected, as well as the range of plant operation modes included in the task analysis.

3.18.12. The main results of the task analysis should be described in a specific subsection.

Staffing and qualifications

3.18.13. This section should describe the analysis of staffing and staff qualifications, as well as the scope of the analysis performed. Consistent with the information provided in para. 3.13.1, it should demonstrate that the staffing requirements in terms of the number of personnel and their qualifications were analysed in a systematic manner, including a thorough understanding of task requirements and applicable regulatory requirements.

3.18.14. The scope of the analysis should include the number of personnel, and their qualifications, that are considered necessary for the full range of plant conditions and tasks, including operational tasks (operational states and accident conditions) and plant maintenance and testing (including surveillance testing). Any other plant personnel who perform tasks that directly relate to plant safety should also be addressed.

Treatment of important human actions

3.18.15. This section should document how important human tasks and actions were identified; how the operator tasks and actions credited in the safety analysis, including relevant factors affecting performance, were analysed; and how the ability of the design solution to ensure that human performance meets the safety requirements was confirmed.

Design of the human–machine interface

3.18.16. This section should describe the application of a structured methodology for human–machine interface design that includes the identification and selection of candidate human–machine interface approaches, the definition of a detailed design, and the performance of human–machine interface tests and evaluations, as necessary.

3.18.17. This section should also describe the process by which human–machine interface design requirements are developed, and the processes by which human–machine interface designs are identified and refined.

Human–machine interface: Design inputs

3.18.18. This section should describe how the design process for human factors engineering translates the function and task requirements into human–machine interface characteristics and functions.

Human–machine interface: Detailed design and integration

3.18.19. This section should describe how the human–machine interface provides the operating personnel with the information necessary to detect changes in system status, to diagnose the situation, to adjust the system (when necessary) and to verify manual or automatic actions.

Human–machine interface: Tests and evaluations

3.18.20. This section should describe how tests and evaluations of concept design features and detailed design features should be conducted during the process of developing human–machine interfaces to support design decisions.

Human–machine interface: Design of the main control room

3.18.21. This section should describe (consistently with chapter 7 of the safety analysis report) how the human–machine interface design provides displays and controls in the main control room for the manual, system level actuation of critical safety functions and for the monitoring of those parameters that support these functions.

3.18.22. This section should also describe how the human–machine interface design of the main control room gives due consideration to the following:

(a) The type of human–machine interface to be used in accordance with its purpose;
(b) The organization of human–machine interfaces into workstations (e.g. consoles, panels);
(c) The arrangement of workstations and supporting equipment in the main control room.

Human–machine interface: Design of the supplementary control room and emergency response facilities on the site

3.18.23. This section should describe how the human–machine interface design considers human factors engineering principles and the human characteristics of personnel under accident conditions, particularly where immediate actions are necessary.

3.18.24. This section should describe (consistently with chapter 7 of the safety analysis report) the human–machine interface design process for the supplementary control room, local control stations and emergency response facilities and how consistency with the design process for the main control room is ensured by using similar procedures, criteria and methods.

3.18.25. This section should also describe the functions of the supplementary control room, local control stations and emergency response facilities that need to be maintained for the control and monitoring of safety functions and to conduct and ensure safe shutdown in the event of internal or external hazards.

Development of procedures

3.18.26. This section should document (consistently with chapter 13 of the safety analysis report) how human factors engineering principles and criteria, along with other design requirements, are taken into account in the development of procedures that are technically accurate, comprehensive, explicit, easy to use and validated.

3.18.27. This section should describe the objectives and scope of the programme for the development of procedures and should address the following:

(a) Plant and system operations in operational states (including startup, power operation, anticipated operational occurrences and shutdown);
(b) Testing and maintenance;
(c) Response to alarms;
(d) Generic technical guidelines for emergency operating procedures;
(e) Accident management guidelines.

Training programme development

3.18.28. This section should document a systematic approach for the development of a training programme. Consistency with the general qualification and training programme for plant staff (see paras 3.13.6–3.13.8) should also be documented.

3.18.29. The overall scope of the training programme should be defined and should include the following:

(a) The full range of positions of operational personnel;
(b) All plant operational states and accident conditions;
(c) Specific operational activities (e.g. operations, maintenance, testing, surveillance);
(d) The full range of plant functions and systems, including those that are different from those of predecessor plants (e.g. passive systems and functions);
(e) The full range of relevant human–machine interfaces (e.g. main control room, supplementary control room, local control stations, emergency response facilities), including characteristics that are different from those of predecessor plants (e.g. display space navigation, operation of ‘soft’ controls).

Verification and validation of human factors engineering analysis results

3.18.30. This section should document that a verification of the human–machine interface design was performed against the task requirements identified in the task analysis. This section should also describe the criteria for this verification, including the selection of standards and guidelines for human factors engineering that were used in the review of the characteristics of the human–machine interface components.

3.18.31. This section should describe the validation concept, with account taken of the independence of this validation from the activities associated with design, test design justifications, scenario selection and criteria selection. This section should also document how the test scenarios used for validation testing allow for the assessment of the resources available to plant personnel, over appropriate lengths of time and in a meaningful number of scenarios.

3.18.32. This section should describe the main findings and conclusions of the final human factors engineering validation of the design.

Design implementation

3.18.33. The objective of this section is to document (in particular, at the stage of the final safety analysis report) how it will be verified that the as-built design conforms to the verified and validated design that resulted from the human factors engineering design process.

3.18.34. The scope should include the following:

(a) Verification and validation of design aspects that cannot be completed as part of the verification and validation programme for the human–machine interface;
(b) Confirmation that the as-built human–machine interface, procedures and training conform to the design intent;
(c) Confirmation that all human factors engineering issues in the issue tracking system are appropriately addressed (see also para. 3.18.37).

3.18.35. The final safety analysis report should describe how aspects of the design that were not addressed in the verification and validation programme will be evaluated.

3.18.36. The final safety analysis report should describe the final (as-built) human–machine interfaces, procedures and training, as well as the process for correcting any identified discrepancies in the human factors engineering design and analysis.

3.18.37. In addition, the final safety analysis report should describe the process for ensuring that all issues relating to human factors engineering that are documented in the issue tracking system will be verified as adequately addressed.

Human performance monitoring

3.18.38. This section should describe how the programme for monitoring human performance is an active and ongoing process to evaluate the continuing effectiveness of the design to properly support personnel in carrying out their work tasks safely and effectively.

3.18.39. This section should describe the objectives and scope of the programme of human performance monitoring to provide reasonable assurance that the following criteria are met during commissioning and operation:

(a) The design can be effectively used by personnel, including within the control room and between the control room, supplementary control room and other emergency response facilities.
(b) Changes made to the human–machine interfaces, procedures and training do not have adverse effects on personnel performance (e.g. changes do not interfere with skills acquired through previous training).

(c) Human actions can be accomplished within established time and performance criteria.
(d) The acceptable levels of performance established during the system validation are maintained.

CHAPTER 19. EMERGENCY PREPAREDNESS AND RESPONSE

3.19.1. Chapter 19 of the safety analysis report should provide information on emergency arrangements, demonstrating in a reasonable manner that, in a nuclear or radiological emergency, all actions necessary for the protection of workers (including emergency workers), the public and the environment could be taken, and that the decision making process for the implementation of these actions would be timely, disciplined, coordinated and effective. This chapter of the safety analysis report should cover on-site emergency arrangements for accident conditions[17] (i.e. design basis accidents and design extension conditions) that could result in harmful effects on the site and off the site warranting protective actions.

3.19.2. The description should include information on the goals of emergency response and the strategy to achieve those goals, and on the organization and management for a coordinated and effective emergency response. It should provide sufficient information to show how the relevant goals of emergency response will be met. A description of how the operating organization meets the relevant requirements of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [55], should also be provided.

3.19.3. The arrangements for liaison and coordination with on-site response organizations should be described in this section. The procedures that will be used to notify off-site notification points and to provide sufficient information for an effective off-site response in all jurisdictions should also be described.

[17] In accordance with IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [55], the operating organization should also make on-site emergency arrangements for preparedness and response for events that are beyond the design basis accidents and, as appropriate, for conditions that are beyond design extension conditions, but these arrangements are beyond the scope of the safety analysis report and this Safety Guide.

3.19.4. The on-site emergency arrangements, including programmes on training and exercises, to ensure that an adequate level of emergency preparedness and response is in place before commissioning should be described. The planned intervals for the periodic drills and exercises to maintain adequate emergency preparedness should also be described, together with a justification for the intervals chosen.

3.19.5. Further guidance and information on emergency preparedness and response are provided in IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [56]; IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [57]; IAEA Safety Standards Series No. GSG-11, Arrangements for the Termination of a Nuclear or Radiological Emergency [58]; and Ref. [59].

Arrangements for performing functions essential for the emergency response

3.19.6. This section should contain a description of the operating organization's arrangements for implementing functions that are essential for an effective emergency response (in accordance with the relevant functional requirements established in section 5 of GSR Part 7 [55]). The description should include arrangements in place to fulfil the following objectives:

(a) Execute promptly and manage safely and effectively the on-site emergency response, including the transition from normal operations to operations under emergency conditions.
(b) Classify promptly the emergency, declare the emergency class, initiate the on-site emergency response, and notify and provide sufficient information to the off-site notification points.
(c) Decide on and take necessary mitigatory actions on the site.
(d) Assess and determine, at the preparedness stage, when and under what conditions assistance from off-site emergency services may need to be provided on the site.
(e) Assess the hazards and possible development of hazardous conditions initially and throughout the emergency to inform decisions of necessary emergency response actions and take necessary urgent protective actions to protect all persons present at the site in an emergency.
(f) Ensure suitable, reliable and diverse means of communication for use in taking protective actions on the site and for communication with relevant off-site officials.

(g) Protect emergency workers responding on the site and assess hazardous conditions in which emergency workers might have to perform response functions.
(h) Communicate with the public effectively and consistently with relevant off-site response organizations.
(i) Manage radioactive waste generated in an emergency safely and effectively.
(j) Terminate the emergency on the site and provide relevant information in this regard to relevant off-site response organizations.
(k) Document, protect and preserve, to the extent practicable, data and information important for an analysis of the emergency and the emergency response.
(l) Analyse the emergency and the emergency response to identify actions to be taken to avoid other emergencies and to improve emergency arrangements.

3.19.7. Arrangements for ensuring the protection of all people present at the site (including emergency workers, non-essential personnel and visitors), and how these arrangements will be coordinated with off-site response organizations, should be described. When necessary, reference should be made to other sections of the safety analysis report where this issue is mentioned.

Emergency response facilities

3.19.8. Information should be provided about the availability of the following, in accordance with Requirement 24 of GSR Part 7 [55]:

(a) Technical support centre, operational support centre and emergency centre in which response personnel will provide advice and support to operating personnel in the control room to mitigate the consequences, decide on, initiate and/or manage the on-site response (except for the detailed control of the plant), and from which data on plant conditions will be transmitted to the emergency operations facility;
(b) Supplementary control room, which has appropriate measures to enable the control of essential safety systems;
(c) Emergency operations facility in which overall emergency response will be coordinated and data on plant conditions and on-site and off-site monitoring results will be assessed.

3.19.9. The description of emergency response facilities should include details of any equipment, communications and other arrangements necessary to support the assigned functions of these facilities and to ensure their continuous operability under accident conditions. The habitability of these facilities and the provisions to protect workers, including emergency workers, during accident conditions should also be described and justified.

Capability of the operating organization to assess potential radioactive releases in accident conditions

3.19.10. This section should provide a demonstration of how the operating organization will carry out the following:

(a) Continuously assess the conditions at the plant, including the actual or predicted levels of core damage;
(b) Predict the extent and significance of any radioactive release if an accident has occurred;
(c) When applicable, provide data and information from off-site monitoring systems to the operating organization and to the regulatory body if required by national arrangements.

3.19.11. It should be demonstrated that the response of the necessary instrumentation or systems at the plant under emergency conditions is sufficient to ensure the fulfilment of the required safety functions. A reference to other chapters of the safety analysis report that describe the equipment qualification programme may also be acceptable.

Emergency preparedness for multiple unit sites

3.19.12. If a new reactor is located on, or near, an operating reactor site with existing emergency arrangements (i.e. a multiple unit site) and the emergency arrangements for the new reactor utilize those of the operating reactor, this section should cover the following:

(a) Address the extent to which the existing on-site emergency arrangements of the operating reactor are credited for the new units, including how the existing arrangements would be able to adequately accommodate an expansion to include one or more additional reactors. It should also consider any required modifications to the existing on-site emergency arrangements (e.g. to address the issue of staffing and the potential for simultaneous accidents involving all the reactors located at the site).
(b) Describe any updates of the existing emergency arrangements, such as emergency response facilities and equipment, including notification and communication systems and support from off-site emergency services,

considering the potential for simultaneous accidents involving several reactors located at the site.
(c) If applicable, describe the training and exercise requirements for the operators of all the reactors.
(d) Describe how emergency arrangements, including the interface with nuclear security measures, are integrated and coordinated with the emergency arrangements of adjacent sites.

CHAPTER 20: ENVIRONMENTAL ASPECTS

3.20.1. Chapter 20 of the safety analysis report should provide a brief description of the approach taken to assess the impact on the environment of the construction, operation (for operational states as well as for all accident conditions) and decommissioning of the plant. The radiological environmental aspects should be included in this chapter of the safety analysis report. [18]

3.20.2. It is assumed that the overall environmental impact of the plant is covered by a dedicated environmental impact assessment report. This chapter of the safety analysis report is a link between the environmental impact assessment report and the safety analysis report itself. Depending on the stage of the project, relevant data from the environmental impact assessment report should be used in the safety analysis report or an appropriate update of the information originally covered by the environmental impact assessment should be provided. In the initial safety analysis report, the sources of information for this chapter of the safety analysis report are the relevant parts of the environmental impact assessment report. In subsequent stages of the safety analysis report, more specific information on the radiological impact of different plant states will be available in chapters 11, 12 and 15 of the safety analysis report. In this case, chapter 20 can be based on appropriate references to other chapters of the safety analysis report.

General aspects of the environmental impact assessment

3.20.3. This section provides the introduction to the chapter. In particular, it should describe the relationship between the environmental impact assessment and the status of the project. In addition, the status of reviews, approvals and consultations associated with the environmental impact assessment should be summarized.

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[18] The scope of the environmental protection aspects included in the safety analysis report is typically commensurate with national regulations.

Site characteristics that are important in terms of environmental impact

3.20.4. This section should briefly summarize each of the site characteristics (i.e. as addressed in chapter 2 of the safety analysis report) that are important in terms of environmental impact, including land, water and ecology as well as relevant data on the population distribution, geology and meteorology.

3.20.5. Requirements relating to the scope of information on site specific factors can be found in SSR-1 [5]. Further recommendations and guidance are provided in GSG-10 [15].

Plant features that minimize the environmental impact

3.20.6. All plant characteristics that determine the characteristics of radioactive releases and/or minimize the radiological impact on the environment should be summarized here, with reference made to other chapters of the safety analysis report, as appropriate.

Environmental impact of construction

3.20.7. The construction of the plant does not directly give rise to a source of radiation. However, other potential sources of radiation, such as adjacent nuclear installations or sealed radioactive sources used during the plant construction, should be considered to quantify the radiological impact of the construction of the proposed plant. The assumptions and methodology used, and the results of the impact analysis, should be described in this section.

Environmental impact of normal operation

3.20.8. The information included in this section should demonstrate compliance with all operational targets for solid, liquid and gaseous discharges and the adequacy of measures to comply with authorized limits. A description of all radiological impacts on the environment during plant operation should be provided, including the following:

(a) Direct radiation from buildings and facilities in which radioactive materials are handled;
(b) Radiation emitted by radionuclides contained in discharges of gaseous radioactive substances from devices in the controlled area;
(c) Radiation emitted by radionuclides contained in discharges of liquid radioactive substances from devices in the controlled area.

3.20.9. Further on, this section should summarize the measures that will be taken to control radioactive discharges to the environment (consistently with chapters 11 and 12 of the safety analysis report). External exposure from discharges (e.g. from radioactive gases and aerosols released from ventilation stacks and from deposition) and internal exposure from inhalation and ingestion of radionuclides should be addressed.

3.20.10. Further recommendations and guidance on methods and approaches to the assessment of the radiological impact of plant operation on the environment are provided in SSG-2 (Rev. 1) [46] and GSG-10 [15], respectively.

Environmental impact of postulated accidents involving radioactive releases

3.20.11. The environmental effects of accidents involving radioactive releases that can be postulated for the plant should be addressed in this section. The list of accidents covered should be provided. The scope of this section should cover the off-site consequences in terms of the projected effective doses at sufficient distance from the plant for design basis accidents as well as for selected design extension conditions with core melting. The type of data and information necessary will be affected by site specific and station specific factors, and the degree of detail should be modified in accordance with the anticipated magnitude of the potential impacts. An overview of the off-site protective actions to limit the radiological impacts during accidents should be provided.

Environmental impact of plant decommissioning

3.20.12. The radiological impacts of plant decommissioning on the environment should be summarized in this section (with reference made to chapter 21), using an approach similar to the one used to assess the environmental impact of normal operation (see paras 3.20.8–3.20.10).

3.20.13. Requirements for decommissioning are established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [60]. Further recommendations and guidance are provided in IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [61], and IAEA Safety Standards Series No. WS-G-5.2, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material [62].

Environmental measurements and monitoring programmes

3.20.14. This section should refer (consistently with chapter 11) to the off-site monitoring regime for contamination levels and radiation levels. This should include a description of the dedicated environmental monitoring programmes and alarm systems that are required to respond to unplanned radioactive releases and, if applicable, the automatic devices designed to interrupt such releases. All routes of possible sources of uncontrolled radiation and releases of radioactive substances should be addressed. Warning signals, automatic blockades and any other automatic measures that prevent unplanned releases should be specified, together with the activation level settings. Further guidance on environmental monitoring can be found in IAEA Safety Standards Series No. RS-G-1.8, Environmental and Source Monitoring for Purposes of Radiation Protection [63].

Records of radioactive releases and availability of information to the authorities and the public

3.20.15. This section should describe the methods to make, store, archive and retrieve records of routine radioactive releases from the site. It should also describe the measures that will be taken to make appropriate data available to the regulatory body, other authorities and the public. It should be demonstrated that the format and deadlines for these records comply with relevant regulations and any conditions specified by the regulatory body in the authorization for operation.

CHAPTER 21: DECOMMISSIONING AND END OF LIFE ASPECTS

3.21.1. This chapter should describe decommissioning as a stage in the lifetime of the plant that comes after the permanent cessation of operation (permanent shutdown) and, where applicable, a plant transition period[19]. The feasibility of decommissioning and the capability to decommission the plant should already be conceptually demonstrated during the design and construction stages, before the initial criticality occurs or before plant operation commences. This demonstration is usually provided in an initial decommissioning plan (see paras 2.8 and 2.10 and Appendix I).

3.21.2. During the siting of a nuclear power plant, the information provided in this chapter should describe how the plant design will minimize the levels

[19] The transition period refers to the period between the permanent shutdown of operations at the plant and the approval of the final decommissioning plan.

of contamination that will need to be addressed during decommissioning. Additionally, it should be described that, during the lifetime of the plant, appropriate radiological surveys will be conducted, including of the subsurface, the site water storage and drainage systems, and the groundwater; it should also be described how the records of the results of these surveys will include the levels of radioactivity that will need to be addressed during decommissioning and how records of residual radioactivity will be maintained. The safety issues associated with this residual radioactivity should be described in this chapter.

3.21.3. This chapter should describe how the initial decommissioning plan will be periodically updated during the operation of the plant, providing an increasing level of detail, introducing new information available from the plant operation, and reflecting regulatory, technical and other developments relating to decommissioning. The level of detail included in the decommissioning plan significantly increases 5–10 years prior to the expected end of the operating lifetime, when detailed planning for decommissioning begins. Where applicable, cost estimates and financial provisions for decommissioning should also be provided. Requirements for decommissioning are established in GSR Part 6 [60], and further recommendations and guidance are provided in SSG-47 [61] and WS-G-5.2 [62], respectively.

General principles and regulations

3.21.4. In addition to the general principles adopted for decommissioning, this section should provide information on the documentation required and the regulations to be followed to ensure that occupational exposures and public exposures are optimized and that the amounts of radioactive waste and other hazardous waste generated are minimized and properly managed.

Decommissioning strategy

3.21.5. This section should present the options identified and the method chosen for decommissioning. The main differences between the decommissioning options should be explained (e.g. in terms of the optimization of protection and safety, the protection of the environment, and minimizing the generation of waste, as well as technological, economic, social and other relevant factors). Options and their effects on the timing of the decommissioning process should also be described.

Facilitating decommissioning during design and operation

3.21.6. This section of the safety analysis report should briefly describe the proposed decommissioning approach, with the following aspects taken into account:

(a) Design solutions that minimize the amount of waste generated and that facilitate decommissioning;
(b) Design solutions that incorporate monitoring or leak detection capabilities to allow for earlier identification of uncontrolled radioactive releases;
(c) Consideration of the types, volumes and activities of radioactive waste generated during operation and decommissioning;
(d) Identified options for decommissioning;
(e) Anticipated technical, organizational and managerial changes that will be necessary during the transition period;
(f) Adequate documentary control and maintenance of suitable and sufficient records;
(g) Anticipated organizational changes, including provisions in place to preserve the institutional knowledge that will be necessary during the decommissioning stage.

Decommissioning plan

3.21.7. This section should present a tentative programme of decommissioning actions, including a timescale, containing the following activities (including their anticipated schedule of implementation):

(a) The development of an engineering study for decommissioning, identifying the policy and objectives.
(b) The selection of a decommissioning strategy that is consistent with the national policy on the management of radioactive waste.
(c) The planning, phasing and staging of the decommissioning process, including appropriate requirements for surveillance and updating of the safety analyses throughout the process. In multiple unit plants, phasing might create a new plant configuration where some units are in a safe configuration following permanent shutdown and others are still operating, which could involve the severing of shared services provided by shared safety and process systems.
(d) Identification of the systems, tools and equipment required during decommissioning, including those already available, and organization of the decommissioning actions.

(e) The development of a safety analysis report for decommissioning.
(f) The development of a programme for bringing the reactor to a safe condition for total or partial dismantling, including possible partial safe storage (in preparation for decommissioning) of selected units in a multiple unit plant.
(g) The development of a programme for ensuring that services (e.g. heating, electricity and water supply) will be available to support the decommissioning work.
(h) The estimation of the types and volumes of waste arising from decommissioning, including radioactive waste.
(i) A description of the waste management strategies for different types of waste and the identification of potentially reusable or recyclable material.
(j) The development of a programme for providing adequate facilities for the handling, processing, storage and transport of the radioactive waste arising during decommissioning.
(k) The provisions for physical protection, monitoring and surveillance during the decommissioning phases.
(l) The tracking of the authorization process for the conduct of decommissioning actions throughout the entire decommissioning stage.

Provisions for safety during decommissioning

3.21.8. This section should provide a short description of the measures necessary to ensure safety during decommissioning. The description should include measures adopted in the design and operation of the plant to fulfil the following objectives:

(a) To minimize the volume of radioactive structures;
(b) To reduce the toxicity of the waste;
(c) To lower the activity level of irradiated components;
(d) To restrict the spread of contamination and permit easier decontamination;
(e) To facilitate the access of personnel and machines and the removal of waste;
(f) To ensure the collection of important data.

3.21.9. An estimate of the expected volume of radioactive waste generated during decommissioning should be provided. The information provided should indicate that special attention has been paid to the following aspects:

(a) Identification of the sources of radioactive materials, including assessing their contribution to the volume of waste generated;

(b) A description of the radioactive (airborne and liquid) substances expected to be released during the decommissioning process, demonstrating that these will be minimized and will be kept within authorized limits;
(c) The practicability of adherence to the concept of defence in depth against radiological hazards during the decommissioning process.

End of life aspects of the decommissioned site

3.21.10. This section should specify the proposed end state of the site to be reached following decommissioning and site clearance works. This should include a description of the possible future use of the site and remaining facilities.

Appendix I

DEVELOPMENT OF THE SAFETY ANALYSIS REPORT IN DIFFERENT LICENSING STAGES

I.1. The key information typically included in the different chapters of the safety analysis report issued for different licensing stages of the nuclear power plant is provided in Table 1.

TABLE 1. INFORMATION INCLUDED IN THE SAFETY ANALYSIS REPORT ISSUED FOR DIFFERENT LICENSING STAGES OF THE NUCLEAR POWER PLANT

Chapter of the safety analysis report		Licensing stages		
		Site permit: Initial SAR	Construction permit: Preliminary SAR	Commissioning: Pre-operational SAR (final SAR)
1	Introduction and general considerations	Preliminary information	Final information	Verified and updated information
2	Site characteristics	Final information	Verified information	Verified and updated information
3	Safety objectives and design rules for structures, systems and components	General design requirements	Design requirements specific to the reactor type	Verified and updated information
4	Reactor	Description of an envelope and general requirements for a given part of the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
5	Reactor coolant system and associated systems	Description of an envelope and general requirements for a given part of the design or SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information

TABLE 1. INFORMATION INCLUDED IN THE SAFETY ANALYSIS REPORT ISSUED FOR DIFFERENT LICENSING STAGES OF THE NUCLEAR POWER PLANT (cont.)

Chapter of the safety analysis report		Licensing stages		
		Site permit: Initial SAR	Construction permit: Preliminary SAR	Commissioning: Pre-operational SAR (final SAR)
6	Engineered safety features	General requirements for the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
7	Instrumentation and control	General requirements for the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
8	Electrical power	General requirements for the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
9	Auxiliary systems and civil structures	General requirements for the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
10	Steam and power conversion systems	General requirements for the design of SSCs	Description of SSCs and requirements for the operation of systems	Verified and updated information
11	Management of radioactive waste	General requirements for the design of SSCs	Description of source terms, SSCs and requirements for the operation of systems	Verified and updated information

TABLE 1. INFORMATION INCLUDED IN THE SAFETY ANALYSIS REPORT ISSUED FOR DIFFERENT LICENSING STAGES OF THE NUCLEAR POWER PLANT (cont.)

Chapter of the safety analysis report		Licensing stages		
		Site permit: Initial SAR	Construction permit: Preliminary SAR	Commissioning: Pre-operational SAR (final SAR)
12	Radiation protection	General requirements for radiation protection	Demonstration of compliance with the requirements	Verified and updated information
13	Conduct of operations	General requirements for the conduct of operations	Demonstration of compliance with the requirements	Verified and updated information
14	Plant construction and commissioning	General requirements for commissioning	Demonstration of compliance with the requirements	Demonstration of compliance with the requirements
15	Safety analysis	General requirements for the scope, methods and criteria for safety analysis	Demonstration of compliance with the requirements	Verified and updated demonstration of compliance with the requirements
16	Operational limits and conditions for safe operation	General requirements for operational limits and conditions	Description and specification of operational limits and conditions	Verified and updated description and specification of operational limits and conditions
17	Management for safety	General requirements of the management system	Description of the management system	Updated description of the management system
18	Human factors engineering	General requirements for human factors engineering	Description of the scope, methodology and results of human factors engineering	Updated description of human factors engineering

TABLE 1. INFORMATION INCLUDED IN THE SAFETY ANALYSIS REPORT ISSUED FOR DIFFERENT LICENSING STAGES OF THE NUCLEAR POWER PLANT (cont.)

Chapter of the safety analysis report		Licensing stages		
		Site permit: Initial SAR	Construction permit: Preliminary SAR	Commissioning: Pre-operational SAR (final SAR)
19	Emergency preparedness and response	General requirements for the emergency preparedness	Description of emergency facilities and emergency plans	Updated description of the emergency facilities and emergency plans
20	Environmental aspects	Preliminary or expected information, consistent with the report on the environmental impact assessment	Updated information, referring to other parts of the SAR	Updated information, referring to other parts of the SAR
21	Decommissioning and end of life aspects	General requirements for decommissioning and end of life aspects	Preliminary information for decommissioning and end of life aspects	Updated information for decommissioning and end of life aspects

Note: SAR — safety analysis report; SSCs — structures, systems and components.

Appendix II

STANDARIZED FORMAT TO DESCRIBE THE DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS AND PLANT EQUIPMENT

II.1. A proposed common format for each section dealing with SSCs (in particular systems) and plant equipment is given below. When a topic is not relevant to a structure, system or component, it is suggested that the subsection be retained and a note be inserted, indicating ‘No description is necessary’.

Functions of each structure, system and component, and item of equipment

II.2. The safety and non-safety functions of the structure, system or component, or equipment, should be described here.

Design basis

II.3. This section should include the safety design criteria, rules and regulations applying to the structure, system or component, such as the following:

(a) List of plant operational conditions and postulated initiating events when the structure, system or component is in operation or will be called on.
(b) Conditions to be practically eliminated, if relevant.
(c) Safety requirements relating to operating conditions, including stresses and environmental conditions (e.g. temperature, humidity, pressure, vibration, irradiation).
(d) Safety classification.
(e) Protection against external hazards.
(f) Protection against internal hazards.
(g) Seismic categorization.
(h) Single failure criterion and protection against common cause failures.
(i) Isolation considerations.
(j) Equipment qualification.
(k) Design standards and requirements.
(l) Fabrication, construction and operational codes and other more specific design aspects, such as the following:
  (i) Overpressure protection;
  (ii) Thermal shock;
  (iii) Leakage detection or collection.

Description of the structure, system or component

II.4. In this section, the structure, system or component should be described. The description should include a list and numbering of individual components, as appropriate; basic drawings of each of the components; and the layout. The main design parameters should be provided, such as the number of components, the dimensions, the operational capacity, the location, the operational parameters and the power supply. The nature and the importance of topics can be different for structures, for mechanical and electrical systems or components, and for instrumentation and control systems.

II.5. A summary of the relevant documentation and records from the manufacturing of the main components should be provided, indicating the supporting information that is available. Relevant information on software based equipment and systems should also be included.

Materials

II.6. In this section, adequate and sufficient information should be provided regarding the materials used in components, the behaviour of these materials under irradiation (when applicable), and the material interactions with fluids that could impair the operation of engineered safety feature systems. The purpose of the information included in this section of the safety analysis report is to demonstrate the compatibility of the materials with the specific fluids to which the materials are subjected. Their specific properties, quality and chemistry requirements should be described.

Interfaces with other equipment or systems

II.7. The support systems (e.g. those providing electrical power, lubrication, ventilation and cooling water), supported systems and other connected systems should be described, as should the corresponding design requirements. Flow diagrams of pipelines, block diagrams of instrumentation and control, single line diagrams, and the locations of units and mechanisms (including valves, pipelines, vessels, instrumentation and control and actuators) should all be presented. The enclosing structures and system layout should also be presented. The boundaries with other systems should be shown.

II.8. The ease of construction or readiness for installation of the structure, system or component, or equipment, at the plant should be described to demonstrate that it can work as designed after installation. Any interference of the structure, system or component, or equipment, with other surrounding structures, systems or components, or equipment, should also be described in the safety analysis report to demonstrate that each SSC and item of equipment can be adequately maintained.

System, component or equipment operation

II.9. This section should summarize the operation of the system, component or equipment.

Instrumentation and control

II.10. This section should describe the method of control and the alarms, indications and interlocks associated with the operation of the structure, system or component.

Monitoring, inspection, testing and maintenance

II.11. This section should present the monitoring, inspection, testing and maintenance (including ageing management) that will help demonstrate the following:

(a) The status of the equipment or system is in accordance with the design intent.
(b) There is adequate assurance that the equipment or system is available and can be relied on to operate as necessary.
(c) There has been no significant deterioration in the availability, performance or integrity of the equipment or system since the last test.

Radiation protection aspects

II.12. This section should describe the measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are as low as reasonably achievable in operational states and in accident or post-accident conditions.

Performance and safety assessment

II.13. This section should present the measures taken to address each of the safety design aspects or requirements listed in para. II.3. This may include a description of the method and results of the analysis demonstrating the required capability of the equipment.

II.14. This section should also describe the assessment of compliance with the applied regulations, codes and standards.

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Annex

TYPICAL TABLE OF CONTENTS OF A SAFETY ANALYSIS REPORT

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8.6.9. Radiological aspects
8.6.10. Performance and safety evaluation

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14.2. Specific information to be included in safety analysis report prior to commissioning

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