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IAEA SAFETY STANDARDS

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

[Revision of GS-G-4.1]

DS449

DRAFT Revised SAFETY GUIDE

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. In order for an operating organization to obtain regulatory approval to build and operate a nuclear power plant, a licence (authorization) is required to be requested from and granted by the regulatory body. In accordance with Requirement 24 from GSR Part 1 (Rev. 1), paras 4.33 and 4.34 [1], the regulatory body shall issue guidance on the format and content of documents to be submitted by the applicant in support of applications for authorization, and the applicant shall be 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 in the form of a report, hereinafter referred to as a safety analysis report. Further requirements on documentation of the safety assessment in the form of a safety report, its objectives, the scope, level of detail and updating are given in Requirement 20 from GSR Part 4 (Rev. 1), paras 4.65 through 4.68 [2].

1.3. This Safety Guide supersedes the guidance provided in the previous version[1]. The update reflects experience from the safety analysis reports for newly built nuclear power plants and good practices used by major nuclear power plant suppliers in developing their safety analysis report for different States; more importantly it reflects the progress in safety assessment approaches since the time of publication of the previous version. In particular, NS-R-1 and NS-R-2 were superseded respectively by SSR-2/1 (Rev. 1) “Safety of Nuclear Power Plants: Design (2016)” [3] and SSR-2/2 (Rev. 1) “Safety of Nuclear Power Plants: Commissioning and Operation (2016)” [4], similarly as NS-R-3 (Rev. 1) “Site Evaluation for Nuclear Installations (2016)” [11], and all associated Safety Guides were subject to revision. The revised Safety Standards aim at significant enhancements of nuclear power plant’s safety, which should be adequately demonstrated in the safety analysis report.

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

1.5. A key target of this Safety Guide is to keep consistency between the content of the safety analysis report and the safety requirements established in the present IAEA Safety Standards. In addition, applicable national or international guidance documents [5-9] were taken into account.

OBJECTIVE

1.6. The objective of this Safety Guide is to provide guidance on the possible content and structure of a safety analysis report in support of a request to the regulatory body for authorization of siting, construction, commissioning, operation and decommissioning of a nuclear power plant. To this end, this Safety Guide is intended to facilitate development of the safety analysis report by the operating organization and checking completeness and adequacy of the safety analysis report by the regulatory body. The content of the safety analysis report recommended in this Safety Guide ensures comprehensiveness of the information about the nuclear power plant safety as required by the current relevant IAEA Safety Standards.

SCOPE

1.7. This Safety Guide is intended mainly for the use in authorization of nuclear power plants, but it may, in parts, have a wider applicability to other nuclear installations or facilities. In accordance with

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

current practices, it is understood that multiunit nuclear power plants having the same unit design will have a common safety analysis report. This Safety Guide was written to apply directly for water cooled reactors and in particular for light water reactors, although many sections and subsections may be applicable for other reactor types as well. The particular contents of the safety analysis report for these reactor types will depend on the specific design of the nuclear power plant proposed, which will determine how sections and subsections described in this Safety Guide are included in the safety analysis report.

1.8. This Safety Guide assumes that it is advantageous to approach the development of the safety analysis report, for the various subsequent stages of the nuclear power plant licensing, as a continuously updated document that reflects the nuclear power plant configuration at a given implementation stage. In consequence, it is expected to maintain the same structure of the safety analysis report throughout its development process from siting up to decommissioning of a nuclear power plant, as much as practicable.

1.9. Although intended mainly for use with new nuclear power plants, the guidance presented in this Safety Guide may also be used, as far as reasonably practicable, for existing nuclear power plants when operating organizations review their existing safety analysis reports to identify any areas in which improvements of the safety analysis report may be appropriate. It is understood that 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

1.10. The current Safety Guide has two main parts, one general and another specific, the latter directly devoted to the structure and content of individual chapters of the safety analysis report.

1.11. The general part is treated in Section 2 and covers the following subsections:

- Roles of the safety analysis report and safety rules of the different origins;
- Structure of the safety analysis report for various stages of the nuclear power plant life time;
- Structure of the safety analysis report;
- Unified description of the design of plant systems;
- Use, review and updating of the safety analysis report during plant operation;
- Formal aspects of the safety analysis report;
- Relation of the safety analysis report to other licensing documents;
- Treatment of sensitive or confidential information;
- Structure of the safety analysis report for different nuclear installations.

1.12. The specific part of this Safety Guide, treated in Section 3, covers the structure and contents of each of the chapters of the safety analysis report and is further supported by two appendices. Appendix I indicates the most relevant information provided in each chapter of the safety analysis report in course of the licensing process. Appendix II presents the unified content and structure of information to be provided for the different systems and components treated in the chapters of the safety analysis report.

1.13. An example of the detailed list of content of the safety analysis report is provided in an Annex.

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

2. GENERAL CONSIDERATIONS

ROLE OF THE SAFETY ANALYSIS REPORT AND SAFETY RULES OF DIFFERENT ORIGINS

2.1. The safety analysis report is a basic licensing document, compiled by the operating organization that the regulatory body uses in assessing the adequacy of the plant safety in all stages of the nuclear power plant life time and the suitability of the licensing basis. The safety analysis report either compiled as a single document or as an integrated set of documents constituting the licensing basis of the plant, should provide adequate justification to demonstrate that a nuclear power plant meets all appropriate safety requirements. At later stages of the plant implementation, it should also provide adequate justification to demonstrate that the plant has been built and commissioned as intended, that all changes in design, construction and commissioning have been properly addressed and that the interactions between the safety aspects of technical, human and organizational factors has been duly considered throughout the report. In addition to providing a documented justification that the plant has been designed to appropriate safety standards, the safety analysis report should be also able to demonstrate that the plant will be operated safely and to provide reference material for the safe operation.

2.2. A nuclear power plant is a strictly regulated nuclear installation, subject to a number of safety rules of different origin, including international conventions, national laws and regulations, international or regional safety and security standards, country of origin’s regulations, quality standards, technical norms and other applicable rules. There may be differences between the various rules. Among these areas, there are standards on the classification of structures, systems and components (SSCs), fire protection, radiation protection, safety of labour and civil construction. The safety analysis report should take into account the whole set of safety rules, including principles for their hierarchical application with specified process to resolve potential differences that may arise between alternative rules. If a hierarchic set of safety rules has not been previously established, such a set should be established for the purpose of the safety analysis report development and afterwards strictly followed throughout the entire life of the safety analysis report.

STRUCTURE OF THE SARSAFETY ANALYSIS REPORT FOR VARIOUS STAGES OF THE NUCLEAR POWER PLANT LIFE TIME

2.3. Common practice indicates that several issues of the safety analysis report are developed for different nuclear power plant licensing stages. Although approaches and titles of the safety analysis report for different licensing stages vary among the States, it is typically developed at least for the three following stages:

- Initial Safety Analysis Report (ISAR), which includes the basis for the authorization of the site;
- Preliminary Safety Analysis Report (PSAR), which includes the basis for the authorization of the construction;
- Pre-operational Safety Analysis Report (POSAR), which includes the basis for the authorization of the nuclear power plant commissioning and operation. During the nuclear power plant operation, the POSAR can be further complemented by additional information, leading to issuance of the Operational Safety Analysis Report (OSAR) or Final Safety Analysis Report (FSAR).

2.4. The structure of the safety analysis report proposed in this Safety Guide is best suited to the PSAR, POSAR and FSAR. Nevertheless, it is recommended to maintain, as far as practicable, the same structure of the safety analysis report throughout its development from the ISAR up to the POSAR. It should be expected that more information will be generated through the operating experience as the nuclear power plant project is near completion. As a guiding principle, any new issue of the safety analysis report should provide updated and revised information on the topics outlined in the previous issue of the safety analysis report, and should explain and justify any significant difference from previous safety considerations. The level of information expected in the individual chapters of different stages of the safety analysis report is indicated in Appendix I.

2.5. At the initial safety analysis report stage, the information about the nuclear power plant may be limited, while information about the site should be reasonably complete. Although the future reactor design could not have been selected yet, the impact of the future nuclear power plant on both the site and its environment need to be based on a reasonable estimate, using e.g. a bounding (enveloping) approach[2]. Rather than describing 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, also 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 often not elaborated in much detail, it may be practicable to combine several subsections of a given chapter of the safety analysis report into one overwhelming section.

2.6. The preliminary safety analysis report should contain sufficiently detailed information, specifications and supporting calculations needed for assessing and demonstrating that the plant can be constructed and operated 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 fulfilled. The safety features incorporated into the design should be described, with due regard to any site specific aspects. The amount of information to be provided in the preliminary safety analysis report should 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.

2.7. The pre-operational safety analysis report should contain revisions and provide more specific information on the topics outlined in the preliminary safety analysis report, taking into account all modifications implemented during the design and construction stages of the nuclear power plant, with 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 essentially justify 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 than in the preliminary safety analysis report issues related 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.

STRUCTURE OF THE SAFETY ANALYSIS REPORT

2.8. The safety analysis report should be structured into the following 21 chapters:

Chapter 1. Introduction and general considerations;

Chapter 2. Site characteristics;

Chapter 3. Safety objectives and design rules of 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. Electric power;

Chapter 9. Auxiliary systems and civil structures;

Chapter 10. Steam and power conversion systems;

Chapter 11. Radioactive waste management;

[2] The bounding approach includes identification of important physical and chemical parameters that may affect the environment for the considered NPP and use of the parameters with the highest impact value.

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;

Chapter 17. Management systems;

Chapter 18. Human factors engineering;

Chapter 19. Emergency preparedness;

Chapter 20. Environmental aspects;

Chapter 21. Decommissioning and end of life aspects.

2.9. The Annex of this Safety Guide provides an example of a detailed structure for individual chapters of the safety analysis report. The main objective of this Annex is to indicate the expected comprehensiveness of information provided in the safety analysis report.

2.10. The proposed safety analysis report structure incorporates into the safety analysis report several new chapters, which were traditionally either missing in the safety analysis report or covered by separate documents. Examples of such chapters are “management systems”, “probabilistic safety assessment”, “emergency preparedness”, “environmental aspects” and “decommissioning and end of life aspects”. While in general it is acceptable to complement the safety analysis report by separate documents, in view of sufficient comprehensiveness of the safety analysis report, use of confidential information and consistency with other licensing documents, it is recommended at least for new nuclear power plants to provide a summary of such documents in the safety analysis report or to make references to them (see para 3.13.28). This need may differ for different stages of the safety analysis report. For example, including environmental aspects is relevant for the initial safety analysis report using information usually available from the Environmental Impact Assessment report, while in subsequent safety analysis reports the radiological impact on people and environment should be comprehensively covered by safety analysis included in Chapter 15 of the safety analysis report.

UNIFIED DESCRIPTION OF THE DESIGN OF PLANT SYSTEMS

2.11. The information to be included in the safety analysis report on various plant systems will depend on the particular type and design of the reactor selected for construction. For some types of reactors, many of the sections discussed below will be entirely relevant, while for other reactor types those sections may not apply directly. However, as a general rule, all systems that have the potential to affect safety should be described in the safety analysis report.

2.12. Description of all the SSCs important to safety should be provided with a demonstration of their conformance to the relevant design requirements. The level of detail of each description should be commensurate with the importance of the item described for safety. In order to ensure consistency and comprehensiveness in the description of all the systems or equipment important to safety, a common structure with more detailed specification of the intended content is provided in Appendix II.

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

2.13. The use of the safety analysis report should not be limited to the licensing and to provide public assurance regarding the safety of the plant prior the operation. The safety analysis report should be continuously used by the licensee to manage safety. It is essential that the operating organization implements the safety intent embodied in the safety analysis report by developing appropriate safety management, 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.

2.14. Since the safety analysis report is an essential part of the overall justification of the safety of the nuclear power plant, it should continuously 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 in adequate time intervals and should be kept up to date accordingly. The updating of the safety analysis report should reflect as appropriate safety related activities performed during the nuclear power plant life time, including, but not limited to, the following:

- Hardware modifications;
- Findings from inspections;
- Procedural changes;
- Maintenance findings;
- Periodic safety reviews;
- Analysis of operational events;
- Analysis of applicable experience from other nuclear power plants;
- Ageing of the SSCs;
- Changes to analytical techniques, standards and criteria;
- Requirements by the regulatory body.

2.15. Ideally, the safety analysis report should correspond to the current plant status at all times. Since such ideal situation is difficult to achieve, it is considered a good practice to update the safety analysis report once a year, e.g. by replacing affected parts of the safety analysis report by the corresponding new versions. As a minimum, updating of the safety analysis report should be a part of the periodic safety review usually scheduled every ten years (see SSG-25 [10]. However, it is essential that all the activities that could impact the validity of the safety analysis report are clearly identified and controlled by procedures that include a requirement to timely review the impact of each event. Between the updates of the safety analysis report, the full impact of any modification on the safety of the nuclear power plant should be evaluated and submitted to the regulatory body for approval before being implemented.

2.16. Changes incorporated into the safety analysis report should be performed in accordance with the procedures established by the operating organization and all easily traceable (e.g. revision number and date of release indicated in all the new pages incorporated); this include those incorporated during the review process of the safety analysis report by the regulatory body.

FORMAL ASPECTS OF THE SAFETY ANALYSIS REPORT

2.17. 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 for the review of the regulatory body. Depth of description in the safety analysis report is determined by the requirement that the safety analysis report is a basic reference material, thus should be sufficiently detailed to be understandable by itself.

2.18. In view of the prime responsibility of the operating organization for safety, when the safety analysis report is developed by a third party, e.g. by the nuclear power plant vendor, it should contain either itself or in complementary documents sufficient and sufficiently detailed information to allow for an independent verification performed either directly by the operating organization or by any other qualified organization on its behalf (see GSR Part 4 (Rev. 1), para 4.64 [2]).

2.19. 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 evaluate the safety level independently. Tables, drawings, plots and figures should be used wherever they contribute to the clarity and brevity of the report.

2.20. The information contained in the safety analysis report should be self-sufficient to a reasonable extent. The most important supporting materials should be supplemented to the safety analysis report. These materials serve to enhance the review process and the later usability of the safety analysis report. Some less essential external references are usually not submitted to the regulatory body together with the safety analysis report, but they should be made available upon request.

2.21. User friendly format of the safety analysis report significantly facilitates its use and review. Therefore the safety analysis report made available should include an electronic form. Additionally, use of internal reference links between safety analysis report chapters and sections in electronic form is useful. Use of external references and their extended use are inevitable (e.g. detailed design documents, references to standards, detailed analysis reports, code validation reports and source material for probabilistic safety assessment). References to lower level documents are also useful (e. g. operational procedures, emergency operating procedures (EOPs) and severe accident management guidelines (SAMG)).

RELATION OF THE SAFETY ANALYSIS REPORT TO OTHER LICENSING DOCUMENTS

2.22. In addition to the safety analysis report, there are other documents used in the licensing process. Typical examples are the reports on Environmental Impact Assessment, probabilistic safety assessment studies and emergency preparedness or decommissioning plans. Some of the information contained in the safety analysis report may be the same as required for other licensing documents. In such cases, the required information needs to be in parallel incorporated in several relevant documents to the appropriate extent. The reason is that these documents may be responsive to different legislative requirements and each of them should be essentially self-contained.

2.23. Consistency and continuity of information provided in different licensing documents as well as in subsequent stages of the safety analysis report should be ensured in accordance with GSR Part 1 (Rev. 1), para 4.28 [1]. In case a subsequent stage of the safety analysis report provides more pessimistic results than the previous stage, the changes incorporated should be justified. Any significant differences between information provided in these documents should be explained and justified.

TREATMENT OF SENSITIVE OR CONFIDENTIAL INFORMATION

2.24 It is understood that certain parts of the safety relevant information maybe of sensitive or confidential nature. It is up to the operating organization to limit the content of such information presented in the safety analysis report or to adopt other adequate countermeasures. The latter may include limitations of access to certain parts of the safety analysis report, to ensure that the information publicly available will not disclose data which could be misused for malicious acts endangering nuclear power plant safety or lead to violation of intellectual property rights.

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

2.25. This Safety Guide is intended to be used for nuclear power plants. Nevertheless, some of its parts may be applied to other nuclear installations, such as nuclear fuel cycle facilities. In that case, it can be taken into account that common or similar SSCs are used in different facilities, as well as considered operating conditions. In a majority of cases, the nature and the magnitude of the associated risk is not comparable with that of a nuclear power plant; therefore, the particular structure and content of the safety analysis report typically depend on the specific type and design of the nuclear installation proposed, determining how different sections from this Safety Guide can be covered in the safety analysis report. Correspondingly, the scope and content of the safety analysis report for some nuclear installations may be significantly simplified as compared to the safety analysis report for the nuclear power plant.

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, which includes:

(a) A statement of the main purpose of the safety analysis report;

(b) The main information about the process of preparation of the safety analysis report;

(c) A description of the structure of the safety analysis report, the objectives and scope of each of its chapters and the connections between them.

Project implementation

3.1.2. Information provided in this section should include a description of the existing authorization status, with indication of future project milestones, as appropriate.

Identification of interested parties

3.1.3. The primary contractors for the design, construction, and operation of the nuclear power plant should be specified in this section, as appropriate. The principal consultants and outside service organizations (such as those providing audits of the management system) should be also identified. The division of responsibilities between the designer(s), architect-engineer(s), constructor(s), and operating organization should also be delineated.

Information on the layout and other aspects

3.1.4. General layout drawings for the entire plant (including multiunit plants) should be included in this section, together with 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 with equipment and systems external to the plant should be described.

3.1.6. This section may also refer to confidential information on the provisions made for the physical protection of the plant. It may also include appropriate coverage of the steps taken to provide protection in the event of a malicious act on or off the site.

General plant description

3.1.7. This section should provide a general description of the plant, including overall safety philosophy, current safety concepts 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 the subsequent chapters.

3.1.8. The section should briefly present (e.g. in a table) the principal elements of the plant, including the number of units, where appropriate, the type of the 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 systems, the thermal power levels in the core, the corresponding net electrical power output for each thermal power level, and any other characteristics necessary for understanding the main technological processes included in the design.

3.1.9. If applicable, it is recommended to compare the plant design with designs licensed earlier, so as to identify the main differences and assist in the justification of any modifications and improvements made. This comparison may focus on new safety features for nuclear power plants that differ from previous reactor designs such as use redundant, diverse, simplified, inherent, passive, or other innovative means to accomplish 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 complemented with 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, including startup, power operation, shutting down, 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 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. Principles of safety management should be described.

Additional supporting/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 relevant regulations, codes and standards representing the safety rules that have been used in the design, including information on the use of the 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.

CHAPTER 2. SITE CHARACTERISTICS

3.2.1. Chapter 2 should provide information on the geological, seismological (including fault displacement), volcanic, hydrological (including flooding), meteorological and geotechnical characteristics of the site and the surrounding region and characteristics of external human induced events, in conjunction with the information on the radiological dispersion characteristics of the site and surrounding environment, the present and projected population distribution and land use that is relevant to the safe design and operation of the plant.

3.2.2. Sufficient data should be included to permit an independent evaluation. Information provided in chapter 2 should be periodically updated, taking into account the latest information and knowledge as a basis for evaluation of safety implications of the changes.

3.2.3. Site characteristics that may affect the safety of the plant should be investigated and the relevant results of the corresponding assessment should be included in this chapter (see NS-R-3 (Rev. 1) [11], NS-G-3.1 [12], NS-G-3.2 [13], NS-G-3.6 [14], SSG-9 [15], SSG-18 [16], SSG-21 [17] and SSG-35 [13].

3.2.4. This chapter of the safety analysis report should provide information concerning the site evaluation as support for the design phase, design assessment phase and periodic safety review. This information should include:

(a) Site specific hazard evaluation for external events of natural origin (such as earthquakes and surface faulting, meteorological events, flooding, geotechnical and volcanic hazards, and hazards from biological organisms) and human induced origin (such as aircraft crashes and chemical explosions);

(b) Design targets in terms of recurrence probability of external events, taking into account their severity and associated uncertainties;

(c) Definition of the design basis of an SSC for external events, depending on the safety importance of each SSC, including consideration of adequate margins;

(d) Collection of site reference data for the plant design (geological, seismological, geotechnical, volcanic, hydrological and meteorological);

(e) Evaluation of the impact of the site related issues to be considered in the parts of the safety analysis report on emergency preparedness and accident management;

(f) Arrangements for the monitoring of site related parameters throughout the lifetime of the plant;

(g) Potential for specific hazards to give rise to impacts simultaneously on several units in case of a multiple unit site.

3.2.5. A discussion of considerations concerning the site exclusion and/or acceptance criteria applied for the purposes of preliminary screening of the site for suitability after the site survey stage should be provided in this section 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 account for such uncertainty levels should be considered in 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 licensee and the surrounding area in which there is a need for consultation with other interested parties on the control of activities with the potential to affect plant operation, including flight exclusion zones.

3.2.8. Information on such activities should include relevant data on the population distribution, including transient populations, and density and on the disposition of public and private facilities (airports, harbours, rail transport centres, factories and other industrial sites, schools, hospitals, police services, firefighting services and 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.

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 to be taken into account in design of SSCs, with due consideration of envisaged evolution of these hazards during expected nuclear power plant lifetime; see NS-R-3 (Rev. 1) [11] provides an overview of hazards to be considered.

3.2.11. 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.

3.2.12. 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 discussed in this section.

3.2.13. The definition of the target probability levels for design against external hazards and their consistency with the acceptable limits should be presented. Attention should be paid to the external hazards which could potentially lead to common cause failures of the safety systems and safety features for design extension conditions.

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

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

3.2.16. Output from the evaluation of potential combinations of such site specific hazards that could affect the safety of the installation should be also part of the information included in this section.

Proximity of industrial, transportation, and military facilities

3.2.17. This section should present identification of locations and routes representing potential risks for the plant and the results of a detailed evaluation of the effects of potential accidents at industrial, transport or other installations in the vicinity of the site. Projected developments over the envisaged nuclear power plant life time relating to this information should also be presented and updated in future stages of the safety analysis report as required.

3.2.18. Any identified threats to the plant should be considered for inclusion in the design basis events to help determine any additional measures considered necessary to mitigate the adverse effects of the potential incidents identified.

Activities at the plant site that may influence the plant's safety

3.2.19. Any processes or activities at the site that, if incorrectly carried out, could influence the safe operation of the plant should be presented and described; examples of such processes or activities are vehicular traffic in the plant area, the storage and potential spillage of fuels, gases and other chemicals, intakes (e.g. of air for control room ventilation) or contamination by harmful particles, smoke or gases.

3.2.20. Measures for site protection (e.g. dams, dykes for flood control and drainage) and any modifications to the site (such as soil substitution or 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

3.2.21. This section should present sufficient information for evaluation of the potential implications of the hydrological conditions at the site for the plant design and safe operation with special attention devoted to the conditions potentially affecting 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 possibilities of using ground water sources in extraordinary situations should be considered.

3.2.22. The conditions to be taken into account should include potential floods resulting from phenomena such as abnormally ice effects, heavy rainfall and runoff floods from watercourses, reservoirs, adjacent drainage areas and site drainage. This section should also include a consideration of flood waves resulting from dam failures, flooding caused by landslides, ice jams and other ice related flooding as well as 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 should be considered (e.g. tides and strong wind).

3.2.23. The information given in this section should be prepared in a way allowing to be used in the assessment of the transport of radioactive material to and from the site and the dispersion of radionuclides to the environment.

Meteorology

3.2.24. 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.

3.2.25. This section should include information relevant to the assessment of the hazards from meteorological events potentially affecting the plant and for assessment of transport of radioactive material to and from the site and the dispersion of radionuclides to the environment.

3.2.26. The extreme values of meteorological parameters or meteorological events, including temperature, humidity levels, rainfall levels, wind speeds for straight and rotational winds including tornadoes (due to the sudden pressure drop that accompanies the passage of the center of a tornado), waterspouts (due to their potential to transfer large amounts of water to the land from nearby water bodies), dust storms, sandstorms and snow loads; see SSG-18 [16], should be evaluated in relation to the design, taking into account envisaged evolution of such extreme parameters over the nuclear power plant life time. The potential for lightning and windborne debris to affect plant safety should be considered, where appropriate.

Geology, seismology, and geotechnical engineering

3.2.27. This section should provide information concerning the geological, seismic and tectonic characteristics of the site and of the sufficiently large region surrounding the site. The evaluation of seismic hazards should be based on a suitable seismotectonic model substantiated by appropriate evidence and data. The results of this analysis to be used further in other sections of the safety analysis report in which structural design, seismic qualification of components and safety analysis are considered should be described in sufficient detail.

3.2.28. Site reference data relating to geotechnical soil properties should be provided. Geotechnical hazards such as slope instability, collapse, subsidence or uplift of the site surface, soil liquefaction, stability of subsurface materials and behaviour of foundations should be characterized in this section. The process of the collection of data for the design of foundations, the evaluation of the effects of soil–structure interaction, the construction of earth structures and buried structures, and soil improvements at the site should be described.

3.2.29. This section should present the relevant data for the site and the associated ranges of uncertainty to be used in the structural design. Reference should be made to the technical reports describing in detail the conduct of the investigation campaigns, and their extension, and the origin of the data collected on a regional basis and/or on a bibliographic basis.

3.2.30. The design of subsurface material and of buried structures, and site protection measures, if relevant, should also be documented. A description of projected developments relating to the above mentioned information should also be provided and should be updated as required.

Site characteristics and the potential effects of the nuclear power plant

3.2.31. The characteristics of the site and surrounding environment regarding dispersion of radioactive material in water, air and soil should be described in this section; (see section 4 from NS-R-3 (Rev. 1) [11]).

Radiological conditions due to external sources

3.2.32. The radiological conditions in the environment at the site and its surroundings, with account taken of the radiological effects of on-site and collocated installations and other external radiation sources, if any, should be described in sufficient detail to serve as an initial reference point and a basis for assessment of radiological conditions at the site and the environment.

3.2.33. A description should be presented of the radiation monitoring systems available 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

3.2.34. The feasibility of emergency preparedness in terms of access to the plant and of transport in an emergency, including a severe accident, should be discussed in this section of the safety analysis report, taking into account all reactor units or other nuclear installations on the given site. Information provided should include availability of adequate access and egress roads for evacuation of personnel, including access to the site, and nearby population sheltering and supply networks in the vicinity of the site.

3.2.35. The availability of local transport networks and communications networks during and after an accident and for the implementation of a suitable emergency plan should be described. It should be ensured that the requirements for adequate infrastructures external to the site are met.

3.2.36. The needs for any necessary administrative measures, such as agreements with local authorities and support services, should be identified, together with the relevant responsibilities of bodies and response organizations other than the operating organization.

Monitoring of site related parameters

3.2.37. The provisions to monitor site related parameters affected by earthquakes and surface faulting, meteorological events, flooding, geotechnical and hazards from biological organisms or human induced hazards (such as aircraft crashes and chemical explosions) should be described in this section. This may be used to provide necessary information for emergency operator actions in response to external events, to support the periodic safety review at the site, to develop dispersion modelling for radioactive material and as confirmation of the completeness of the set of site specific hazards taken into account.

3.2.38. On-site meteorological monitoring programme should be described which can be potentially used for updating meteorological data in the future, for prediction of dispersion of radioactive substances during plant operation or for early warning against extreme meteorological events. Monitoring of demographic and hydrological conditions over the life time of the plant should be described in this section as well (see NSR-3 (Rev. 1), para 5.1 [11]).

3.2.39. Long term monitoring programmes should include the collection of data recorded using site specific instrumentation and data from specialized institutions for use in comparisons to detect significant changes from the design basis; for example, those due to the possible effects of global warming.

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

CHAPTER 3. SAFETY OBJECTIVES AND DESIGN RULES OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.3.1. Chapter 3 should outline the general design concepts, requirements, codes and standards, applicable for different kinds of SSCs 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 description of different SSCs.

General safety design basis aspects

3.3.2. The overall safety philosophy and general approaches for ensuring safety should be presented in this section. These approaches should be based on the IAEA Safety Requirements established regarding nuclear power plant design, (SSR-2/1 (Rev. 1)) [3] and safety assessment (GSR Part 4 (Rev. 1)) [2]. Several relevant subjects are discussed in the following subsections.

Safety objectives

3.3.3. This subsection should summarize the overall safety philosophy, safety objectives and high level principles used in the project. These should be based on the safety principles set out in the 'Fundamental Safety Principles' (SF-1) [19].

Safety functions

3.3.4. This subsection should identify plant specific safety functions to fulfil the fundamental safety functions by the plant design features, in accordance with the Requirement 4 of SSR-2/1 (Rev. 1) [3] and depending on the nature of the facility or activity. The corresponding relevant SSCs necessary to fulfil these safety functions should be introduced.

3.3.5. If fundamental safety functions are subdivided into more detailed specific safety functions and functional criteria, with the objective to facilitate 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 subsection should describe in general terms the design approach adopted to meet the fundamental safety objective (see SF-1, para 2.1 (a) [19]) and to ensure that, in all plant states, radiation doses within the installation or in the plant surroundings due to any release of radioactive material are kept below authorized limits and as low as reasonably achievable (ALARA), economic and social factors being taken into account.

3.3.7. Relevant radiological acceptance criteria for nuclear power plant staff and for the public assigned for each category of plant states consistently with their concurrency (normal operation, anticipated operational occurrences, design basis accidents and design extension conditions) should be introduced in this subsection.

General design basis and plant states considered in the design

3.3.8. This subsection should describe the plant capabilities 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 (anticipated operational occurrences), design basis accidents , design extension conditions (design extension conditions) without significant fuel degradation, as well as design extension conditions with core melting (severe accidents).

3.3.9. 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 Section 3 (Chapter 15) of this Safety Guide.

Defence in depth

3.3.10. This subsection 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 considered in all stages of the lifetime of the nuclear power plant, for all plant states and for all safety related activities in accordance with SSR-2/1 (Rev.1), §2.12-§2.18 [3].

3.3.11. It should be demonstrated that there are physical barriers to the release of radioactivity and systems to protect integrity of the barriers and measures are taken to ensure robustness of provisions at each level of defence in depth. It should also be demonstrated that measures are taken for adequate independence of levels. Particular emphasis should be placed on robustness and independence of safety systems and safety features provided for design extension conditions with core melting.

3.3.12. Where appropriate, any envisaged operator actions to mitigate the consequences of events and to assist in the performance of important safety functions essential for defence in depth should be described.

Application of general design requirements and technical acceptance criteria

3.3.13. This subsection should include 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 or internationally recognized industrial codes and standards, or in regulatory guidance documents, the use of such design approaches should be elaborated in this subsection of the safety analysis report, with reference made to the specific applicable codes and standards.

3.3.14. The scope of implementation of the single failure criterion and how compliance with this criterion is achieved should be described here, as part of the design basis. If relevant, consideration is given to the possibility of a single failure occurring while a redundant train of a system is out for maintenance and/or is impaired by internal or external hazards.

3.3.15. Provisions to comply with requirements 21 and 23-26 from SSR-2/1 (Rev. 1) [3] for protection against common cause failures should also be addressed here.

3.3.16. Any other relevant approaches aimed at ensuring safety should be specified here, such as: (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) leak before break concepts for safe state design.

3.3.17. Specific technical acceptance criteria associated with integrity of individual barriers against releases of radioactive materials used in the design should be listed here. If probabilistic safety objectives or criteria have been used in the design process, these should be also specified in this subsection.

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

3.3.18. This subsection should summarize the design and operational provisions implemented to demonstrate the 'practical elimination'[3] of the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release (see SSR-2/1 (Rev. 1), para 5.31 [3].

3.3.19. In this subsection, reference should be also made, as appropriate, to other sections of the safety analysis report (see Chapter 15) where relevant confirmatory analysis is presented.

Safety margins and avoidance of cliff edge effects

3.3.20. This subsection should summarize the approach taken to ensure adequate margins to prevent cliff-edge effects related to damage of barriers against releases of radioactive substances to the environment; see SSR-2/1 (Rev. 1) [3].

3.3.21. The subsection should in particular describe the approach and other assumptions for deterministic safety analysis (conservative or realistic) selected for demonstration of adequate safety margins, including use of sensitivity studies to demonstrate the avoidance of cliff edge effects, in the analyses applicable for design extension conditions.

3.3.22. The subsection should also describe the approach used for demonstration of safety margins for internal or external hazards. In case of external hazards, it should be described how adequate safety margins are ensured for beyond design basis external events, see requirement 17 from SSR-2/1 (Rev. 1) [3].

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[3] SSR 2/1 (Rev 1) [3], footnote 4: 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

Design approaches for reactor core and fuel storage

3.3.23. This subsection should describe differences in design approaches adopted to demonstrate performance of the safety functions in the reactor and in the fuel storages, in particular in the spent fuel pool. These differences may imply differences in implementation of defence in depth, different specification of derived safety functions, different monitoring means and substantial differences in time evolution of accidents. According to requirement 4 from SSR-2/1 (Rev. 1) [3], there is a need to consider shielding of the irradiated fuel elements as necessary for meeting the limits for occupational radiation doses. More detailed description of design provisions is to be included in relevant sections of chapters 4 and 9; demonstration of evolution of the accidents and availability of sufficient margins should be included in chapter 15. (See NS-G-1.4) [20].

Considerations of interactions between multiple units

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

3.3.25. Interconnections between the units appropriate for further safety enhancement, if any, should be explicitly described in this subsection explaining the positive, as well as the adverse, effects of such interconnections.

3.3.26. When one or more units are mothballed [conserved] and kept in safe-storage state (e.g. in preparation for future decommissioning), a description should be provided of any severed interconnections or services provided by shared systems. In addition, results of analyses addressing the impact of severing the interconnections and shared services on other operating units should be provided.

Design provisions for ageing management

3.3.27. This subsection should define the design life time of items important to safety and should describe how relevant mechanisms of ageing and wear out were taken into account in nuclear power plant design in order to ensure design life time of most important nuclear power plant components. Special attention should be devoted to the reactor pressure vessel, in particular to its neutron embrittlement.

3.3.28. It should be described how adequate margins are maintained, taking into account ageing relevant degradation mechanisms, including those caused by testing and maintenance, plant states during a postulated initiating event and plant states following a postulated initiating event.

3.3.29. It should be described how consideration of ageing effects caused by environmental factors (such as conditions of vibration, irradiation, humidity or temperature) over the expected service life of the items important to safety have been considered in their qualification programme. Reference should be made to a comprehensive ageing management programme (see chapter 13).

Classification of structures, systems and components

3.3.30. This section should provide information on the approach adopted for the categorization of safety functions, identification of SSCs needed to perform the safety functions and safety classification of the SSCs; see SSR-2/1 (Rev.1) [3] and SSG-30 [21]. In particular, the description should include the details of the following:

- Methodology and criteria applied for safety classification;
- Categorization of the safety functions;
- Safety classification of the SSCs;
- Associated engineering, design (e.g. environmental qualification, seismic categorization) and manufacturing rules for different safety classes of SSCs;
- Verification of the classification.

3.3.31. 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 those with a higher classification.

3.3.32. A list of main structures, systems and components important to safety with their related safety functions, safety classification, seismic categorization and any other associated safety requirements, should be included here, either in an annex to or as a reference in the safety analysis report.

Protection against external hazards

3.3.33. This section should provide a list of external hazards considered in the design, quantitative design parameters of individual hazards, relevant design criteria, codes and standards, methods of assessment and a description of 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.

3.3.34. Both hazards of natural origin as well as human induced hazards found relevant for the given site should be taken into account. Hazards with potential impact on several nuclear power plant units simultaneously should be specifically considered (see para 5.15B from SSR-2/1 (Rev. 1) [3] and NS-G-1.5 [22]. An indicative list of external hazards to be considered should be identified in Chapter 2.

3.3.35. Consideration should be given to causation and likelihood in postulating combinations of potential hazards, (para 5.17 of SSR-2/1 (Rev. 1) [3], such as induced effects caused by primary external hazards, for example flooding following an earthquake. More generally, combinations of various kinds of loads, including loads from randomly occurring individual events, should be considered and described here, unless their probability is very low.

3.3.36. The general information concerning different hazards taken into consideration in the design should be presented in this chapter. The detailed design information, including calculation and test results should be presented in Chapters 4-12.

Seismic design

3.3.37. The seismic design characteristics and codes and standards applicable for the design, methodologies, basic assumptions, specific requirements to be taken into account should be presented in this section; see SSR-2/1 (Rev.1) [3]. The design solutions for ensuring the required safety/performance and compliance with the requirements should be presented in Chapters 4-12. Information provided should include:

- Seismic design parameters;
- Design ground motion;
- Applicable seismic system analysis;
- Seismic analysis methods;
- Procedures used for analytical modelling;
- Interaction of structures with different safety classification;
- Seismic instrumentation;
- Control room operator notification.

Extreme Weather Conditions

3.3.38. This subsection should present the design basis weather conditions for the extreme meteorological hazards (as identified in Chapter 2 of the safety analysis report), codes and standards applicable for the design, methodologies with basic assumptions, specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety objectives and compliance with the requirements are presented in Chapters 4-12.

3.3.39. Possible off-site protective actions and the required human interactions to mitigate extreme weather conditions should be specified in Chapter 13 and described in details with the justification of the successful protection against the design basis hazard for each case.

External Flooding

3.3.40. This subsection should present the design basis external flooding conditions and hazards as identified in Chapter 2 of the safety analysis report, codes and standards applicable for the design, methodologies basic assumptions, specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety objectives and compliance with the requirements are presented in Chapters 4-12.

3.3.41. This subsection should 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

3.3.42. This subsection should specify and describe all structures, systems (or parts of systems) and components that are to be protected against damage from aircraft crash. These are the SSCs necessary to perform functions required to attain and maintain a safe shutdown condition or to mitigate the consequences of an accident. It should define the design basis aircraft crash characteristics as defined in Chapter 2 of the safety analysis report and applicable design codes and standards, assumptions and specific requirements. The design measures for ensuring the required safety/performance and demonstration of compliance with the requirements should be presented in Chapters 4-12.

Missiles

3.3.43. The level of protection against all external missiles (other than aircraft) identified in Chapter 2 of the safety analysis report should be included. This part 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, methodologies with basic assumptions and specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety/performance and compliance with the requirements should be presented in Chapters 4-12.

External fires, explosion and toxic gases

3.3.44. This subsection should discuss the protection against external fires, explosions and toxic gases originated from other industrial and transportation activities. The design basis external fire, explosion and toxic gases hazards as identified in Chapter 2 of the safety analysis report should be discussed including the codes and standards applicable for the design, methodologies basic assumptions, specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety/performance and compliance with the requirements should be presented in Chapters 4-12.

Other External Hazards

3.3.45. This subsection should discuss the protection against any other external hazards considered in the design, covering each in a separate subsection. The design basis hazards should be discussed including the codes and standards applicable for the design, methodologies with basic assumptions, specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety/performance and compliance with the requirements should be presented in Chapters 4-12.

Protection against internal hazards

3.3.46. This section should provide a list of internal hazards considered in the design, quantitative design parameters of individual hazards, relevant design criteria, codes and standards, methods of assessment and a description of 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 in order to ensure safe shutdown of the plant; see SSR-2/1 (Rev. 1) [3], NS-G-1.7 [23] and NS-G-1.11 [24]. The list of internal hazards should include the following:

- Internal fires and explosions;
- Heavy load drops;
- Internal flooding;
- Pipe whipping following their ruptures and dynamic effects associated with high energy pipe ruptures;
- Internal missiles such as those originated from rotating structures;
- Dynamic effects associated with high energy pipe rupture;
- Failures of pressurized components, supports or any other structures.

3.3.47. Similarly as in the case of external hazards, consideration should be given to non-negligible combination of internal hazards (such as flooding due to an internal missile) or plausible combination of external and internal hazards.

Internal Fire, Explosion and Toxic Gases

3.3.48. This subsection should summarize the protection against internal fires, explosions and toxic gases originated from the on-site activities and technological failures. The design parameters, the loads and exposures, protection measures and the required human interactions should be specified and described with the justification of the successful protection. The description and justification of the relevant countermeasures should be included in part 9A of the safety analysis report. Confirmation of adequacy of the design measures for ensuring the required safety level and compliance with the requirements should be presented in Chapters 4-12.

Internal Flooding

3.3.49. This subsection should summarize the protection against internal floods. The design requirements, the resulting loads and their implications, off-site protective actions and the required human interactions should be specified and described with the justification of the successful protection. This includes the identification of all of the potential flooding mechanisms of water or steam floods and the protection and drainage measures in relation with the particular SSC. In addition, the analysis of the damage of the SSC should be covered by this subsection. The design measures for ensuring the required safety level and compliance with the requirements should be presented in Chapters 4-12.

Internal Missiles

3.3.50. This subsection should describe the protection against internal missiles. The design requirements, the loads and their implications, off-site protective actions and the required human interactions should be specified and described with the justification of the successful protection. This includes the identification of all potential missile generating events, the parameters of generated missiles, including turbine missiles and any other missiles either inside or outside the containment. The design measures for ensuring the required safety level and compliance with the requirements should be presented in Chapters 4-12.

High Energy Line Breaks

3.3.51. This subsection should describe the protection against high energy line breaks. The design requirements, the loads and their implications, off-site protective actions and the required human interactions should be specified and described with the justification of the successful protection. This includes the identification of all postulated failures of high energy pipelines and the dynamic effects of the pipe break and the SSCs potentially affected. The design measures for ensuring the required safety level and compliance with the requirements should be presented in Chapters 4-12.

Other Internal Hazards

3.3.52. This subsection should describe the protection against any other internal hazards considered in the design, each covered in a separate subsection. The design basis hazards should be discussed including the codes and standards applicable for the design, methodologies with basic assumptions, specific requirements regarding loads and load combinations to be taken into account. The design measures for ensuring the required safety level and compliance with the requirements should be presented in the specific Chapters 4-12.

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

3.3.53. This section should present relevant information on the design approaches to civil engineering of buildings and structures, including their foundations. It should also briefly introduce the way in which the margins have been taken for the construction of buildings and structures that are relevant to nuclear 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 provided in Chapter 9B.

3.3.54. General information on civil engineering works and structures should be composed of 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.

3.3.55. 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 subsection should focus on the specific information related to foundations.

3.3.56. This section should also specify the safety requirements for the containment building itself, including its leak tightness, mechanical strength, pressure resistance and resistance to hazards. Specific information should be provided for concrete containments and for steel and concrete internal structures of the containment. The major structures to be addressed should include:

- Reactor support system;
- Steam generator support system;
- reactor coolant pump support system;
- Primary shield wall and reactor cavity secondary shield walls;
- Other major internal structures, such as supports, the refuelling cavity walls, in-containment refuelling water storage tank, the operating floor, intermediate floors, and various platforms.

The detailed descriptions of the structures containing general arrangement layouts, sections and principal features of major internal structures should be presented in Chapters 9B.

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

- Applicable Codes, Standards, and Specifications, Loads and Load Combinations;
- Structural Acceptance Criteria;
- Testing and In-service Inspection Requirements.

3.3.58. Other buildings, for which the design rules should be described, include:

- Auxiliary building;
- Safety building;
- Fuel storage building;
- Control building;
- Diesel generator building.

General design aspects for mechanical systems and components

3.3.59. Relevant information on design principles and criteria, and the codes and standards used in the design of mechanical components, should be included in this section. Information should be provided concerning the design loads and load combinations with appropriate specified design and service limits for components and supports.

3.3.60. Methods, assumptions, computer programmes or experimental verification used in dynamic and static analyses to determine the structural and functional integrity of the mechanical components should be presented. Information concerning the design transients and resulting loads and load combinations with appropriate specified design and service limits for classified components and supports should be presented.

3.3.61. A complete list of transients used in the design and fatigue and fracture analysis of all reactor coolant system and core support components, component supports, and reactor internals should be presented. The list should include the number of events for each transient, as well as the number of load and stress cycles per event and for events in combination, the number of transients assumed for the design lifetime of the plant and describe the environmental conditions to which equipment important to safety will be exposed over the design lifetime of the plant (e.g., coolant water chemistry).

3.3.62. Requirements for ensuring structural integrity of pressure-retaining components, component supports, and core support structures designed and constructed in accordance with the rules should be described. This discussion should also incorporate design information related to component design and include current design information, representative, or bounding information.

3.3.63. This section should describe the approach and rules for the design and analyses of the piping system, including piping components and associated supports. The discussion should cover requirements and procedures used in preparing the design specification of the piping system, including loading combinations, design data, and other design inputs. The specific information on piping design of particular systems is given in Chapters 5, 6 and 9A.

General design aspects for Instrumentation and control systems and components

3.3.64. Relevant information on 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 on general design principles should be provided regarding: (a) Performance; (b) Reliability; (c) Independence of provisions for the different plant states; (d) Qualification; (e) Single failure criterion application; (f) Access to equipment; (g) Quality; (h) Testing and testability; (i) Maintainability; (j) Identification of items important to safety.

3.3.65. The design basis should identify functions, conditions and requirements for the overall instrumentation and control and each individual instrumentation and control system. This information is then used to categorize the functions and assign them to systems of the appropriate safety class; see SSG-30 [21].

General design aspects for electrical systems and components

3.3.66. Relevant information on 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 on general design principles regarding: (a) Redundancy; (b) Independence; (c) Diversity; (d) Controls and monitoring; (e) Identification; (f) Capacity and capability of systems for different plant states.

Equipment qualification

3.3.67. This section should describe, consistently with SSR-2/1 (Rev. 1) [3], the scope of qualification and qualification procedure 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 when subjected to the range of individual or combined environmental challenges identified for the situations they are supposed to perform, throughout the lifetime of the plant.

3.3.68. It should be presented how the qualification programme takes account of all identified and relevant potentially disruptive influences on the plant under which the SSCs are performing, including internal and external hazard based events. If acceptance criteria are used for the qualification of plant items by testing or analysis, these should be described here.

3.3.69. The section should include information on the methods used to ensure that the SSCs are suitable for their design duty, remain fit for purpose and continue to perform 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).

3.3.70. The criteria should be provided that are used for qualification, including the decision criteria for selecting a particular test or method of analysis, the considerations defining conditions resulting from the applicable plant conditions and post-accident environmental conditions and the seismic and other relevant dynamic load input motion, and the process to demonstrate the adequacy of the qualification program. The criteria should be presented for electromagnetic qualification, including the decision criteria for selecting a particular test or method of analysis, the considerations defining the electromagnetic impact, and the process to demonstrate the adequacy of the electromagnetic qualification program.

3.3.71. A list of important equipment items, together with their qualification, should be established and provided or referenced here.

In-service monitoring, tests, maintenance and inspections

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

Compliance with national and international standards

3.3.73. This section should include a statement of the conformance of the plant design with the design principles and criteria, which themselves will allow compliance with the safety objectives adopted for the plant.

CHAPTER 4. REACTOR

3.4.1. This chapter should provide relevant information on the reactor to demonstrate its capability to perform relevant safety functions throughout design lifetime 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 this chapter should demonstrate compliance with the requirements 43 to 46 from SSR 2/1 (Rev. 1) [3]; recommendations to meet the requirements applicable to this chapter are provided in NS-G-1.12 [25].

Summary description[4]

3.4.2. A summary description should be provided of the mechanical, nuclear, thermal-hydraulic behaviour of the various reactor components, including the fuel, reactor vessel internals, 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.

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[4] Appendix II provides guidance to describe NPP systems design in the Safety Analysis Report

Fuel design[5]

3.4.4. A description should be provided of the main fuel elements with safety substantiation for the selected design bases. The justification for 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 relevant plant states.

Nuclear design

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

(i) The nuclear design bases, including nuclear and reactivity control limits such as limits on excess reactivity, fuel burnup, reactivity coefficients, power distribution control and reactivity insertion rates;

(ii) The nuclear characteristics of the lattice, including core physics parameters, fuel enrichment distributions in ^{235}U and Pu vectors contents (if applicable), burnable poison rods distributions and concentrations, burnup distributions, control rods type and locations, and refuelling schemes;

(iii) 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;

(iv) The design basis of power distributions within fuel elements, fuel assemblies and the core as a whole, providing information on axial and radial power distributions and overall capability for reactivity control;

(v) The neutronic stability of the core, including Xenon stability, throughout an operating cycle, with consideration given to the possible anomalies in the different modes of normal operation covered by the design basis;

(vi) Special core configurations such as mixed core or modes of normal operation.

Thermal-hydraulic design

3.4.6. This section should provide the following information:

(i) The thermal-hydraulic design bases for the reactor core and attendant structures, and the interface requirements for the thermal-hydraulic design of the reactor coolant system;

(ii) The analytical tools and methods and computer codes (including their verification and validation with uncertainties) used to calculate thermal-hydraulic parameters;

(iii) Flow, pressure and temperature distributions, with the specification of limiting values and their comparison with design limits;

(iv) Justification of the thermal-hydraulic stability of the core.

Design of the reactivity control systems

3.4.7. All reactivity control systems should be described. A demonstration should be provided that the reactivity control systems, including any essential ancillary 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 or design evaluation of reactivity control systems should be provided.

[5] 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.

Evaluation of combined performance of reactivity control systems

3.4.8. This section should describe the relevant situations and evaluate the combined functional performance for accidents where two or more reactivity control systems are used.

3.4.9. This section should also include failure analyses to demonstrate that the reactivity control systems are not susceptible to common-cause failures when used redundantly. These failure analyses should consider failures originating within any of reactivity control system as well as those originating from plant equipment other than reactivity systems and should be comprehensively provided with supporting discussion and logic.

Core components

3.4.10. Descriptions of the following aspects should be provided:

(i) 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. the fuel assembly or fuel bundle), related components required for fuel positioning and all supporting elements internal to the reactor, including any separate provisions for moderation and fuel location (description of interfaces); reference should be made to the other sections of the safety analysis report that cover related aspects of the reactor core and also fuel handling and storage;

(ii) The physical and chemical properties of the materials used for the core components, as well as nuclear physics, thermal-hydraulic, structural and mechanical characteristics of the components;

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

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

(v) A consideration of the effects of service on the performance of safety functions, including both 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 should provide relevant information on the reactor coolant system and its associated systems, where possible in the scope of information and format described in Appendix II. The contents of this chapter should demonstrate compliance with the applicable design requirements from SSR-2/1 (Rev. 1) [3] (see the general requirements 21 and 23 to 26 and the specific requirements 47 to 50). Specific guidance for the design of these systems is provided in NS-G-1.9 [26].

3.5.2. A description and justification should be provided to demonstrate that the reactor coolant systems will retain its required level of structural integrity in operational states and accident conditions (for non-affected SSCs). Information on integrity of the reactor coolant pressure boundary should contain the results of the detailed stress evaluations and studies of engineering mechanics and fracture mechanics of all components comprising the reactor coolant pressure boundary 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 systems and its 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 systems should be provided, together with the corresponding applicable 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. A description of design features and justification of the performance should be provided ensuring that the various components of the reactor coolant systems and the subsystems interfacing with the reactor coolant systems meet the safety requirements for design. This should include, where applicable, the reactor coolant pumps in pressurized water reactors (PWR) or recirculation pumps in boiling water reactors (BWR), the steam generators (PWR) or boilers (BWR), 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 (PWR), the pressurizer relief discharge system and the residual heat removal system, including all components such as pumps, valves and supports.

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

Materials

3.5.7. A justification should be provided of the materials used for the components of the reactor coolant system and associated systems, specifically those forming the primary pressure boundary. Information should be provided on the corresponding material specifications, including chemical, physical and mechanical properties, resistance to corrosion, dimensional stability, strength, toughness, crack tolerance and hardness. The properties and required performance of seals, gaskets and fasteners in the pressure boundary should also be considered. The section should address applicable degradation mechanisms and fabrication challenges, including stress corrosion cracking and sensitization of welds explicitly. It should also address the necessary precautions or analysis justifying the adequacy of the chosen materials or processes in light of the above be provided.

Reactor coolant system and reactor coolant pressure boundary

3.5.8. This section should describe measures implemented to ensure the integrity of the reactor coolant systems throughout the plant lifetime, including prevention of cold over-pressurization. In addition, this section should provide information on means of overpressure protection of the reactor coolant pressure boundary including all pressure-relieving devices (isolation, safety and relief valves). Coolant leakage detection provisions should be described, too.

3.5.9. Description should be also provided in this section of the scope of the leak before break concept or break preclusion concept, and its implementation in the reactor coolant systems piping. The description should include monitoring means and analytical demonstration important to ensure limitation of the break size in the reactor coolant systems. It should be also described the implications of the concept used on the design of other systems or components (such as reactor internals) and on the scope of postulated initiating events covered by the analysis in chapter 15.

Reactor vessel

3.5.10. The description of the reactor vessel design should be provided in this section in a manner that is detailed enough to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards. Design information should include the reactor vessel materials, the pressure–temperature limits and the integrity of the reactor vessel, including embrittlement considerations.

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

Reactor coolant pumps

3.5.12. A description and justification should be provided of the performance and design features that have been implemented to ensure that the reactor coolant pumps meet 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 in the event of pump trip to avoid undesirable thermal-hydraulic conditions. The information should present the provisions made to preclude rotor over-speeding and to address cavitation and possible vibration of the reactor coolant pump and associated structures in the event of a design-basis loss of coolant accident. The description should also address seal performance, including performance under prolonged station black-out conditions.

Primary heat exchangers (steam generators)

3.5.13. A description and justification should be provided of the performance and design features that have been implemented to ensure that the steam generators meet the safety requirements for design. The description should include the internal structures of the steam generators and connections to feedwater and steam exit and drains, as well as accesses for inspection and leak detection.

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

3.5.15. Potential effects of heat exchange tube damage and the design criteria to prevent it should be specified, including

(1) Design conditions and plant states considered for the steam generator tubes and accident conditions selected that define the allowable stress intensity limits to be used and the justification for this selection;

(2) Extent of tube-wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined in (1) 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

3.5.16. A description and justification should be provided of the performance and design features that have been implemented to ensure that 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

3.5.17. A description and justification should be provided of the performance and design features that have been implemented to ensure that the reactor pressure control system meets the safety requirements for design. In addition to the pressurizer systems (pressurizer heaters and sprays), these should include also the pressurizer relief tank, the piping connections from the tank to the pressurizer relief and safety valves, the relief tank spray system and associated piping, the nitrogen supply piping, and the piping from the tank to the cover gas analyser and the reactor coolant drain tank.

3.5.18. Distinction should be made between reactor depressurization systems used for design basis accidents and those used for design extension conditions due to the relevance of these systems for the independence of the levels in defence in depth.

Reactor coolant system component supports and restraints

3.5.19. A description and justification should be provided of the performance and design features that have been implemented to ensure the integrity of supports and restraints and their adequacy.

Reactor coolant system and connected system valves

3.5.20. A description and justification should be provided of the performance and design features that have been implemented to ensure that the valves interfacing with the reactor coolant systems meet the safety requirements for design. This description should include safety and/or relief valves, valve discharge lines and any associated equipment.

Access and equipment requirements for in-service inspection and maintenance

3.5.21. In this section, information should be provided on the system boundary, subject to inspection. In particular, components and associated supports should be discussed, including all pressure vessels, piping, pumps, valves, and bolting, covering the following areas:

- Accessibility;
- Examination categories and methods;
- Inspection intervals;
- Provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects;
- System pressure tests.

The programmes and their implementation milestones should be described and reference to any applicable standards made.

Reactor auxiliary systems

3.5.22. This section should provide a description and justification of the performance and design features that have been implemented to ensure that the various connected or associated systems interfacing with the reactor coolant systems meet the safety requirements for design. Selection of the systems to be covered in this section should be done without repetition of the information in other chapters, in particular in chapter 6, chapter 9 and chapter 10.

3.5.23. Examples of the associated systems to be covered in this section include:

- Chemical and inventory control systems for the reactor coolant;
- Reactor coolant make-up and cleanup systems;
- Residual heat removal system;
- reactor coolant systems high point vents;
- Heavy water collection system for pressurized heavy-water reactors;
- Moderator system and its cooling system for pressurized heavy-water reactors;
- Reactor core isolation cooling system for boiling water reactors;
- Isolation condenser system for boiling water reactors.

CHAPTER 6. ENGINEERED SAFETY FEATURES

3.6.1. Chapter 6 should present relevant information on the engineered safety features and associated systems. Engineered safety features to be covered in chapter 6 are understood as those SSCs needed for performing safety functions adequately in case of design basis accidents and design extension conditions including core melt accidents.

3.6.2. Description of the engineered safety features should demonstrate their capability to mitigate the consequences of the accidents and to bring the nuclear power plant to the controlled or safe shutdown state, in accordance with the relevant requirements established in SSR-2/1 (Rev. 1), requirements 51-58 and 65-67 [3].

3.6.3. It is assumed that each group of the systems covered in different sections below will separately address safety systems and safety features for design extension conditions as appropriate, with focus on adequate independence between relevant two levels of defence.

3.6.4. Systems and provisions necessary for transferring heat to the ultimate heat sink/diverse heat sink should be presented with special care and their function of heat transfer for cases of natural hazards exceeding site design basis should be addressed.

3.6.5. The engineered safety features provided in different plant designs may vary. The engineered safety feature explicitly discussed in this chapter are those that are typically used to limit the consequences of postulated accidents in light-water-cooled power reactors, and should be treated as illustrative of the engineered safety feature and of the kind of informative material that is needed.

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

3.6.7. For each of the engineered safety features, detailed description should, as far as reasonable, include the items specified in Appendix II. In describing the materials used in engineered safety feature components, material interactions with fluids that could potentially impair operation of engineered safety features should be taken into account. The description should cover the compatibility of materials 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 or coatings, electrical cable insulation and asphalt.

Emergency core cooling system/Residual heat removal systems

3.6.8. This section should present relevant information on the emergency core cooling system and associated systems. The description should cover both emergency core cooling system designed for heat removal following design basis accidents and safety features for residual heat removal in case of design extension conditions including core melt accidents. Additionally, it should provide relevant information on the high and low pressure safety injection systems and the passive safety injection systems in accordance with the general design aspects presented in Chapter 3 in order to meet the requirement 52 of SSR 2/1 (Rev. 1) [3] and the guidance provided in NS-G-1.9 [26]. Relevant coolant storage tanks should be also described in this section. The actuation logic (protection systems) should be described in Chapter 7 and not be included here.

Emergency feedwater system

3.6.9. This section should provide required information on the emergency feed water system (if not covered in section 10.3) as essential means for residual heat removal through the secondary side of the steam generators in case of pressurized water reactors (accident conditions). The information provided should be linked to general design aspects presented in chapter 3 and should demonstrate compliance with the requirements of SSR-2/1 (Rev. 1) [3] and NS-G-1.9 [26].

Steam dump system

3.6.10. Similarly as the emergency feed water system description above, this section should describe the emergency steam dump system as another essential means for excessive or residual heat removal from the steam system under certain emergency situations; see SSR-2/1 (Rev. 1) [3] and NS-G-1.9 [26]. Optionally, the description of this system can be included in chapter 10 of the safety analysis report.

Emergency borating system

3.6.11. This section should provide information on any means for ensuring reactor shutdown by injecting concentrated boron in addition to those provided by the standard emergency core cooling system.

Corium localization system

3.6.12. This section should provide relevant information on the corium localization system as a necessary means for molten corium solidification either inside the reactor pressure vessel or in a dedicated corium localization system as a necessary precondition for containment basemat protection and ensuring containment integrity in the long-term.

Containment systems

3.6.13. This section should present relevant information on the containment systems incorporated 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 containment functional requirements of SSR-2/1 (Rev. 1) [3] and NS-G-1.10 [27] were met. The section in combination with chapter 15 should provide sufficient demonstration of containment integrity for all plant states and should provide the basis for development of procedures, specification of needed instrumentation, operator response and equipment response.

3.6.14. Description of the systems in this section should include both primary and secondary containment systems. Description and justification of the required performance should be provided for design of the concrete and steel internal structures of the containment. The systems to be covered should include, as applicable:

- Containment active heat removal systems/the containment spray system and other active heat removal systems;
- Containment passive heat removal systems;
- Containment isolation system;
- The system for protection of the containment against overpressure and underpressure;
- The system for control of combustible gases in the containment;
- The containment annulus ventilation system;
- The containment filtered venting system;
- Containment penetrations, airlocks, doors and hatches.

3.6.15. In addition, containment leakage testing system should be described in this section. It should be demonstrated that the containment itself, containment penetrations, and other containment isolation barriers allows for periodic leakage testing as part of the operational programmes. This section should provide sufficient basis for development and implementation of such testing programme; see SSR-2/1 (Rev. 1) [3] and NS-G-1.10 [27]. The following tests should be considered with information on the proposed schedule for performing preoperational and periodic leakage rate tests and the relevant special testing requirements:

- Containment integral leak rate test;
- Containment penetration leak rate test;
- Containment isolation valve leakage rate test.

Habitability systems

3.6.16. This section should present relevant information on the habitability systems. The habitability systems are those engineered safety features provided to ensure that essential plant personnel can remain at their posts, including those in the main and supplementary control rooms, technical support centres, emergency centres as well as other relevant places, needed to take actions to operate the plant safely in operational states and to maintain acceptable conditions in case of accidents. Examples of means for ensuring habitability of control places include shielding, air filtration/purification systems, compressed air storage systems, and other provisions for control of working conditions

3.6.17. Habitability of control places under design extension conditions with core melting should be addressed in this section of the safety analysis report.

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). In addition, the following specific information should be presented to demonstrate the performance capability of these systems: considerations of the coolant pH and chemical conditioning in all necessary conditions of system operation; effects on filters of postulated design basis loads due to fission products; and the effects on filter operability of design basis release mechanisms for fission products.

Other engineered safety features

3.6.19. This section(s) should present relevant information on any other engineered safety features implemented in the plant design and not covered by previous sections. Examples include, but are not limited to: the steam dump to the atmosphere and backup cooling systems. The list of these systems will depend on the type of plant under consideration. It may be decided whether certain systems (such as auxiliary feed water system) are described here or in chapter 9 dealing with auxiliary systems in much broader sense or in chapter 10 dealing with steam and power conversion system.

CHAPTER 7. INSTRUMENTATION AND CONTROL

Instrumentation and control system description

3.7.1. This chapter should provide relevant information on instrumentation and control systems as described in Appendix II. The chapter should describe how the requirements 59 to 67 of SSR 2/1 (Rev. 1) [3] are met. Recommendations regarding design of instrumentation and control systems are provided in SSG-39 [28].

3.7.2. This chapter should identify those instruments and their associated equipment that constitute the plant control and protection systems and those systems relied upon by the operating organization to monitor plant conditions and to shut down the plant and maintain it in a safe shutdown state under accident conditions.

3.7.3. This chapter should establish that instrumentation and control systems and components are qualified for their intended function during their service life in all plant states.

3.7.4 This chapter should also provide information on the instrumentation and control systems not important to safety used to control the plant in normal operational states.

Instrumentation and control system design bases, overall architecture, and functional allocation

3.7.5. This section should identify all instrumentation, control, and supporting systems that are important to safety, including alarm, communication, and display instrumentation and should specify functions allocated to individual systems.

Further on this sub-section should describe:

- instrumentation and control system design basis;
- Classification;
- Defence in depth and diversity strategy;
- Identification of safety criteria.

General design considerations for instrumentation and control systems

3.7.6. This section should describe how the applicable criteria according to the importance to safety of the system are addressed, including:

- Quality of components and modules;
- Software quality;
- Description how the performance requirements of all supported systems are met;
- Potential hazards to the system, including inadvertent actuations, error recovery, self-testing, and surveillance testing;
- Unauthorized access control;
- Redundancy and diversity requirements;
- Independence requirements;
- Fail safe design of the protection systems;
- System testing and surveillances;
- Status of the data communication systems in the design of bypass and inoperable status indications;
- Prevention of a fault propagation path for environmental effects (e.g., high-energy electrical faults and lightning) from one redundant portion of a system to another, or from another system to a safety system;
- Defence in depth and diversity analyses for each potential failure mode, exposure of the system to seismic hazards.

Control systems important to safety

3.7.7. This section should provide relevant information on the control system and demonstrate that Requirement 60 from SSR 2/1 (Rev.1) [3] is met. In particular, information that appropriate and reliable control system is provided to maintain and limit the relevant process variables within the specified operational ranges.

Reactor protection system

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

(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 the chapter of the report on safety analyses;

(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 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) Where the actuation logic for reactor trip is implemented by digital means, a discussion of the software life-cycle activities for digital systems, and the software verification and validation should be provided.

Actuation systems for engineered safety features

3.7.9. This section should provide relevant information on the actuation systems for engineered safety feature actuation system and demonstrate that Requirement 61 from SSR 2/1 (Rev.1) [3] is met.

3.7.10. In some plant designs, the actuation systems for reactor trip and the engineered safety feature actuation system are designed as one single system. In this case it should be demonstrated how independence of safety functions is met and the strategies to protect against common cause failure within the safety systems should be specified.

Diverse actuation system

3.7.11. This section should provide an assessment of level of diversity in digital instrumentation and control system architecture, description of independence of safety functions, application of single failure criterion, consideration of common cause failure, safety classification and qualification requirements.

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

Hazard analysis for instrumentation and control systems

3.7.13. This section should provide relevant information to demonstrate that hazard analysis for instrumentation and control systems consider all plant states and modes of normal operation, including transitions between different modes of normal operation. Degraded states should also be included.

Information systems important to safety

3.7.14. This subsection should describe plant information systems important to safety that includes:

(a) A list of the parameters that are measured and the physical locations of the sensors and the environmental qualification envelope, which should be defined by the most severe operational or accident conditions, and the duration of the time period for which the reliable operation of the sensors is required;

(b) A specification of the parameters monitored by the plant computer and the characteristics of any computer software (scan frequency, parameter validation, cross-channel sensor checking) used for filtering, trending, the generation of alarms and the long term storage of data and displays available to the operating organization s in the control room, the supplementary control room and other emergency response facilities. If data processing and storage are performed by multiple computers, the means of achieving the synchronization of the different computer systems should be described.

3.7.15. Further on, this section should 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 (e.g. valve interlocks at interfaces between low pressure and high pressure fluid systems whose operation could result in an intersystem loss of coolant accident).

Interlock systems important to safety

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

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

Automatic control systems not important to safety

3.7.17. This section should describe the automatic control systems not important for safety. It should be demonstrated that postulated failures of 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.

Data communication systems

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

3.7.19. Justification should be provided to demonstrate that the data communication systems conform to the relevant recommendations in the regulatory guides and industry codes and in the standards applicable to data communication systems.

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

Instrumentation and control in the main control room

3.7.21. 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.

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

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

- Sufficient displays in the control room to monitor all functions important to safety;
- The status of the plant;
- Safety status and trends of the key plant parameters;
- Safety classified indications and controls to implement emergency operating procedures and severe accident management guidelines.

3.7.24. This section should describe how human–machine interface design of the main control room conforms to the human factors engineering programme as described in Chapter 18.

3.7.25. Habitability of main control room, supplementary control room and other emergency response facilities are addressed in other chapter of the safety analysis report (Chapter 6).

Instrumentation and control in a supplementary control room[6]

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

3.7.27. 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,

[6] Emergency response facilities other than the supplementary control room are to be included in this section also.

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.

3.7.28. This section should describe how human–machine interface design of supplementary control room conforms to the human factors engineering programme as described in Chapter 18.

3.7.29. The means of physical and electrical isolation between the plant systems and 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.

3.7.30. The mechanisms for the transfer of 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

3.7.31. This section should describe the instrumentation and control in the emergency response facilities (see Chapter 19 of this Safety Guide) and should demonstrate that Requirement 67 from SSR 2/1 (Rev. 1) [3] is met. In particular, it should be shown that information about important plant parameters and radiological conditions at the plant and in its surroundings, means of communication on- and off- site are provided to the facilities available for the plant staff to perform expected tasks for managing an emergency under conditions generated by accidents and hazards, in some cases including certain control functions.

Digital instrumentation and control systems application guidance

3.7.32. If digital instrumentation and controls systems are used, the overall scope of the application should include information on (1) the design qualification of digital systems, (2) protection against common-cause failure, and (3) functional requirements when implementing a digital protection system. The description should demonstrate that Requirement 63 of SSR 2/1 (Rev. 1) [3] is met.

CHAPTER 8. ELECTRIC POWER

Description of the electrical power system

3.8.1. This chapter 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 should describe how the Requirement 68 from SSR-2/1 (Rev. 1) [3] is met. Specific recommendations regarding the design of electrical power systems are provided in SSG-34 [29].

3.8.3. Chapter 8 should provide definitions, design features and classifications of preferred power supply, off-site power system, on-site power system, standby power system, and alternate AC power system.

3.8.4. Chapter 8 should also provide relevant information on how the safety power system can be supplied, i.e. by either the preferred power supplies or the standby power sources. A description should include the alternate AC power system that supplies the safety power systems in design extension conditions.

General principles and design approach

3.8.5. Among the safety design criteria, rules and regulations, the following information specific to electrical systems should be described:

(a) Anticipated electrical events considered in the design with all functional requirements 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 maintain safety functions and to remove decay heat from spent fuel for the period for which the plant is in a station blackout condition;

(d) The design for reliability (redundancy, independence, diversity);

(e) The possibility of common cause failures, which could render the safety power systems unavailable to perform their safety functions when called upon, in the design, maintenance, testing and operation of the safety power systems and their support systems;

(f) The plant specific divisions of the electrical power systems, including various system voltages and designation of parts of the system that are considered to be essential;

(g) Substantiation 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) and 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).

Off-site power systems

3.8.6. 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.

3.8.7. This section should also provide the off-site power system design requirements such as 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.

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

On-site AC power systems

3.8.9. This subsection should provide relevant information on the plant specific AC power system. 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 non-interruptible AC power system available for anticipated operational occurrences and design basis accidents.

3.8.10. This subsection should describe the power requirements for each plant AC load, including: (a) the steady state load; the start-up kilovolt-amperes for motor loads; (b) the nominal voltage; (c) the allowable voltage drop (to achieve full functional capability within the required time period); (d) the sequence and time necessary to achieve full functional capability for each load; (e) the nominal frequency; (f) the allowable frequency fluctuation; (g) the number of trains, and the minimum number of trains of engineered safety features to be energized simultaneously.

3.8.11. This subsection should describe how:

(a) On-site AC power system breakers are co-ordinated to ensure the reliable delivery of emergency power to engineered safety features and non-interruptible AC power system loads;

(b) In loss of off-site power condition, the standby AC power source is started and safety loads are sequenced to the safety buses without the overloading a primary mover, and in time frames consistent with the assumptions presented in the chapter 15 on safety analyses;

(c) In accident conditions with a subsequent loss of off-site power, the required safety loads can be sequenced onto the standby AC power source in case of design basis accidents without overloading the primary mover and in time frames consistent with the assumptions presented in the chapter 15 on safety analyses;

(d) Non-interruptible AC power is continuously provided to essential safety systems and important to safety instrumentation and control systems while normal off-site AC power systems are available and during postulated loss of off-site power events;

(e) An alternate AC power supplies is provided at the nuclear power plant if the plant's design depends on AC power to bring the plant to a controlled state following loss of off-site power and safety standby power sources considering the diversity (e.g. not susceptible to the events that caused the loss of on-site and off-site power sources), sufficient capacity to operate systems necessary for coping with a station blackout, and auxiliaries qualified for their intended use;

(f) There are adequately robust features to enable the safe use of non-permanent equipment to restore the necessary electrical power supply in core melt accidents (see Requirement 68, para 6.45A from SSR-2/1 (Rev. 1) [3]).

On-site DC power systems

3.8.12. This subsection should provide relevant information on the DC power system. This includes the description of 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:

- 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);
- Major DC loads present (including the non-interruptible AC power system inverters and any DC loads not important to safety such as the lubrication oil pumps for the turbine bearings);
- A description of the fire protection measures for the DC battery vault area and cable systems.

3.8.13. The power requirements for each plant DC load should be justified, including:

- Steady state load;
- Surge loads (including emergency conditions);
- Load sequence;
- Nominal voltage;
- Allowable voltage drop (to achieve full functional capability within the required time period);
- Number of trains;
- Minimum number of engineered safety feature trains to be energized simultaneously (if more than two trains are provided).

3.8.14. This subsection should demonstrate continuity of DC power supply so that the monitoring of the key plant parameters and for the completion of short term actions necessary for safety is maintained in the event of loss of all the AC (alternating current) power sources.

Electrical equipment, cables and raceways

3.8.15. This subsection should demonstrate that 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. In the justification, account should be taken of the cumulative radiation effects and thermal ageing expected over their service life. Seismic qualifications and fire resistance of buses, cable trays and their supports should be also described.

3.8.16. This subsection should identify at least three classes of cables: (1) control and instrumentation cables, (2) low voltage power cables (e.g. 1000 V or less), and (3) medium voltage power cables (e.g. 20 kV or less).

3.8.17. This subsection should describe the environmental qualification of cables 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.

Grounding and lightning protection

3.8.18. A description should be provided 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 be also included.

3.8.19. The industry-recognized consensus standards used in designing the subsystems should be identified, as well as the bases for the related acceptance criteria. Analyses and any underlying assumptions used should be provided 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 has two main parts. Part A of chapter 9 should provide information about the auxiliary systems not included in other chapters. In particular, this chapter should identify systems that are essential for safe shutdown of the plant or for protection of the health of the public. For each system, the description should, to the extent possible, follow the structure given in Appendix II. Description of auxiliary systems should meet the requirements 69, 71-74, 76 and 80 from SSR-2/1 (Rev. 1) [3]. Recommendations on safety design of auxiliary systems are provided in DS440 [30] *(editorial note: reference conditioned to publication schedule)*.

3.9.2. Part B of chapter 9 should describe civil structures of the plant. This part should describe how various civil structures in the plant comply with the general design requirements and other rules specified in chapter 3. For each civil structure the description should, to the extent possible, follow the structure of information given in Appendix II. Design of civil structures should follow the general design rules using recognized engineering practices, as stated in SSR-2/1 (Rev. 1) [3], requirements 11, 17, 18 and 58.

3.9.3. It is clear that both plant auxiliary systems as well as civil structures can vary between the designs. The examples of subsystems provided below are not therefore intended to represent a complete list of systems to be discussed in this chapter of the safety analysis report. The structure of the chapter can be modified accordingly to the specificities of the design, also taking into account information provided in other chapters of the safety analysis report.

9A AUXILIARY SYSTEMS

Fuel storage and handling systems

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

- New fuel storage and handling system;
- Spent fuel storage and handling system;

- Spent fuel pool cooling and clean-up system;
- Handling systems for refuelling fuel casks.

3.9.5. For fresh fuel, information provided should include considerations such as packaging, storage, criticality prevention and fuel integrity monitoring and control.

3.9.6. For reprocessed and irradiated fuel, 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 to ’practical elimination’ of severe fuel damage in a spent fuel pool.

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

Heat transport systems

3.9.8. This section should provide relevant information on the water systems associated with the plant. It should include, in particular the following systems:

- Service water system;
- Component cooling water system for reactor auxiliaries (intermediate cooling circuits);
- Essential chilled water system;
- De-mineralized water make-up system;
- Ultimate heat sink system;
- Condensate storage and transfer system.

3.9.9. Robustness of the systems necessary for transfer of residual heat to the ultimate heat sink system and of the heat sink itself in 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 separately in another section of this chapter, while the chemical control and volume control system was already covered in chapter 5.

Air and gas systems

3.9.11. The air systems that provide station air for service and maintenance uses should be described in this section, including compressed air systems and service gas systems. A description should be also provided of the capabilities to interconnect and/or isolate the instrumentation and control air system from the station 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, air conditioning subsystems should be covered:

- Control room[7] heating, ventilation, air conditioning;
- Spent fuel pool area heating, ventilation, air conditioning;

[7] It also applies to the supplementary control room and to other emergency response facilities

- Auxiliary and radioactive waste area heating, ventilation, air conditioning;
- Turbine building heating, ventilation, air conditioning;
- Engineered safety features heating, ventilation, air conditioning;
- Chilled water system for heating, ventilation, air conditioning.

Fire protection systems

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

3.9.14. The extent to which the design has been successful in providing adequate fire protection should be assessed; this section may refer to other sections of the safety analysis report for this information (e.g. the chapter 15 on safety analysis). Where appropriate, the provisions to ensure the fire safety of personnel should also be described in this section.

Support systems for diesel generators or for gas turbine generators

3.9.15. Support systems for the diesel generators (or for gas turbines) should be covered by this section. The electrical part of the AC systems has been already covered in chapter 8. 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. The following subsystems should be typically addressed in this section:

- Diesel generator (or gas turbine generator) fuel oil storage and transfer system;
- Diesel generator (or gas turbine generator) cooling water or cooling air system, as applicable;
- Diesel generator (or gas turbine generator) starting system;
- Diesel generator (or gas turbine generator) lubrication system;
- Diesel generator (or gas turbine generator) combustion air intake and exhaust system.

Overhead heavy-load handling system

3.9.16. The overhead heavy-load handling system should be described in this section with the associated safety requirements. Related rules and assumptions for design should be given and justified. Focus should be given on critical load handling operations with potential effect on performance of safety functions.

3.9.17. Information to be provided should include: (a) parameters defining the load that, if dropped, would cause the greatest damage; the areas of the plant where the load would be handled; (b) the design of the overhead heavy-load handling system; (c) and the operating, maintenance and inspection procedures applied to the load handling system. The following systems should be described in particular:

- Reactor building crane;
- Fuel building crane.

Miscellaneous auxiliary systems

3.9.18. This section should provide relevant information on any other plant auxiliary system whose operation may 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:

- Equipment and floor drainage system;
- Communications systems;
- Interfacing water systems (raw water reserves, demineralized water system, potable and sanitary water system).

9B CIVIL ENGINEERING WORKS AND STRUCTURES

3.9.19. Part 9B of the safety analysis report should describe how general design requirements specified in chapter 3 have been complied with in the design of nuclear power plant specific structures. Three groups of civil structures should be considered: foundations, reactor building[8], and other civil structures. In description of the structures, the unified format of 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 defined safety requirements of 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 and/or seismic classification for buildings and structures have been used, the basis of the classification should be described for the design option outlined. It should be demonstrated that the safety classification of buildings containing items important to safety is consistent with the classification of structures, systems and components that it contains; see NS-G 1.6 [31];

(d) If a structure is intended to provide separate additional functions 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, if appropriate.

Foundations and buried structures

3.9.21. In this section, information on foundations should be provided, including plan and section views, in order to define the primary structural aspects and elements relied on to perform the foundation function. The description should include soil-structure interaction (see NS-G-3.6 [14]). Additionally, the type of foundation and its structural characteristics and the general arrangement of each foundation should be presented. In particular, foundations of steel or concrete containment should be discussed, as well as all seismically classified structures.

Reactor building/Containment

3.9.22. This section should describe design features of the reactor building provided to comply with the applicable safety requirements of SSR-2/1 (Rev. 1) [3], including requirements 53 to 56, in accordance with NS-G 1.10 [27]. Specific design features of the primary containment such as its leak tightness, mechanical resistance, pressure retaining capability and protection against hazards should be covered. Concrete and steel internal structures of the containment should be described. If the design incorporates a secondary containment, this should also be described here.

3.9.23. This section should also provide sufficient information to demonstrate containment performance in all plant states and combination of loads in accordance with the acceptance criteria established; see NS-G-1.10 [27].

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[8] Reactor building means a building covered with the primary containment or the secondary containment.

Other structures

3.9.24. Similarly as in previous cases, other civil structures of the plant that are relevant to nuclear safety, should be described in this section.

CHAPTER 10. STEAM AND POWER CONVERSION SYSTEMS

3.10.1. Chapter 10 should provide information on the design of plant steam and power conversion systems. The information provided should follow to the applicable extent the structure specified in Appendix II and demonstrate how the system design meets the Requirement 77 from 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 generator(s) in operational states;

(b) A description of the main steam line piping and the associated control valves, the main condensers, the main condenser evacuation system, the turbine gland sealing system, the turbine bypass system, the circulating water system, the condensate clean-up system, the condensate and feedwater system, and, where applicable, the steam generator blowdown system;

(c) The water chemistry programme, together with a discussion 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. The chapter 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 radioactivity. 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.

3.10.3. Where appropriate, this chapter should summarize the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems.

Role and general description

3.10.4. In this section, a summary description indicating 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.

Main steam supply system

3.10.5. In this section, the main steam supply system and main steam line piping should be described, including piping and instrumentation diagrams showing system components, including interconnecting piping.

3.10.6. Descriptions should include sufficient details for ensuring reliable performance of safety functions, including fast and reliable isolation and steam relief. Demonstration that separation of steam lines prevents leakage from one affecting the other and protection against aircraft crash should also be included.

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

3.10.8. For the pressurized water reactor plants, description of the main steam system should extend from the connections to the secondary sides of the steam generators up to and including the turbine stop valves and includes the containment isolation valves, safety and relief valves, connected piping of larger 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. Steam bypass/dump station to the atmosphere (if not included in chapter 6) may be also described in this section.

Feedwater systems

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

3.10.10. The description should include analysis of component failure and of the effects of equipment malfunction on the reactor coolant systems and an analysis of detection and isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break and/or degradation of the integrity of safety-related equipment.

Turbine generator

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

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

3.10.13. The section should describe the turbine generator system equipment design and design bases, including the performance requirements under operating conditions. It should also describe the intended mode of normal operation (e.g. base load or load following), functional limitations imposed by the design or operational characteristics of the reactor coolant systems (e.g., the rate at which the electrical load may be increased or decreased by means of reactor control rod motion or steam bypass), and design codes to be applied.

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

Turbine and condenser systems

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

- Main condenser;
- Condenser air extraction system (off-gas treatment in boiling water reactor);
- Circulating water system;
- Condensate system;
- Condensate clean-up system;
- Turbine auxiliary systems;
  - turbine gland sealing system
  - turbine by-pass system to the condenser
- Generator auxiliary systems.

Steam generator blowdown processing system

3.10.16. The steam generator blowdown processing system and its design basis should be described in this section in terms of its ability to maintain optimum secondary-side water chemistry in recirculating steam generators of pressurized water reactors during normal operation and anticipated operational occurrences (e.g., main condenser in-leakage and primary-to-secondary leakage).

3.10.17. The design bases should include consideration of expected and design flows for all modes of normal operation (i.e., process and process bypass), process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process instrumentation and control for maintaining operations within established parameter ranges.

Break preclusion implementation for main steam and feedwater lines

3.10.18. This section should describe the scope of the break preclusion implementation in the main steam and feedwater lines. Those aspects should be emphasized which are important from the viewpoint of the direct impact on the plant safety (either direct effects on performance of the fundamental safety functions, or indirect effects like secondary damage of the plant systems e.g. by pipe whip or extraordinary pressure loading). If relevant, the description should include how the leak before break concept has been implemented.

CHAPTER 11. RADIOACTIVE WASTE MANAGEMENT

3.11.1. This chapter should describe the adequacy of the measures proposed for the safe management of radioactive waste of all types that is generated throughout the lifetime of the plant. Treatment of radioactive waste is covered by requirements 78 and 79 from SSR-2/1 (Rev. 1) [3] and by Requirement 21 from SSR-2/2 (Rev. 1) [4]. Further information on matters to be covered in this chapter of the safety analysis report is provided in GSR Part 5 [32] ("Predisposal Management of Radioactive Waste"); GSG-3 [33] ("The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste"); SSG-40 [34] ("Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors").

3.11.2. More specifically, this chapter should describe among others:

1. The capabilities of the plant to control, collect, handle, process and store liquid, gaseous, and solid wastes that may contain radioactive materials, and
2. The instrumentation used to monitor the releases of radioactive wastes, both on-site and off-site.

Disposal of the waste takes place in a dedicated facility (final radioactive waste repository) and is therefore not covered in this chapter.

3.11.3. Sources of radioactive waste described in this chapter should cover radioactive wastes generated during normal operation (i.e. in different operational activities, such as refuelling, purging, equipment downtime and maintenance). Radioactive wastes potentially generated during anticipated operational occurrences and accident conditions should be determined and described separately in chapter 15.

3.11.4. Sections below should provide relevant information on the radioactive waste processing (i.e., pretreatment, treatment and conditioning) systems. They should include description of the design features of the facilities that control, collect, handle, process and store solid, liquid and gaseous forms of radioactive waste arising from all activities on the site throughout the lifetime of the plant. Conditioning of liquid and solid waste for future disposal should be also covered. The description should include the SSCs provided for these purposes and also the instrumentation incorporated to monitor possible leaks or escapes of radioactive waste. Scope and structure of the description of systems for the processing of radioactive wastes should follow, to the extent practicable, the common structure specified in Appendix II.

Source terms

3.11.5. Description of the main sources of solid, liquid and gaseous waste and estimates of their generation rate and their normal operational releases, in compliance with the design requirements, should be provided in this section.

3.11.6. Assessment of gaseous and liquid releases resulting from anticipated operational occurrences and accident conditions should be covered in chapter 15 and used as input here.

3.11.7. This section should also provide information on the accumulation rates and the quantities, conditions and forms of radioactive waste resulting from normal operation, and on the methods and technical means for its processing, storage and transport. Information on radioactive waste resulting from accidents may be derived from safety analysis results reported in Chapter 15.

3.11.8. The consideration of waste should cover solid, liquid and gaseous waste, as appropriate, in all stages of the development of measures to deal with radioactive waste safely throughout the lifetime of the plant. This section should consider the options for the safe predisposal management of waste.

3.11.9. Measures to minimize the accumulation of waste generated 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. Measures should also be aimed at minimizing both the volume and the activity of the waste in such a way as to meet any specific requirements that may be posed by the design of the waste storage facility.

Liquid waste management systems

3.11.10. This section should describe the capabilities of the plant to control, collect, process, handle, and store liquid radioactive waste generated during operation and resulting from accident conditions.

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

- Control and containment of waste, including proposals to categorize and separate it, as necessary;
- Handling of waste, including provisions for its safe handling while transporting it from the point of origin to the specified storage point. A consideration of the possible need to retrieve waste at some time in the future, including during the decommissioning stage, should be made;
- Processing of waste in accordance with established procedures, taking into account the interdependences among all steps in the predisposal management of radioactive waste, as well as the impact of 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 case preferences for waste disposal change over the lifetime of the plant. The possible need for specialized systems to deal with issues of processing, such as volatility, chemical stability, reactivity and criticality, should be addressed, and any such system in place should be described;
- Storage of waste, its quantities, types and volumes. 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 long term storage, such as cooling, containment, volatility, chemical stability, reactivity and criticality, should also be addressed, and any such system in place should be described.

3.11.12. This section should include assessment of liquid discharges during operational states. Assessment of radioactive releases in accident conditions and resulting radiological consequences should be included in chapter 15.

3.11.13. This section should also address the conceptual solutions available to deal with potentially large volumes of contaminated water generated under accident conditions.

Gaseous waste management systems

3.11.14. This section should describe the capabilities of the plant to control, collect, process, handle and store gaseous radioactive waste generated during operation.

3.11.15. This section should also include assessment of gaseous discharges during operational states. Assessment of radioactive releases in accident conditions and resulting radiological consequences should be included in chapter 15.

Solid waste management systems

3.11.16. This section should describe the capabilities of the plant to control, collect, handle, process, package, and temporarily store - prior to shipment - wet and dry solid radioactive waste generated during operation. In this section, the term "solid waste management system" means a permanently installed system.

3.11.17. Similarly as in the case of liquid wastes, information provided for solid waste should cover their control, handling, processing, storage and preparations for safe transport of waste to another facility for long term storage or disposal.

Process and effluent radiological monitoring including on-site and off-site monitoring

3.11.18. This section should describe the systems and equipment that monitor the process and effluent streams in order to control releases of radioactive materials generated in operational states and accident conditions. This section should also demonstrate that the means of radiation monitoring are in accordance with Requirement 82, paras 6.77 to 6.82, from SSR-2/1 (Rev. 1) [3] and those for off-site monitoring with para 6.84 of the same reference.

CHAPTER 12. RADIATION PROTECTION

3.12.1. This chapter should provide information on the policy, strategy, methods and provisions for radiation protection. The expected occupational radiation exposures during operational states, including measures to avoid and restrict exposures, should also be described. However, public exposure for all plant states, including determination of doses during normal operation, should be addressed separately in chapter 15 of the safety analysis report.

3.12.2. Potential radiation exposures of workers in the nuclear power plant under accident conditions, including those with core melting, should be addressed and the means and other measures taken to minimize the exposure described.

3.12.3. This chapter should deal only with radiation exposure of occupationally exposed workers in the nuclear power plant.

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

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

3.12.6. Scope of the information provided in this chapter should reflect high level safety requirements on relevant design provisions and on the operational radiation protection programme established in accordance with SSR-2/1 (Rev. 1) [3] and SSR-2/2 (Rev. 1) [4], with additional recommendations provided in *DS453 [35], (“Occupational Radiation Protection”; draft Safety Guide in Step 12).*

As low as reasonably achievable considerations

3.12.7. This section should provide a description of the implemented design provisions and operating organization’s policy and the operational application of the ALARA principle both in operational states as well as in accident conditions for the entire lifetime of the plant, including decommissioning. It should be in line with the general design requirements (chapter 3).

3.12.8. Specific measures taken to comply with ALARA principle should be described. This section should provide the estimated annual occupancy of the plant's radiation areas during normal operation and in anticipated operational occurrences. The necessity of workers' presence in certain plant areas where radiation levels are high should be investigated, in order to limit working hours in those areas and, consequently, to reduce radiation doses to workers.

Radiation sources

3.12.9. This section should provide a description of all on-site radiation sources existing both in operational states as well as in accident conditions, with account taken of both contained and immobile sources, and potential sources of airborne radioactive material.

3.12.10. The sources should include contained and immobile radiation sources (such as reactor core; reactor coolant; chemical and volume control system; spent fuel pool cooling system; liquid, gaseous and solid radioactive waste systems -determined consistently with chapter 11-; residual heat removal systems; spent fuel; irradiated control rods and other core internals) as well as sources of airborne radioactive material (such as leakages from systems and equipment for transport of radioactive fluids; activation of air and gaseous leakages from distribution of coolant from spent fuel pool affecting containment atmosphere; fuel building atmosphere and auxiliary building atmosphere).

3.12.11. Special source terms should be discussed for accident conditions including design extension conditions with core melting. Quantitative characteristics of different sources should be provided.

3.12.12. This section should also describe possible pathways of radiation exposures for the workers in the nuclear power plant associated with the potential sources in all operational states as well as in accident conditions.

Radiation protection design features

3.12.13. This section should provide description of the design features of the equipment and the facility that ensure radiation protection. It should provide information on the variety of means for minimizing the source term, minimizing the total working time in a radiation zone, measures taken to lower the radiation level of the area around any equipment or component, to reduce the generation of activated corrosion products and minimize their transport and deposit.

3.12.14. Description of the means for reducing the radiation exposure should cover among others:

- Minimizing contamination by choosing more corrosion-resistant material, using adequate water chemistry regime, enhancing the purifying capacity of the primary coolant and decontaminating the facilities, use of shielding, remote control and other staff actions, and shortening exposure time to reduce external exposure;
- Reducing internal exposure by isolation, ventilation, decontamination and use of protective clothing and respiratory equipment;
- Dividing the plant areas by radiation and contamination level into zones and restricting access to controlled area;
- Categorizing the plant personnel by working conditions and carrying out corresponding control and supervision;
- Monitoring individuals and working areas;
- Establishing signs to avoid inadvertent access and the resulting unnecessary exposure.

3.12.15. The principles of radiation protection applied in the design should be described, including description of means implemented to ensure that:

(a) No person receives doses of radiation in excess of the authorized dose limits as a result of normal plant operation;

(b) The occupational exposures in all plant states are ALARA;

(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 prevent exposure due to accidents with radiological consequences;

(f) All practicable steps are taken to mitigate the radiological consequences of any accident.

3.12.16. This section should also provide complementary information (in addition to information already provided in Chapter 11) for the monitoring of all significant radiation sources, in all activities throughout the lifetime of the plant. It should include individual monitoring by personal dosimeters and workplace monitoring in accordance with Requirement 82, para 6.83, from SSR-2/1 (Rev. 1) [3].

3.12.17. This section should contain description of the instrumentation for fixed area monitoring of radiation and continuous monitoring of airborne radioactive material. In addition, it should provide the criteria for the instrumentation selection and placement, and should address design provisions for any decontamination of equipment, if necessary.

3.12.18. Means for monitoring and decontamination of personnel should be described. This should include adequate provisions for monitoring during operational states, design basis accidents and design extension conditions including where appropriate severe accidents.

Dose assessment

3.12.19. Radiation dose targets for the plant staff in all plant operating states should be stated here, consistently with Chapter 3 (see para 3.3.7). The section should demonstrate that the established dose targets are achievable in plant operational states and accident conditions. Assessment of potential effective and equivalent doses from different sources of radiation and for various staff activities should be presented.

3.12.20. Dose assessment should be based on radiation monitoring (if already available, during plant operation), on operational experience from similar plants or on appropriate computational models. Data from similar plants and description of computational models should be provided in the safety analysis report or should be adequately referred to.

Operational radiation protection programme

3.12.21. This section should describe (consistently with operational programmes described in Chapter 13) the administrative measures, the equipment, instrumentation, facilities and procedures for the radiation protection programme. It should be demonstrated that the plant radiation protection programme 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 of workers to different management levels;

(b) The designation and functions of qualified experts, as appropriate;

(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 classification of working areas and access control;

(e) Issuance of radiation protection procedures and relevant documents to radiation protection personnel, local rules and supervision of work;

(f) Monitoring of individuals and the workplace, keeping investigation records on radiation and contamination in the plant, results of processes and area monitoring and other radioactive information;

(g) Limiting the number of personnel for working in the controlled areas, and management of work planning and work permits;

(h) Protective clothing and protective equipment;

(i) Facilities, shielding and equipment;

(j) Establishment and storage of permanent records on dose equivalent of plant personnel, health surveillance of plant personnel;

(k) Application of the principle of optimization of protection;

(l) Source reduction;

(m) Strengthening the training, retraining and personnel qualification review;

(n) Investigation and reporting of any radiation accidents, and taking corrective actions against recurrence of such an accident;

(o) Arrangements for response to emergencies.

CHAPTER 13. CONDUCT OF OPERATIONS

3.13.1. In this section it should be described how the operating organization takes over its prime responsibility for safety in the operation of a nuclear power plant in accordance with the requirements included in SSR-2/2 (Rev. 1) [4]. More specifically, the chapter should address:

- Important operational issues relevant to safety;
- Approaches adopted by the operating organization to address the identified issues by implementing relevant operational programmes;
- Provisions taken by the operating organization to establish and maintain an adequate staff and the related technical competence, skills and the operating procedures to be followed by the operator to ensure public health protection and safety.

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

Organizational structure of 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 and advisory panels) should also be described. The description of the organizational structure should allow to verify that all the management functions for the safe operation of the power plant, such as policy making functions, operating functions, supporting functions and reviewing functions, are adequately addressed.

3.13.4. The description should cover the functions and responsibilities of individual organizational items and the process for qualification of operating personnel and should be directed to activities that include design, manufacturing, construction, commissioning, operation and decommissioning of both the plant and the plant configuration control.

3.13.5. This section should also identify qualification requirements for the key staff.

Training

3.13.6. This section should provide information allowing verifying that the 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. Information provided should include the initial qualification requirements, the staff training programme, refresher training and retraining and the applicable documentation system the current positions for plant staff. Training programmes and facilities, including simulator facilities, should reflect the status, characteristics and behaviour of the plant units, and should be briefly described.

3.13.7. It should be described in this section how a systematic approach to training is to be adopted, including consideration of updating 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, this section should describe the system and explain the provisions that will be put in place to comply with the licensing requirements.

Operational safety programme implementation

Conduct of Operation

3.13.9. Operational programmes are specific programmes performed by the operating organization to ensure the adequate state of the plant towards relevant requirements in terms of safe operation. This section of the safety analysis report should sufficiently describe such programmes or indicate the plans for implementation such programmes in future stages of the nuclear power plant implementation.

Maintenance, surveillance, inspection and testing

3.13.10. In this sub-section the safety analysis report should provide a description and justification of the arrangements that the operating organization intends to identify, control, plan, execute, and review maintenance, surveillance, inspection and testing practices that influence reliability and affect nuclear safety.

3.13.11. The surveillance programmes should be described including predictive, preventive and corrective maintenance activities to be conducted to control potential degradations of structures, systems and components and to prevent failures; see Requirement 31 from SSR-2/2 (Rev. 1) [4]. In addition, it should be demonstrated that the surveillance programme is adequately specified to ensure compliance with the operational limits and conditions.

3.13.12. This sub-section should also include information about approaches and methods used in demonstrating the appropriateness of the plant inspections, including in-service inspections. In particular, emphasis should be placed on the adequacy of the in-service inspections for 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. The operating organization should also identify the different types of testing that can affect the safety functions of a nuclear power plant and the way for ensuring that testing is initiated, carried out and confirmed within the timescales allowed.

Core management and fuel handling

3.13.14. This sub-section should describe how the operating organization makes the necessary arrangements for all 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, e.g. those used in safety analysis in chapter 15.

3.13.15. It should also be shown 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 defaults of 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.

Management of ageing

3.13.16. This sub-section should describe all parts of the plant that can be affected by ageing and should present the proposals made for addressing the selected issues identified, in accordance to the safety relevance of SSCs. The long term operation programme focused on ageing management should be described; the description should cover appropriate material monitoring and sampling programmes needed for verification of the ability of equipment and the structures, systems and components to perform their safety function throughout the lifetime of the plant. Appropriate consideration should be given to the feedback of operational experience with respect to ageing. Recommendations on ageing management are provided in *DS485 [36] ("Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants", Draft Safety Guide Step 10, revision of NSG-2.12).*

Control of modifications implementation

3.13.17. The operating organization should in this sub-section 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, where necessary. Recommendations regarding plant modifications are provided in NS-G-2.3 [37].

3.13.18. It should be confirmed in this sub-section that the modification control process covers safety significant changes made to the plant systems and components, operational limits and conditions, plant procedures and process software, permanent and temporary changes to the plant.

Programme for the feedback of operating experience

3.13.19. This sub-section should present a programme for the feedback of operating experience to be implemented. The programme should provide measures to ensure that operational events and incidents on the given plant and on other relevant nuclear power plants are identified, recorded, notified, investigated internally, and used to incorporate, when appropriate, lessons for its own operation. The programme should include consideration of technical, organizational and human factor aspects. More detailed guidance is provided in *DS479 [38] ("Operating Experience Feedback for Nuclear Installations", draft Safety Guide Step 9, revision of NS-G-2.11).*

Documents and records

3.13.20. Information on the management system provisions for the records and reports relevant for the operation of the plant over its lifetime should be provided in this sub-section. The associated retention times should be taken in accordance with the level of importance towards plant licensing, operation and decommissioning. In particular, this should include the operating organization's documentary provisions for the management of plant configuration, as well as the management of waste and decommissioning of the plant.

Outages

3.13.21. In this sub-section, a description should be provided of the relevant arrangements for conducting periodic shutdowns of the reactor as the operating cycle and safety or performance improvements. Description on how the plant configuration in accordance to OLCs and safety analysis report is maintained should be given in this section.

Plant procedures and guidelines

3.13.22. This section should describe all relevant documents that will be used by the plant staff to ensure that procedures and guidelines for normal operation, anticipated operational occurrences and accident conditions are conducted in intended manner. It is not expected to include here detailed written procedures. Depending on the stage of the project, the safety analysis report should either provide preliminary arrangements and schedules for their preparation, or should provide a brief description of the nature and content of the procedures and guidelines, and a schedule for their preparation. Three categories of procedures and guidelines, respectively, should be covered as described below.

Operating procedures

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

Emergency operating procedures

3.13.24. This sub-section should provide a description of the procedures that will be used by the operating organization in anticipated operational occurrences or in accident conditions (mainly in design basis accidents). A justification of the selected approach should be provided. Both event based approaches and symptom based approaches can be used and, where appropriate, linked to the results of the plant safety analyses. The required operator actions to diagnose and deal with accidental conditions should be covered appropriately. The approach used for verification and validation of the procedures should be presented, including, when it applies, human factor engineering (see chapter 18). More detailed guidance on the development and implementation of emergency operating procedures is provided in *DS483 [39] ("Severe Accident Management Programmes for Nuclear Power Plants", draft Safety Guide step 10, revision of NS-G-2.15).*

Severe accident management guidelines

3.13.25. This sub-section should provide a description of the selected approach to plant accident management. The corresponding severe accident management guidelines (SAMG) developed to prevent severe 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 SAMG are provided in *DS483 [39].*

3.13.26. In relevant cases, multi-unit events, contingency for an alternate water and electric power supply as well as degraded regional infrastructure should be addressed. It should also be confirmed that severe accident management guidelines have been developed in a systematic way, with account taken of:

- The results from the plant specific severe accidents analysis;
- The identified vulnerabilities of the plant to such accidents;
- The strategies selected to deal with these vulnerabilities.

Nuclear security

3.13.27. Security issues are usually dealt with separately according to special regulations, and the related documents are withheld from public disclosure. Although applicant's plans for physical protection of the facility are described in a separate and confidential part of the application, this section of the safety analysis report should allow to verify that such plans have been prepared according to the applicable Nuclear Security Standards (see NSS-13 [40] and NST-023 [41]) and that can be reviewed by the regulatory body. Optionally, a short description of the security programme for the site and the implementation schedule for the programme can be provided in this section

3.13.28. This confidential section should indicate how the operating organization ensures that the implementation of safety requirements and security requirements satisfies both safety and security objectives without compromising each other, in accordance with Requirement 17 from SSR-2/2 (Rev. 1) [4] and with Requirement 5.13 from NSS-13 [40]. This includes the establishment of an effective system to address safety and security aspects in a coordinated manner and involving all interested parties, together with the identification of specific provisions important for integration of safety and security.

CHAPTER 14. PLANT CONSTRUCTION AND COMMISSIONING

3.14.1. Chapter 14 should include demonstration that the nuclear power plant will be suitable for service prior to entering the construction stage, in accordance with SSR-2/1 (Rev. 1) [3], Requirement 11, and with SSR-2/2 (Rev. 1) [4], Requirement 25 (paras 6.14 and 6.15).

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

3.14.3. A link from the plant safety justification to the commissioning programme should be demonstrated. The commissioning programme should, among other things, confirm that the separate plant items important to safety will perform within their specifications and ensure that the safety functions can be reliably performed.

3.14.4. In addition, as part of the commissioning programme, validation of the operating procedures which is conducted with the participation of future operating personnel should be justified.

3.14.5. This chapter should also present the details of the commissioning organization, including the appropriate interfaces between design, construction and operating organizations during the commissioning period, which should include any provisions for additional personnel and their interactions with the commissioning organization.

3.14.6. It should also be shown that qualified operating personnel at all levels will be directly involved in the commissioning process. The processes established to develop and approve test procedures, to control test performance and to review and approve test results should be described in detail. This should include the actions to be taken when the initial outcomes of the tests do not fully meet the design requirements.

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

3.14.7 Specific information to be included in the safety analysis report prior to plant construction should include:

- Description of the major stages of the initial test programme and discussion of the overall test[9] objectives and general prerequisites for each major phase;
- Description of preoperational stage and/or commissioning planned for each unique or first-of-a-kind design feature, including specification of the test method and test objectives;
- Plans to follow guidance in applicable regulatory guides in the development and conduct of the initial test programme;
- Plans for the utilization of available information on plant operating experiences to establish where emphasis may be warranted in the test programme;
- Description on the overall schedule, relative to the expected fuel loading date, for developing and conducting the major stages of the test programme;
- Plans pertaining to the trial use of plant operating and emergency procedures during the period of the initial test programme;
- General plans for the assignments of additional personnel to supplement plant operating and technical staff during each major stage of the test programme.

[9] Test includes such as vendor inspections, welding inspections, leak tightness test and pressurised test for pressure boundary and fuel assembly inspections at fuel fabrication facility and the site, prior to non-nuclear commissioning for each SSC in construction stage.

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

3.14.8. Specific information to be included in the safety analysis report prior to plant commissioning should include updated information on:

- Description of the major stages of the commissioning programme, including both non-nuclear testing, comprising individual pre-operational tests, overall pre-operational systems tests, structural integrity tests, integrated leakage tests for the containment and of the primary and secondary and system and nuclear testing, comprising initial fuel loading, subcritical tests, initial criticality tests, low power tests and power ascension tests and the specific objectives to be achieved for each major stage;
- Description of the organizational units and any external organizations or other personnel that will manage, supervise, or execute any stages of the commissioning programme;
- Description of the system that will be used to develop, review, and approve individual commissioning procedures, including the organizational units or personnel that are involved and their responsibilities;
- Description of the administrative controls that will govern the conduct of each major stage of the commissioning programme;
- The measures to be established for the review, evaluation, and approval of commissioning results for each major stage of the programme;
- Baseline data for equipment and systems for future reference;
- The applicant's requirements pertaining to the disposition of commissioning procedures and test data following completion of the commissioning programme;
- The list of regulatory guides applicable to initial commissioning programmes that will be used or alternative methods along with justification for their use;
- Information on the programme for utilizing available information on plant operating experiences in the development of the initial commissioning programme, including identification of the participating organizations in the programme, and a summary description of their qualifications;
- Schedule for 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 use-tested during the initial commissioning programme;
- Description of the procedures that will guide initial fuel loading and initial criticality, including the safety and precautionary measures to be established for safe operation;
- The schedule, relative to the fuel loading date, for conducting each major stage of the commissioning programme;
- Abstracts for all commissioning tests that will be conducted during the initial commissioning programme, with emphasis on safety systems and safety features that (1) are relied on for the safe shutdown and cool down of the plant in operational states and accident conditions, (2) are relied on for establishing conformance with operational limits and conditions that will be established by the technical specifications, and (3) are relied on to prevent or mitigate the consequences of anticipated operational occurrences and accident conditions;
- Summary of the commissioning programs implemented in the main stages of the commissioning programme, including an assessment on the achievement of test objectives.

CHAPTER 15. SAFETY ANALYSIS

3.15.1. Chapter 15 should provide a description of the safety analyses performed to assess the safety of a plant 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 core melting accidents, and probabilistic safety assessment.

3.15.2. The description of the analyses and the associated assumptions provided in chapter 15 may be supported by reference material, where necessary. The level of detail provided in this chapter should increase as a nuclear power plant project develops from the siting stage through the construction stage up to the commissioning and operation stages.

3.15.3. Scope of information provided in chapter 15 should reflect the requirements on safety analysis relevant for nuclear power plant design; see SSR-2/1 (Rev. 1) [3], in particular requirements 16, 17, 19, 20 and 42, and GSR Part 4 (Rev. 1) [2], requirements 14 to 21. More specifically, guidance on deterministic safety analysis is provided in *DS491 [42]* and on probabilistic safety assessment in SSG-3 [43] and SSG-4 [44].

3.15.4. The information provided in this chapter should be sufficient to justify and confirm the design basis for the items important to safety, and to ensure that the overall plant design is capable of meeting the established acceptance criteria, in particular the authorized limits for radiation doses and radioactive releases for each plant state. In addition, the level of detail should provide sufficient information to allow for independent verification of safety analyses, as required by GSR Part 4 (Rev. 1) [2], Requirement 21, when applicable.

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

General considerations

3.15.6. This section should provide an introduction to the chapter on safety analysis, covering both deterministic and probabilistic analyses. The scope of safety analysis and the approach adopted (e.g. conservative or realistic, as appropriate) should be described here, individually for each plant state or accident scenario from normal operation up to design extension conditions with core melting.

3.15.7. This section should also explain how previously identified generic issues and relevant operating experience have been utilized in enhancing quality of safety analysis, as required in several paragraphs of GSR Part 4 (Rev. 1) [2], including Requirement 19.

3.15.8. The approach should also include description of and how the loads due to internal or external hazards have been considered as initiators for postulated initiating events.

3.15.9. Any applicable reference documents on the methodology used in safety analysis should be introduced here. Due to the large complexity of the chapter it is also appropriate to describe in more detail the structure of the whole chapter in this section.

Identification and categorization of postulated initiating events and accident scenarios

3.15.10. The approach used to identify postulated initiating events and accident scenarios both for deterministic and probabilistic analyses should be described in this section. This may include, among other things, the use of analytical methods such as screening of defence-in-depth, master logic diagrams, hazard and operability analysis and failure mode and effects analysis, see *DS491 [42]*.

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 (effect on the plant). The purpose of this categorization is: (a) To justify the basis for the range of events under consideration; (b) To reduce the number of initiating events requiring detailed analysis to a set based on the most bounding cases in each of the various event groups credited in the safety analyses, in order to avoid events with very similar system performance (such as in terms of timing, plant systems response and radiological release fractions); (c) To allow for 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. Besides 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 through low power up to full power operation) should be covered, including potential events which could occur during commissioning and testing of the nuclear power plant. Since design extension conditions typically develop due to additional multiple failures, such multiple failures supposed 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) and all modes of normal operation (including operation at power or during shutdown and refuelling) that will be analysed, should be presented in this section.

3.15.15. Where appropriate, considered interactions between the electric grid and the plant, and interactions between different reactor units on the same site should be described in this section.

3.15.16. Considered failures initiated in other plant systems besides the reactor itself, such as the storage for fresh or irradiated fuel and storage tanks for radioactive gaseous or liquid wastes, should be also described here.

3.15.17. It should be also described how relevant internal and external hazards, both of natural as well as of human induced origin, leading to initiating events which may potentially challenge the safety functions, have been considered in determination of postulated initiating events.

3.15.18. This section should also describe how the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release were 'practically eliminated' due to measures taken to prevent the occurrence of such sequences and to their very low likelihood, with reference to specific analyses presented in this safety analysis report.

Safety objectives and acceptance criteria

3.15.19. This section should describe how specific safety analysis refers to the principles and objectives of nuclear safety and to general acceptance criteria introduced in Chapter 3 on general approaches to design of SSCs.

3.15.20. Both radiological acceptance criteria related to the radiological consequences and technical acceptance criteria related to the integrity of barriers should be specified in this section for different categories of events and types of analyses. Information on acceptance criteria given in this section should be consistent with more general information provided in chapter 3.

3.15.21. If probabilistic values such as core damage frequency or large releases frequency are set up as acceptance criteria or design objectives, these specific values should be also provided here.

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

Human actions

3.15.23. This section should describe the approaches adopted to take into account human actions and the methods selected to model these actions in both deterministic and probabilistic analyses; see GSR Part 4 (Rev. 1) [2], Requirement 11. Differences in approaches to consideration of human actions between deterministic and probabilistic analyses should be described.

3.15.24. 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 multi-unit plants.

Deterministic safety analyses

General description of the approach

3.15.25. In this subsection it should be described how sufficient margins in safety analysis have been ensured using acceptable approaches (i.e., conservative or realistic, as suggested in *DS491 [42]*), and how in the case of realistic analysis the uncertainties in the computer codes and other input data were taken into account.

3.15.26. The computer codes used for the deterministic analyses should be briefly described. The version of a computer code used should be clearly identified with reference to the relevant supporting documentation.

3.15.27. Emphasis should be given to the brief substantiation of the applicability of the computer code to the particular analysis. In particular, a summary of the scope of validation of the computer codes should be presented, with references to more detailed topical reports.

3.15.28. The plant models 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. The key validations of the plant model should also be summarized. Sufficient plant data used for development of the plant models should be provided in order to allow for independent verification of safety analysis, if applicable; see GSR Part 4 (Rev. 1) [2].

3.15.29. Main simplifications made in development of plant models should be described and justified. The set of assumptions for safety analysis used in the deterministic safety analyses performed for the different types of scenarios should be described in this section.

3.15.30. Any additional guidelines (such as on the choice of operating states of systems and/or support systems, conservative time delays and operator actions) for development of the plant models should be described or referred to.

Analysis of normal operation

3.15.31. This section should demonstrate that the normal operation can be carried out safely and hence should confirm that

- Radiation doses to members of the public corresponding to the planned discharges and/or releases of radioactive material from the plant are within the authorized limits;
- Plant parameters in normal operational regimes are maintained within the boundaries specified by the relevant Operational Limits and Conditions (OLCs), and that a reactor trip or initiation of the limiting and safety systems would be avoided.

3.15.32. 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, handling of irradiated fuel, and off-loading of irradiated fuel from the reactor to the spent fuel pool.

Analysis of anticipated operational occurrences and design basis accidents

3.15.33. 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. Sufficient information, confirming the adequacy of the design of nuclear power plant systems or components as well as of the envisaged operator actions by demonstrating compliance with the associated acceptance criteria, should be provided.

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

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

3.15.36. 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.

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

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

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

3.15.40. 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. Selection of a bounding case with its justification should be described;

(b) Tools and methodology: Computer codes and models used for the analysis;

(c) Plant parameters: Specific values of important plant parameters and initial conditions used in the analysis, with indication of reference (nominal) values and uncertainties of 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 the case when the approach with quantification of uncertainties was selected, the ranges and probability distribution of parameters should be specified and justified;

(d) Availability of systems and operator actions: A detailed description of the plant operating configuration prior to the occurrence of the postulated initiating event should be provided. This description should include information on availability of systems (including consideration of the worst single failure in safety systems) and operator actions that are credited in the analysis. Assumptions on 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; see *DS491 [42]*;

(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 should be provided;

(f) Plant response assessment: A discussion of the modelled plant behaviour, highlighting the timing of the main events (initial event, any subsequent failures, times at which various safety groups are actuated and 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 criterion 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 the physical barriers and the fulfilment of the safety functions should also be discussed;

(g) Assessment of radiological consequences: The results of the assessment of radiological consequences, if applicable for a given event, should be presented. The key results should be compared with the radiological acceptance criteria. 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 having radiological consequences, with appropriate selection of bounding cases for different categories of events;

(h) Sensitivity studies and uncertainty analyses: The sensitivity studies and uncertainty analyses, whenever needed in accordance with *DS491 [42]*, should be performed and presented to demonstrate the robustness of the results and to support conclusions of the accident analyses.

3.15.41. In order to support presentation of independence between levels of defence and robustness in anticipated operational occurrences in particular, it is recommended to include into safety analysis report also the realistic analysis of certain anticipated operational occurrences, with the main objective to demonstrate that the plant operational 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 occurrences. Detailed guidance for performing conservative and realistic analysis of anticipated operational occurrences is provided in *DS491 [42]*.

Analysis of design extension conditions without significant fuel degradation

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

3.15.43. Scope and components of the information provided should be similar as described above for design basis accidents, taking into account the main differences in approaches to safety analysis as described in *DS491 [42]*.

Analysis of design extension conditions with core melting

3.15.44. 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 materials to the containment. The analysis presented in this section should identify the most severe plant parameters resulting from the core melt sequences, and demonstrate that:

- The plant can be brought into a state where the containment functions can be maintained in the long term;
- The plant structures, systems and components (e.g., the containment design) are capable of preventing an early radioactive release or a large radioactive release, including containment by-pass;
- Control locations remain habitable to allow performance of required staff actions;
- Compliance with the acceptance criteria is achieved by features implemented in the design and not only by implementation of severe accident management guidelines.

3.15.45. Scope and components of the information provided for this category of design extension conditions should be similar as described above for design basis accidents, taking into account the main differences in approaches to safety analysis as described in *DS491 [42]*.

3.15.46. Rather than presenting large number of accident scenarios, the information provided should focus on significant phenomena maximizing main parameters determining potential loss of containment integrity and releases of radioactive materials to the environment.

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

3.15.47. This section should present the safety analysis performed for postulated initiating events specifically initiated in the spent fuel pool. Specific operating modes related to fuel handling (e.g. emergency core unloading) should be also considered. It should be demonstrated that the relevant acceptance criteria (usually more restrictive than in events initiated in the reactor coolant system) regarding maintaining subcriticality, heat removal, structural integrity, shielding and confinement of radioactive gases released from irradiated fuel in the spent fuel pool are complied with.

3.15.48. Scope and components of the information provided should be similar as described above for design basis accidents, taking into account differences in systems involved, large thermal inertia of the spent fuel pool, more stringent acceptance criteria, and specific pathways for releases of radioactive substances. The information presented should contribute to confirmation that accidents with significant fuel degradation in the pools are practically eliminated.

Analysis of radioactive releases from a subsystem or component

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

3.15.50. Scope and components of the information provided should be similar to those described above for design basis accidents, taking into account that the main focus of the analysis is on the dispersion of radioactive substances in the environment rather than on analysis of processes inside the nuclear power plant.

Analysis of internal and external hazards

3.15.51. Analysis of all relevant site specific internal and external hazards should be presented in this section for hazards specified in chapter 3.

3.15.52. The analysis of hazards should show (if not already covered in other chapters) that such a hazard can be screened out due to its negligible likelihood, or that the nuclear power plant design is robust enough to prevent any transition from 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 analysis of postulated initiating events.

3.15.53. The analyses should be conveniently subdivided into hazards initiated inside the nuclear power plant (internal hazards), external hazards caused by natural events, and external hazards initiated by human activities using appropriate engineering tools for each kind of the hazard.

3.15.54. 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 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 SSR-2/1 (Rev. 1), para 5.21A [3].

Probabilistic safety analyses

3.15.55. 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 consideration of events in the spent fuel pool and hazards, as applicable. The complete probabilistic safety assessment study itself should be made available for review as a separate report to the regulatory body, if required.

General approach to probabilistic safety analysis

3.15.56. This section should describe the scope of the probabilistic safety assessment performed with justification of the selected scope. The methodology and computer codes used should be characterized. Sources of important input data should be introduced with 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 here.

3.15.57. The basic data used for the assessment should be provided, including the assessment of the frequency of initiating events, component reliability, common cause failure probabilities and human error probabilities.

Results of probabilistic safety assessment Level 1

3.15.58. The methods used and results of probabilistic safety assessment Level 1 should be summarized in this section. The results should include the results of accident sequence modelling, including event sequence and system modelling human performance analysis, dependence analysis and classification of accident sequences into plant damage states.

3.15.59. Quantification of accident sequences should be provided. The results of probabilistic safety assessment Level 1 study should include a delineation of the likely frequency of core damage from events which occur when the plant is operating at power as well as when it is shutdown, considering in detail the occurrence of events both internal and external to the plant.

Results of probabilistic safety assessment Level 2

3.15.60. The methods used and results of probabilistic safety assessment Level 2 should be summarized in this section, focused on the expected magnitude (source term) and frequency of radioactive material release to the environment as a consequence of core melting.

3.15.61. Results of the plant damage state analysis providing a structured interface between the Level 1 and Level 2 probabilistic safety assessment should be presented. Use of plant damage state as the input to the containment behaviour analysis performed by the containment event tree model should be described.

3.15.62. Main results of the containment performance analyses, i.e. containment event trees evaluations, and the source term evaluations should be summarized in this section.

Probabilistic safety assessment insights and applications

3.15.63. The summary results of the probabilistic analyses should be described in this part of the safety analysis report. Assessment of compliance with the 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. Intended use of probabilistic safety assessment to support future plant operation should be introduced.

Summary of results of the safety analyses

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

3.15.65. It should be confirmed in this section that the requirements of the analyses have been met in every respect, providing justification if requirements have been changed, or have been changed as a result of further considerations. In the latter case any compensatory measures taken to meet the safety requirements should be specified.

CHAPTER 16. OPERATIONAL LIMITS AND CONDITIONS

3.16.1. Chapter 16 should describe plant operational limits and conditions (OLCs) and should demonstrate that they will ensure compliance with SSR-2/1 (Rev. 1), Requirement 6 [3], and SSR-2/2 (Rev. 1), requirements 25 (para 6.14) and 28 (para 7.10) [4] and that they include all required components in accordance with SSR-2/1 (Rev. 1), Requirement 28 (para 5.44).

3.16.2. Chapter 16 should also document, in accordance with SSR-2/2 (Rev. 1) [4], requirements 6 (paras 4.6 to 4.16) and 25 (para 6.4), that the OLCs are consistent with the design and with relevant safety analysis, that proper measures are taken to ensure operation in compliance with OLCs, that the staff is 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. 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 generally presented in the form of safety limits, limiting setting for safety systems, limits and conditions for normal operation; surveillance and testing requirements and action statements for deviations from normal operation that are formally derived from the limiting plant configuration, possible plant states and acceptable range of operating parameters verified in relevant chapters of the safety analysis report, in particular chapter 15. This is to ensure that the operation of the plant will be at all times within the safe operating regime established for the plant. The OLCs should provide clear and unambiguous instructions to the operating organization staff that are clearly linked to the safety justification 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 be substantiated by means of a written indication of the reason for its adoption and 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, once in operation.

Safety limits

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

Limits and conditions for normal operation, surveillance and testing requirements

3.16.7. The corresponding 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 taking into account insights from a probabilistic safety assessment. The actions to be taken in the event that operational limits and conditions are not fulfilled should also be clearly established.

Administrative requirements

3.16.8. In some cases, essential administrative aspects, such as the minimum shift composition and the frequency of internal reviews, may be also covered by the operational limits and conditions. Reporting requirements for operational events should also be covered. The relevant administrative requirements should be described in this section.

CHAPTER 17. MANAGEMENT SYSTEMS

3.17.1. Chapter 17 should describe the overall management of all safety related activities to ensure compliance with principle 3 of SF-1 [19]. The information provided should cover establishing, assessing, sustaining and continuously improving effective leadership and management of safety and should allow for verifying compliance with GSR Part 2 (“Leadership and Management for Safety”) [45].

3.17.2. Justification that responsibilities of the operating organization have been established according to the applicable requirements should be provided, covering management of safety in design, established according to requirements 1 to 3 from SSR-2/1 (Rev. 1) [3], and management of operational safety, established in accordance with requirements 5, 8 and 9 from SSR-2/2 (Rev. 1) [4]. Recommendations to meet these requirements are provided in GS-G-3.1 [46] and GS-G-3.5 [47].

3.17.3. Chapter 17 should describe how overall safety objectives are established, controlled, monitored and reviewed, giving safety the highest priority with adequate consideration of safety culture.

General considerations

3.17.4. This section should describe senior management of the operating organization responsible for establishing, applying, sustaining and continuously improving the management system to ensure safety.

3.17.5. This section should include accountability for the management system even where individuals are assigned the responsibility for the coordination, development, application and maintenance of the management system. The responsibility of the operating organization for establishing safety policy should also be included.

Goals, strategies, plans and objectives

3.17.6. In this section should be described that operating organization establishes goals, strategies, plans and objectives for the organization that are consistent with the organization’s safety policy.

Specific aspects

3.17.7. Site management and corporate structure and technical support organization of the operating organization should be described in this section. The description should provide justification on the way how effective management control of the design and operating organizations will be achieved so as to promote safety. The description should include also measures employed to ensure the implementation and observance of the management safety procedures.

Integration of the elements of the Management System

3.17.8. This section should describe how the management system integrates its elements, including those regarding safety, health, environment, security, quality, human-and-organizational-factor, societal and economic, so that safety is not compromised.

Management of processes and activities

3.17.9. This section should describe that processes and activities shall be developed and shall be effectively managed to achieve the organization’s goals without compromising safety.

Fostering a culture for safety

3.17.10. This section should describe how individuals in the operating organization, from senior managers downwards, foster a strong safety culture, in accordance with Requirement 12 from GSR Part 2 [45]. According to that, the information provided should describe how the management system and leadership for safety foster and sustain a strong safety culture.

3.17.11. This section should include how senior management plan to perform regularly assessments of leadership for safety and of safety culture in its own organization. The information provided should demonstrate that the necessary arrangements are adequate and are in place at the plant. These arrangements should be aimed at promoting good awareness of all aspects of safety and at regularly reviewing with staff the level of safety awareness achieved on the site.

3.17.12. This section should describe, in accordance with Requirement 14 from GSR Part 2 [45], how senior management plan to ensure that self-assessment of leadership for safety and of safety culture include assessment at all organizational levels and for all functions in the organization, and that such self-assessment makes use of recognized experts in the assessment of leadership and of safety culture.

CHAPTER18. HUMAN FACTORS ENGINEERING

3.18.1. Chapter 18 of the safety analysis report should describe how human factors engineering principles are incorporated into the human machine interface design in order to meet the Requirement 32 (paras 5.53 to 5.62) from SSR-2/1 (Rev. 1) [3]; further guidance is provided in *DS492 [48]*. The same applies to all operational modes and accident conditions and to all plant locations where such interactions are anticipated. In particular the following should be addressed:

(1) The planning and management of human factor engineering activities;

(2) The plant design process;

(3) The characteristics, features and functions of the human-machine interfaces, procedures, and training;

(4) The implementation of the human-machine interface design;

(5) Monitoring of performance at the site.

3.18.2. This chapter should provide information how human characteristics and capabilities were taken into account in the nuclear power plant design to support the reliability of the operator’s performance.

3.18.3. Although this chapter should cover the issues associated with the human factors comprehensively, such factors should be also considered in other chapters of the safety analysis report, including those relevant for management systems, siting, operation, safety analyses, radiation protection and decommissioning.

Human factor engineering programme management

3.18.4. This section should describe the human factor engineering programme, including the following topics:

- General human factor engineering programme management;
- Analysis of operating experience, functional requirements, analysis and function allocation, tasks analysis, staffing, organization and qualification, and treatment of important human actions;
- human-machine interface design;
- Human factors verification and validation;
- human factor engineering implementation;
- Human performance monitoring.

Task Analysis

Review of nuclear power plant operating experience

3.18.5. This section should describe the review of operating experience, how it was used to identify human factor engineering related safety issues and document that the human factor engineering related safety issues were identified and analysed.

3.18.6. In addition, this section should also describe a methodology used for the development and assessment, and to summarize the results of the assessment.

Functional requirements analysis and function allocation

3.18.7. This section should describe the functional requirements analysis and the scope of the analyses performed.

3.18.8. This section should include identification and analysis of those functions that must be performed to satisfy the plant's safety objectives; that are, to prevent accidents that could cause undue risk to the health and safety of the public, and to mitigate the consequences of such accidents if they were to occur.

Task analysis

3.18.9. This section should describe the objectives and scope of the task analysis including assumptions and bounding conditions, in order to analyse the context used to accomplish the task from the standpoint of its users (e.g. human-machine interface, procedures and organizational arrangements).

3.18.10. This section should describe whether specific tasks needed for accomplishment of a function in different locations (e.g. control room, supplementary control room, field and technical support centres) are identified for all plant states, for all modes of normal operation and considering all groups of operating personnel (including reactor operator, turbine operator, shift supervisor, field operator, safety engineer, and operation and maintenance staff).

3.18.11. Description of the scope should address how representative human important tasks (maintenance, test, inspection and surveillance) were selected, as well as the range of normal operation modes included in the analyses.

Staffing and qualifications

3.18.12. This section should describe the staffing and qualifications analyses, and the scope of the analyses performed. In coordination with Section 13.1, it should document that the requirements for the number and qualifications of personnel were analysed in a systematic manner, including a thorough understanding of task requirements and applicable regulatory requirements.

3.18.13. The scope should include the number and qualifications of personnel for the full range of plant conditions and tasks, including operational tasks (normal, abnormal, and emergency), and plant maintenance and testing (including surveillance testing).

3.18.14. In addition, any other plant personnel who perform tasks that directly relate to plant safety should be addressed.

Human reliability analysis

3.18.15. This section should describe the use of the human reliability analysis in the human factor engineering programme. This section should document how the human reliability analysis results were addressed in other activities of the human factor engineering programme such that important human tasks have been thoroughly addressed.

Human-machine interface design

3.18.16. This section should describe a structured methodology applied for human-machine interface design that permits 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 chapter should also describe the process by which human-machine interface design requirements are developed and human-machine interface designs are identified and refined.

Human-machine interface design inputs

3.18.18. This section should describe how the human factor engineering design process 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 human-machine interface provide the operating organization with the information necessary to detect changes in system status, to diagnose the situation, to affect 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 concepts 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) how human-machine interface design provide displays and controls in the main control room for manual, system level actuation of critical safety functions, and for monitoring those parameters that support them.

3.18.22. This section should also describe how the main control room human-machine interface design is giving due consideration to:

- Type of human-machine interface to be used according to its purpose;
- Organization of human-machine interfaces into workstations (e.g. consoles and panels);
- Arrangement of workstations and supporting equipment in the main control room.

Human-machine interface of Supplementary control room

3.18.23. This section should describe how human-machine interface design considers human factor engineering principles and human characteristics of personnel under emergency conditions, particularly for immediate actions.

3.18.24. This section should describe (consistently with chapter 7) how the human-machine interface design process for the supplementary control room and other emergency response facilities is performed to ensure the design process for the main control room, using similar procedures, criteria and methods.

3.18.25. This section should also describe the functions of the supplementary control room and other emergency response facilities required to be maintained in case of internal or external hazards for the control and monitoring of the critical functions and to conduct and ensure safe shutdown.

Procedure development

3.18.26. This section should document, in coordination with chapter 13, that the procedure development incorporates human factor engineering principles and criteria, along with other design requirements, to develop 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 procedure development programme. This section should address the following:

- Plant and system operations (including start-up, power, and shutdown operations);
- Test and maintenance;
- Abnormal and emergency operations;
- Alarm response;
- Generic technical guidelines for emergency operating procedures;

- Accident management guidelines.

Training programme development

3.18.28. This section should document, in coordination with chapter 13, a systematic approach for the development of personnel training.

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

- Categories of personnel to be trained, including the full range of positions of operational personnel;
- The full range of plant conditions (normal, upset, and emergency);
- Specific operational activities (e.g., operations, maintenance, testing and surveillance);
- The full range of plant functions and systems, including those that may be different from those in predecessor plants (e.g., passive systems and functions);
- The full range of relevant human-machine interfaces (e.g., main control room, remote shutdown panel, local control stations and technical support centre) including characteristics that may be different from those in predecessor plants (e.g., display space navigation, operation of "soft" controls).

Verification and validation of human factor engineering results

3.18.39. This section should document that verification of human-machine interface design was performed against task requirements that have been identified in task analysis.

3.18.31. This section should document whether the test scenarios used for validation testing allow for the assessment of the resources placed at the personnel's disposal over appropriate lengths of time and in an appropriate meaningful number of scenarios.

3.18.32. This section should document the criteria applied for the verification, including the selection of standards and guidelines (human factor engineering guideline) suitable for the review of characteristics of the human-machine interface components included in the scope of the evaluation.

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 factor engineering design process.

3.18.34. The scope should include the following considerations:

- Verification and validation of design aspects that cannot be completed as part of the human-machine interface verification and validation program;
- Confirmation that the as-built human-machine interface, procedures, and training conform to the approved design;
- Confirmation that all human factor engineering issues in the tracking system are appropriately addressed.

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

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

Human performance monitoring

3.18.38. This section should describe a human performance monitoring programme for determining that no significant safety degradation occurs because of any changes that are made in the plant and to confirm that the conclusions that have been drawn from the integrated system validation remain valid over time.

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

- The design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centres;
- 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 previously trained skills);
- Human actions can be accomplished within established time and performance criteria;
- The acceptable level of performance established during the integrated system validation is maintained.

CHAPTER 19. EMERGENCY PREPAREDNESS

3.19.1. This chapter should provide information on emergency preparedness, demonstrating in a reasonable manner that, in the event of an accident, all actions necessary for the protection of the public, workers and the plant could be taken, and that the decision making process for implementation of these actions would be timely, disciplined, co-ordinated and effective. The emergency preparedness arrangements should cover accident conditions (design basis accidents and design extension conditions) that would have adverse effects on the environment and the off-site areas where preparations for the implementation of off-site protective actions are warranted.

3.19.2. The description should include information on the objectives and strategies, organization and management, and should provide sufficient information to show how the practical goals of the emergency plan will be met; see GSR Part 7 ("Preparedness and Response for a Nuclear or Radiological Emergency" [49**].**

3.19.3. Liaison and co-ordination with the actions of other authorities and organizations involved in the response to an emergency should be described in detail. This should include a description of the procedures used to implement off-site protective actions for all jurisdictions where urgent off-site protective actions may be warranted in the event of a severe accident.

3.19.4. The provisions, including on-site and off-site exercises, to ensure that appropriate arrangements for emergency preparedness and response are in place before commissioning should be described. The intervals foreseen for regular drills and exercises to maintain adequate emergency preparedness should be established and justified.

3.19.5. Further discussion on matters to be covered in this chapter of the safety analysis report is provided in GSR Part 7 [49]; EPR-nuclear power plant Public Protective Actions ("Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor") [50]; GSG-2 ("Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency") [51]; GS-G-2.1 ("Arrangements for Preparedness for a Nuclear or Radiological Emergency") [52].

Emergency management

3.19.6. This section should contain an appropriate description of the operating organization's response to an emergency; see Section 5 of GSR Part 7 [49]. Description should be provided here of the emergency arrangements for the protection of workers and the public in the event of an accident, including measures for:

- Establishing emergency management;
- Identifying, classifying and declaring emergency conditions;
- Notifying off-site officials;
- Activating the response;
- Taking mitigatory actions;
- Taking early protective actions, urgent protective actions and other response actions on and off the site;
- Protecting emergency workers and helpers;
- Assessing the initial phase;
- Managing the medical response;
- Mitigating non-radiological consequences;
- Managing radioactive waste; and
- Keeping the public informed.

3.19.7. Measures for ensuring the protection of the plant staff and how these will be co-ordinated with other emergency response actions should be described. When necessary, reference to other sections of the safety analysis report where this issue is discussed should be made.

Emergency response facilities

3.19.8. Information should be provided about the particular availability at the plant, including resistance to external hazards and habitability conditions, of the following:

(a) An on-site emergency facility in which response personnel will decide on, initiate and manage all on-site measures, except for the detailed control of the plant, and for transmitting data on plant conditions to the off-site emergency facility;

(b) Appropriate measures to enable the control of essential safety systems from a supplementary control room;

(c) An off-site emergency facility in which response personnel will assess information gained from on-site measurements, provide advice and support to bring the plant under control and protect the staff, if necessary, and co-ordinate with all emergency response organizations in order to inform and, if necessary, protect the public;

(d) Off-site monitoring systems for passing data and information to the regulatory body if appropriate or if required by national arrangements.

3.19.9. Description of emergency response facilities should include details of any equipment, communications and other arrangements necessary to support the specific facilities' assigned functions. The habitability of these facilities and the provisions to protect workers during accidents should also be described and justified.

Capability of the operating organization for the assessment of the consequences of accidents

3.19.10. This section should provide a demonstration that the operating organization will have measures or arrangements in place for:

(a) The early detection, monitoring and assessment of conditions for which accidental operating procedures are warranted, to mitigate the consequences of an accident, to protect on-site personnel and to recommend appropriate protective actions to off-site officials. This assessment should include the assessment of actual or predicted levels of core damage;

(b) The prediction of the extent and significance of any release of radioactive material if an accident has occurred;

(c) The prompt and continuous assessment of the on-site and off-site radiological conditions;

(d) The continuous assessment of conditions at the plant to modify, as appropriate, ongoing response actions.

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

Emergency preparedness for multi-unit sites

3.19.12. If a new reactor is located on, or near, an operating reactor site with an existing emergency arrangements (i.e., multiunit site), and the emergency plan for the new reactor includes various elements of the existing one, this section should:

(1) Address the extent to which the existing site's emergency plan is credited for the new unit(s), including how the existing plan would be able to adequately accommodate an expansion to include one or more additional reactors and include any required modification of the existing emergency plan for staffing, training, emergency action levels, and the like, considering also potential simultaneous accidents on all reactors located at the site;

(2) Describe any required updates to existing emergency facilities and equipment, including the alert notification system, considering also potential occurrence of an emergency on several reactors at the same time;

(3) Incorporate any required changes to the existing on-site and off-site emergency response arrangements and capabilities with state and local authorities or private organizations;

(4) If applicable, address the exercise requirements for collocated licensees;

(5) Describe how emergency arrangements, including potential interface with security measures, are integrated and coordinated with emergency arrangements of adjacent sites.

CHAPTER 20. ENVIRONMENTAL ASPECTS

3.20.1. This chapter should provide a brief description of the approach taken to assess the impact on the environment of the plant operation for operational states as well as for accident conditions, including severe accidents. Only radiological environmental aspects should be included in this chapter of the safety analysis report.

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

General aspects of the environmental impact assessment

3.20.3. This section provides the introduction to the chapter. In particular, the interrelation of the environmental impacts assessment to the status of the project and the status of reviews, approvals, and consultations associated with the environmental impact assessment should be summarized.

Site characteristics important for the environmental impact

3.20.4. This section should briefly summarize all site characteristics addresses in chapter 2 of the safety analysis report which are important from environmental impact point of view, including land, water, ecology as well as, relevant data on the population distribution, geology, and meteorology.

3.20.5. The appropriate scope of relevant information on site specific factors can be found in NS-R-3 (Rev. 1) [11] ("Site Evaluation for Nuclear Installations") and NS-G-3.2 [13] ("Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants").

Plant features minimizing environmental impact

3.20.6. Plant characteristics determining radiological releases or minimizing the radiological environmental impact should be summarized here, with references made to other chapters of the safety analysis report as appropriate.

Environmental impacts of construction

3.20.7. The construction of the plant itself does not represent a source of radiation. However, other potential sources of radioactivity, such as adjacent nuclear installations or sealed radiation sources, used during the plant construction should be considered for determination of the quantitative radiological impact on construction workers at the site of the proposed plant. Assumptions, methodology and results of such radiological impact analysis should be described in this section.

Environmental impacts of normal operation

3.20.8. This section should demonstrate compliance with all operational targets for solid, liquid and gaseous discharges and adequacy of measures to comply with authorized limits. All radiation impacts on surroundings under plant operation should be considered, including:

- Direct ionizing radiation from the buildings and facilities in which radioactive materials are handled;
- Ionizing radiation emitted by radioactive nuclides in gaseous discharges from controlled area devices;
- Ionizing radiation emitted by radioactive nuclides in liquid discharges from controlled area devices.

3.20.9. Further on, the section should summarize the measures that will be taken to control radioactive discharges to the environment (consistently with chapters 11 and 12). External exposure from the plume of radioactive gases and aerosols released from the ventilation stacks, external exposure from radioactive fall-out (deposition) and internal exposure from inhalation and ingestion of radionuclides should be addressed.

3.20.10. Further information on methods and approaches for the assessment of radiological consequences of plant operation for the environment is provided in GSR Part 4 (Rev. 1) [2] ("Safety Assessment for Facilities and Activities"); *DS491 [42] ("Deterministic Safety Analysis for nuclear power plants", draft Safety Guide Step 8)*; *DS427 [13] ("Prospective radiological environmental impact assessment for facilities and activities"; draft Safety Guide step 11*).

Environmental impacts of postulated accidents involving radioactive materials

3.20.11. The environmental effects of accidents involving radioactive material that can be postulated for the plant under review should be addressed in this section. The list of accidents covered should be provided. The scope of the section should cover the off-site consequences in terms of projected effective doses for sufficient distance from the plant for design basis accidents as well as for selected design extension conditions with core melting (except those which are practically eliminated). The type of data and information needed will be affected by site- and station-specific factors, and the degree of detail should be modified according to the anticipated magnitude of the potential impacts. An overview of the off-site protective actions to limit adverse radiological impacts during accidents should be described.

Environmental impacts of plant decommissioning

3.20.12. Similarly as it was done for the plant normal operation, radiological impacts of the plant decommissioning should be summarized in this section, with the reference made to chapter 21.

3.20.13. Further information on issues associated with decommissioning can be found in GSR Part 6 [53] ("Decommissioning of Facilities"); *DS452 [54] ("Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities"; draft Safety Guide step 11)*; WS-G-5.2 [55] ("Safety Assessment for the Decommissioning of Facilities Using Radioactive Material").

Environmental measurements and monitoring programmes

3.20.14. This section should refer to the off-site monitoring regime for contamination levels and radiation levels consistently with Chapter 11. The dedicated environmental monitoring programmes and alarm systems should be described that are required to respond to unplanned radioactive releases and the automatic devices to interrupt such releases, if applicable. All routes, which could be the source of uncontrolled ionization radiation and radioactive substance leakage beyond the power plant systems, should be addressed. Warning signals or automatic blockades preventing the unauthorized regime, together with the activation levels settings, should be specified. Further information on issues associated with environmental monitoring can be found in RS-G-1.8 [56] ("Environmental and Source Monitoring for Purposes of Radiation Protection").

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 and retain records of radioactive releases that will routinely be made from the site. Further on, this section should describe the measures that will be taken to make appropriate data available to the authorities and the public. It should be demonstrated that the forms and deadlines of the records comply with relevant regulations and conditions given by the regulatory body in the operation licence.

CHAPTER 21. DECOMMISSIONING AND END OF LIFE ASPECTS

3.21.1. Chapter 21 should describe decommissioning as a stage in the lifetime of a plant, which comes after the permanent cessation of operation (permanent shutdown) and plant transition period. The feasibility of decommissioning and capability to decommission the plant should be demonstrated already during design and construction stages, before the initial criticality occurs or before plant operation commences. This demonstration is usually done in an initial decommissioning plan. If the initial decommissioning plan is part of the safety analysis report, a discussion of its content should be presented or reference be made to its contents in this chapter.

3.21.2. Already during nuclear power plant siting, it should be demonstrated how the plant design allow minimizing contamination during decommissioning. Additionally, it should be also described that during plant lifetime, appropriate radiological surveys are conducted including of the subsurface, site water storage and drainage systems and groundwater and records maintained of residual radioactivity. The associated safety issues should be described in this chapter.

3.21.3. During operation of the plant, the initial decommissioning plan should be periodically updated to allow for an increasing level of detail, introducing new information available from the plant operation, and reflecting regulatory, technical and other developments related to decommissioning. It should be noted that the level of detail in the initial decommissioning plan takes a sharp increase beginning 5-10 years prior to the expected end of operating lifetime, when detailed planning for decommissioning begins. Where applicable, cost estimates and financial provisions should also be provided. Decommissioning related considerations should be maintained in the initial decommissioning plan and its supporting documents, as required by GSR Part 6 [53] ("Decommissioning of Facilities"). Further information on decommissioning is provided in *DS452 [54] ("Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities"; draft Safety Guide step 11) and in* WS-G-5.2 [55] ("Safety Assessment for the Decommissioning of Facilities Using Radioactive Material").

General principles and regulations

3.21.4. In addition to general principles adopted for decommissioning, this section should provide information on the documentation required and regulations to be followed, which ensure that both the radiation exposures to workers and to the public, and the amount of radioactive waste and hazardous material generated, are properly managed and minimized.

Decommissioning strategy

3.21.5. This section should present the options identified and the method chosen for decommissioning. The main differences between the alternative approaches should be explained (e.g. minimization of the radiological consequences for personnel, the public and the environment and optimization of the technological, economic, social and other relevant indicators). Options and their effects on the timing of the decommissioning process should also be discussed.

Facilitating decommissioning during design and operation

3.21.6. This section of the safety analysis report should briefly discuss the proposed decommissioning concept, 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 release of radioactivity;

(c) Consideration of the types, volumes and activities of radioactive waste generated during operation and decommissioning;

(d) Identified options for decommissioning;

(e) Anticipated programmatic changes necessary to transition;

(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 needed at the decommissioning stage for the duration of the decommissioning project.

Decommissioning plan

3.21.7 This section should present a tentative programme of decommissioning actions, including a timescale, containing the following basic 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 management of radioactive waste;

(c) Planning, phasing or staging of the decommissioning process, including appropriate requirements for surveillance and updated safety analyses throughout the process. In multiple unit plants, phasing may create a new plant configuration where some units are safe-stored (mothballed) and others are operating, which could involve severing of shared services provided by shared safety and process systems;

(d) Identification of the systems, tools and equipment required during decommissioning, and organization of the decommissioning;

(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 (mothballed in preparation for decommissioning) of selected units in a multiple unit plants;

(g) The development of a programme for ensuring that services (heating, electricity and water supply) will be available to support the work;

(h) Estimation of types and volumes of wastes arising from decommissioning;

(i) The development of a programme for providing adequate facilities for the handling, processing, storage and transport of the radioactive waste arising during decommissioning;

(j) Providing for the physical protection, monitoring and surveillance of the unit during the decommissioning phases identified;

(k) 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 on the basis of the specified safety principles and safety objectives. The measures should be described that are adopted at the design and required in future operation with the following objectives: (a) to minimize the volume of radioactive structures, (b) to reduce 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, and (f) to ensure the collection of important data.

3.21.9. An estimate of the volume of low and intermediate level waste should be provided. Special attention should be paid to the following aspects:

(a) Sources of radioactive materials should be identified and assessed;

(b) Radioactive (airborne and liquid) discharges during the process should be in accordance with the ALARA principle and should be kept within authorized limits;

(c) The practicability of adherence to the concept of defence in depth against radiological hazards during the decommissioning process should also be demonstrated.

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 identification of potentially reusable or recyclable materials arising from decommissioning.

APPENDIX I

DEVELOPMENT OF THE SAFETY ANALYSIS REPORT IN THE COURSE OF THE LICENSING STAGES

	Chapter of Safety Analysis Report	Project phases		
		Site Permit Initial SAR[10]	Construction Permit Preliminary SAR	Commissioning Pre-operational SAR (Final SAR)
1	**Introduction and General Description of the Plant**	Preliminary information	Final information	Verified/updated information
2	**Site Characteristics**	Final information	Verified information	Verified/updated information
3	**Safety Objectives and Design Rules of Structures, Systems, and Components**	General design requirements	Reactor type specific design requirements	Verified/updated information
4	**Reactor**	Description of an envelope and general requirements on a given part of the design or SSC	Description of SSC and requirements on operation of systems	Verified/updated information
5	**Reactor Coolant and Associated Systems**	Description of an envelope and general requirements on a given part of the design or SSC	Description of SSC and requirements on operation of systems	Verified/updated information
6	**Engineered Safety Features**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
7	**Instrumentation and Control**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
8	**Electric Power**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
9	**Auxiliary Systems and Civil Structures**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
10	**Steam and Power**	General	Description of	Verified/updated

[10] SAR: Safety Analysis Report

	Chapter of Safety Analysis Report	Project phases		
		Site Permit Initial SAR[10]	Construction Permit Preliminary SAR	Commissioning Pre-operational SAR (Final SAR)
	Conversion System	requirements on the design of SSC	SSC and requirements on operation of systems	information
11	**Radioactive Waste Management**	General requirements on the design of SSC	Description of source terms, SSC and requirements on operation of systems	Verified/updated information
12	**Radiation Protection**	General requirements on radiation protection	Demonstration of compliance with the requirements	Verified/updated information
13	**Conduct of Operations**	General requirements on conduct of operations	Demonstration of compliance with the requirements	Verified/updated information
14	**Plant Construction and Commissioning**	General requirements on commissioning	Demonstration of compliance with the requirements	Demonstration of compliance with the requirements
15	**Safety Analysis**	General requirements on scope, methods and criteria for safety analysis	Demonstration of compliance with the requirements	Verified/updated demonstration of compliance with the requirements
16	**Operational Limits and Conditions**	General requirements on operational limits and conditions	Description and specification of operational limits and conditions	Verified/updated description and specification of operational limits and conditions
17	**Management Systems**	General requirements on management system	Description of management system	Updated description of management system
18	**Human Factors Engineering**	General requirements on human factor engineering	Description of scope, methodology and results of human factor engineering	Updated description of human factor engineering
19	**Emergency Preparedness**	General requirements on emergency preparedness	Description of emergency facilities and emergency plans	Updated description of emergency facilities and emergency plans
20.	**Environmental Aspects**	Preliminary or expected information, consistent with EIA document	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 on	Preliminary information on	Updated information on

	Chapter of Safety Analysis Report	Project phases		
		Site Permit Initial SAR[10]	Construction Permit Preliminary SAR	Commissioning Pre-operational SAR (Final SAR)
		decommissioning and end of life aspects	decommissioning and end of life aspects	decommissioning and end of life aspects

APPENDIX II

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

II.1 A common structure with basic specification of the content proposed for sections dealing with structures, systems and components (in particular systems) is given below. When a topic is not relevant to a SSC, it is suggested to keep the section and to note in the content guidance that “No description is necessary.”

Structure, system and component or equipment functions

II.2 The safety and non-safety functions of the SSC or equipment should be described here.

Design basis

II.3 This section should include the safety design criteria, rules and regulations applying to the SSC, such as:

- List of plant operational conditions and postulated initiating events when the SSC is in operation or will be called upon;
- Safety requirements related to operating conditions, including stresses and environmental conditions (e.g. temperature, humidity, pressure, vibration and irradiation);
- Safety classification;
- Protection against external hazards;
- Protection against internal hazards;
- Seismic categorization;
- Single failure criterion and protection against common cause failures;
- Isolation considerations;
- Equipment qualification;
- Design standards, requirements and fabrication, construction and operational codes and other more specific design aspects such as:
    - Overpressure protection;
    - Thermal shock;
    - Leakage detection or collection.

Structure, system and component or equipment description

II.4 In this section, the SSC should be described. The description includes list and numbering of components, basic drawings of the components and the layout. Main design parameters should be provided, such as number of components, dimensions, operational capacity, location, operational parameters and power supply. The nature and the importance of topics can be different for structures, mechanical, electrical or instrumentation and control systems or components.

Materials

II.5 In this section, adequate and sufficient information should be provided regarding the materials used in components, as well as the material interactions with fluids that could potentially impair operation of engineered safety feature systems. The intent of the information included in this section of the SAR is to ensure compatibility of the materials with the specific fluids to which the materials are subjected. Their specific properties, quality and chemistry requirements are described in this section.

Interfaces with other equipment or systems

II.6 The support systems (e.g., those providing electric power, lubrication, ventilation and cooling water), supported systems and other connected systems should be described as well as the corresponding design requirements. Flow diagrams of pipelines and block-diagrams of instrumentation and controls, and locations of units and mechanisms including valves, pipelines, vessels, instrumentation and control and actuators should be presented. The boundaries with other systems should be shown.

System, component or eEquipment operation

II.7 This section should summarize the operation of the system or equipment.

Instrumentation and control

II.8 This section should describe the method of control, the alarms, indications and interlocks associated with operation of the SSC.

Monitoring, inspection, testing and maintenance

II.9 This section should present the monitoring, inspection, testing and maintenance which will help demonstrate that:

- The status of the equipment/system is in accordance with the design intent;
- There is adequate assurance that the equipment/system is available to operate as required;
- There has been no significant deterioration in equipment/system availability, performance and integrity since the last test.

Radiological aspects

II.10 This section should describe the measures taken to ensure that the dose rates to operating personnel, arising from the equipment/system operation or maintenance, are as low as reasonably achievable in operational states and in accident or post-accident conditions.

Performance and safety assessment

II.11 This section should present the measures taken to address each of the safety design aspects or requirements listed in the above section 2. This may include description of the method and results of the analysis demonstrating required capability of the equipment.

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ANNEX

TYPICAL TABLE OF CONTENT OF A SAFETY ANALYSIS REPORT

1 Introduction and General Description of the Plant

1.1 Introduction
1.2 Project implementation
1.3 Identification of interested parties
1.4 Information on the layout and other aspects
1.3 General plant description
1.4 Comparison with other facilities
1.5 Additional information concerning new safety features
1.6 Modes of normal operation of the plant
1.7 Principles of safety management
1.8 Additional documents considered as a part of the safety analysis report
1.9 Drawings and Other Detailed Information
1.10 Conformance with applicable regulations, codes and standards

2 Site Characteristics

2.1 Geography and demography
2.2 Evaluation of site specific hazards
2.3 Proximity of industrial, transportation and other facilities
2.4 Activities at the plant site that may influence the plant's safety
2.5 Hydrology
2.6 Meteorology
2.7 Geology, seismology and geotechnical engineering
2.8 Radiological conditions due to external sources
2.9 Site related issues in emergency arrangements and accident management
2.10 Monitoring of site related parameters

3 Safety Objectives and Design Rules for Structures, Systems and Components

3.1General safety design basis
  3.1.1 Defence in depth
  3.1.2 Safety functions
  3.1.3 General design basis and plant states
  3.1.4 Radiation protection and radiological acceptance criteria
  3.1.5 Deterministic and probabilistic design principles and criteria
3.2 Classification, load combinations, and allowable stresses
  3.2.1 Classification of structures, systems, and components
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13.3.1 Maintenance, surveillance, inspection and testing
13.3.2 Core management and fuel handling
13.3.3 Management of ageing
13.3.4 Control of modifications
13.3.5 Programme for the feedback of operational experience
13.3.6 Documents and records
13.3.7 Outages
13.4 Plant Procedures
13.4.1 Administrative Procedures
13.4.2 Operating Procedures
13.4.3 Emergency Operating Procedures
13.4.4 Accident Management Guidelines
13.5 Nuclear security

14 Plant Construction and Commissioning

14.1 Specific Information to be included in the safety analysis report prior to construction
14.1.1 Initial test programme and discussion of the overall test objectives and general prerequisites
14.1.2 Each unique or first-of-a-kind design feature
14.1.3 Plans to follow guidance
14.1.4 Plans for the utilization of available information
14.1.5 Overall schedule
14.1.6 Trial use of plant operating and emergency procedures
14.1.7 General plans
14.2 Specific Information to be Included in safety analysis report prior to commissioning
14.2.1 Commissioning programme, pre-operational and start-up testing programmes, and the specific objectives
14.2.2 System used to develop, review, and approve individual commissioning procedures, the organizational units or personnel
14.2.3 Administrative controls
14.2.4 Measures to be established for the review, evaluation, and approval of commissioning results
14.2.5 Disposition of commissioning procedures and test data
14.2.6 list all regulatory guides applicable and alternative method
14.2.7 Information on the programme for utilizing available information
14.2.8 Schedule of commissioning program including fuel loading date
14.2.9 Description of the procedures
14.2.10 Abstracts for all commissioning tests
14.2.11 Summary results of the commissioning programs

15 Safety Analysis

- 15.1 General considerations
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  - 15.1.2 Scope of safety analysis and approach adopted
  - 15.1.3 Analysis of design basis conditions
  - 15.1.4 Analysis of design extension conditions
  - 15.1.5 Analysis of the hazards
  - 15.1.6 Applicable reference documents
  - 15.1.7 Structure of chapter 15
- 15.2 Safety objectives and acceptance criteria
  - 15.2.1 Safety objectives and safety analysis
  - 15.2.2 Identification and classification of postulated initiating events and accident scenarios
    - 15.2.2.1 Basis for categorization of postulated initiating events and accident scenarios
    - 15.2.2.2 Categorization of events according their frequencies
    - 15.2.2.3 Grouping of events according their type
    - 15.2.2.4 List of postulated initiating events and accident scenarios
    - 15.2.2.5 List of internal and external hazards
  - 15.2.3 Deterministic acceptance criteria
    - 15.2.3.1 Acceptance criteria for analysis of core cooling and system pressure
    - 15.2.3.2 Acceptance criteria for analysis of radiological effects of design basis conditions and design extension conditions
    - 15.2.3.3 Acceptance criteria for analysis of pressure-temperature transients in the containment
    - 15.2.3.4 Acceptance criteria for pressurized thermal shocks
    - 15.2.3.5 Acceptance criteria for analysis of primary to secondary system leakages
    - 15.2.3.6 Acceptance criteria for hazards
  - 15.2.4 Probabilistic acceptance criteria
- 15.3 xxx
- 15.4 Human actions
  - 15.4.1 General considerations
  - 15.4.2 Human actions in deterministic safety analysis
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- 15.5 Deterministic analyses
  - 15.5.1 General description of the approach
    - 15.5.1.1 Safety margins in safety analysis
    - 15.5.1.2 Description of the computer codes used
    - 15.5.1.3 Description of the mathematical models
    - 15.5.1.4 Input data for the deterministic safety analysis
  - 15.5.2 Analysis of normal operation
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    - 15.5.2.2 Method and scope of analysis
    - 15.5.2.3 Results of analysis
    - 15.5.2.4 Conclusions
  - 15.5.3 Analysis of anticipated operational occurrences and design basis accidents
    - 15.5.3.1 Analysis of core cooling and system pressure for reactivity induced accidents
    - 15.5.3.2 Analysis of core cooling and system pressure for a decrease of reactor coolant flow
    - 15.5.3.3 Analysis of system pressure for increase of reactor coolant inventory
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15.5.8 Analysis of radioactive releases from a subsystem or a component
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- 18.2.3 Results

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18.4 Task Analysis
- 18.4.1 Objectives and Scope
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18.5 Staffing And Qualifications
- 18.5.1 Objectives and Scope
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- 18.10.1 Objectives and Scope
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