This publication has been superseded by SSG-61.

IAEA SAFETY STANDARDS SERIES

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

SAFETY GUIDE

No. GS-G-4.1

IAEA
International Atomic Energy Agency

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish standards of safety for protection against ionizing radiation and to provide for the application of these standards to peaceful nuclear activities.

The regulatory related publications by means of which the IAEA establishes safety standards and measures are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety, and also general safety (that is, of relevance in two or more of the four areas), and the categories within it are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides.**

Safety Fundamentals (blue lettering) present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.

Safety Requirements (red lettering) establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.

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Information on the IAEA's safety standards programme (including editions in languages other than English) is available at the IAEA Internet site

www-ns.iaea.org/standards/

or on request to the Safety Co-ordination Section, IAEA, P.O. Box 100, A-1400 Vienna, Austria.

OTHER SAFETY RELATED PUBLICATIONS

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

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Other IAEA series that include safety related publications are the **Technical Reports Series,** the **Radiological Assessment Reports Series,** the **INSAG Series, the TECDOC Series**, the **Provisional Safety Standards Series**, the **Training Course Series,** the **IAEA Services Series** and the **Computer Manual Series**, and **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**. The IAEA also issues reports on radiological accidents and other special publications.

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

The following States are Members of the International Atomic Energy Agency:

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

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SAFETY GUIDE

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FOREWORD

by Mohamed ElBaradei
Director General

One of the statutory functions of the IAEA is to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes, and to provide for the application of these standards to its own operations as well as to assisted operations and, at the request of the parties, to operations under any bilateral or multilateral arrangement, or, at the request of a State, to any of that State's activities in the field of nuclear energy.

The following bodies oversee the development of safety standards: the Commission on Safety Standards (CSS); the Nuclear Safety Standards Committee (NUSSC); the Radiation Safety Standards Committee (RASSC); the Transport Safety Standards Committee (TRANSSC); and the Waste Safety Standards Committee (WASSC). Member States are widely represented on these committees.

In order to ensure the broadest international consensus, safety standards are also submitted to all Member States for comment before approval by the IAEA Board of Governors (for Safety Fundamentals and Safety Requirements) or, on behalf of the Director General, by the Publications Committee (for Safety Guides).

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA for its assistance in connection with the siting, design, construction, commissioning, operation or decommissioning of a nuclear facility or any other activities will be required to follow those parts of the safety standards that pertain to the activities to be covered by the agreement. However, it should be recalled that the final decisions and legal responsibilities in any licensing procedures rest with the States.

Although the safety standards establish an essential basis for safety, the incorporation of more detailed requirements, in accordance with national practice, may also be necessary. Moreover, there will generally be special aspects that need to be assessed on a case by case basis.

The physical protection of fissile and radioactive materials and of nuclear power plants as a whole is mentioned where appropriate but is not treated in detail; obligations of States in this respect should be addressed on the basis of the relevant instruments and publications developed under the auspices of the IAEA. Non-radiological aspects of industrial safety and environmental protection are also not explicitly considered; it is recognized that States should fulfil their international undertakings and obligations in relation to these.

The requirements and recommendations set forth in the IAEA safety standards might not be fully satisfied by some facilities built to earlier standards. Decisions on the way in which the safety standards are applied to such facilities will be taken by individual States.

The attention of States is drawn to the fact that the safety standards of the IAEA, while not legally binding, are developed with the aim of ensuring that the peaceful uses of nuclear energy and of radioactive materials are undertaken in a manner that enables States to meet their obligations under generally accepted principles of international law and rules such as those relating to environmental protection. According to one such general principle, the territory of a State must not be used in such a way as to cause damage in another State. States thus have an obligation of diligence and standard of care.

Civil nuclear activities conducted within the jurisdiction of States are, as any other activities, subject to obligations to which States may subscribe under international conventions, in addition to generally accepted principles of international law. States are expected to adopt within their national legal systems such legislation (including regulations) and other standards and measures as may be necessary to fulfil all of their international obligations effectively.

EDITORIAL NOTE

An appendix, when included, is considered to form an integral part of the standard and to have the same status as the main text. Annexes, footnotes and bibliographies, if included, are used to provide additional information or practical examples that might be helpful to the user.

The safety standards use the form 'shall' in making statements about requirements, responsibilities and obligations. Use of the form 'should' denotes recommendations of a desired option.

The English version of the text is the authoritative version.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. In order for an operating organization to obtain regulatory approval to build and operate a nuclear power plant, an authorization is required to be requested from and granted by the regulatory body. Paragraphs 5.3 and 5.4 of the Safety Requirements publication GS-R-1, Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety [1], state that: "5.3. Prior to the granting of an authorization, the applicant shall be required to submit a detailed demonstration of safety, which shall be reviewed and assessed by the regulatory body in accordance with clearly defined procedures."; and that "5.4. The regulatory body shall issue guidance on the format and content of documents to be submitted by the operator in support of applications for authorization. The operator shall be required to submit or make available to the regulatory body, in accordance with agreed timescales, all information that is specified or requested." This information should be presented in the form of a report, hereinafter referred to as a safety analysis report (SAR).

1.2. The requirements for SARs depend strongly on the type of regulatory regime adopted by a State, which may affect the scope and depth of the information presented in the SAR. For States with small nuclear power programmes or States that import nuclear power plants, there may be significant reliance on the vendor State's practices or on international work that helps to demonstrate the safety of the design. In any event, there should be a dialogue between the regulator and the operator at an early stage, possibly at the siting stage, to agree on what is necessary to demonstrate the required level of safety of any proposed installation and the programme of submissions. Some States give very comprehensive guidance on the contents of a SAR; a widely quoted and used document is the Standard Format produced by the United States Nuclear Regulatory Commission (NRC) [2]. The NRC document and other relevant references are considered in this Safety Guide, which draws heavily on IAEA safety standards [1, 3–6] and other publications to present one possible format and content option for a comprehensive SAR for any type of nuclear power plant. Alternative formats to that presented in this Safety Guide may be used; in such cases the recommendations presented here should be regarded as potential elements for use in such alternative formats.

OBJECTIVE

1.3. The objective of this Safety Guide is to provide recommendations and guidance on the possible format and content of a SAR in support of a request to the regulatory body for authorization to construct and/or operate a nuclear power plant. As such, this Safety Guide recommends how to meet the requirements established in the Safety Requirements publication GS-R-1 (Ref. [1], para. 5.4) and complements the related Safety Guide [3].

1.4. Guidance on the assessment and verification to be conducted by the design and operating organizations in preparing the SAR is provided in Safety Guide NS-G-1.2, Safety Assessment and Verification for Nuclear Power Plants [4]. Guidance on the review and assessment to be performed by the regulatory body in the authorization process is provided in another Safety Guide, GS-G-1.2, Review and Assessment of Nuclear Facilities by the Regulatory Body [7].

SCOPE

1.5. This Safety Guide is intended primarily for use with land based stationary thermal nuclear power plants but it may, in parts, have a wider applicability to other nuclear facilities. The particular contents of the SAR will depend on the specific type and design of the nuclear power plant proposed, and this will determine how sections as in this Safety Guide are included in the SAR. Although intended mainly for use with new plants, the guidance presented here would also be useful for existing nuclear power plants when operating organizations periodically review their existing SARs to identify any areas in which improvements may be appropriate and/or to review the licensing basis. This Safety Guide covers at the same level of importance both technical aspects and human factors that should be addressed adequately in a SAR in order to substantiate plant safety.

STRUCTURE

1.6. Section 2 covers general considerations. Section 3 sets out the standard content of a SAR for a nuclear power plant. Section 4 provides recommendations for the review and updating of a SAR.

2. GENERAL CONSIDERATIONS

2.1. The SAR represents an important communication between the operating organization and the regulatory body, and it forms an important part of the basis for licensing a nuclear power plant and an important part of the basis for the safe operation of a plant. The SAR should therefore contain accurate and sufficiently precise information on the plant and its operating conditions and should typically include information on, for example, safety requirements, the design basis, site and plant characteristics, operational limits and conditions, and safety analyses in such a way that the regulatory body will be able to evaluate independently the safety of the plant. In particular, it should be demonstrated that the interdependence between the safety aspects of technical factors and human factors have been considered throughout the report. The SAR should present sufficient information on the plant that, for the purposes of nuclear and radiation safety assessment, the amount of additional documentation necessary to allow the authorization process to proceed is minimized. The SAR may refer to more detailed supplementary information, which should be made available to the regulatory body, if requested.

2.2. It is common practice in many States that SARs are issued in successive and complementary parts, which may include:

(a) An initial (preliminary) SAR or pre-construction SAR (PCSAR) that supports the application for authorization for siting and/or construction.
(b) An updated (intermediate) SAR or pre-operation SAR (POSAR) that, in the licensing process, precedes an application for authorization to operate. In some States licensing arrangements envisage the issue of formal permission for the commissioning of a nuclear power plant. In such cases, this intermediate version of the SAR, amended as a result of the primary regulatory review of the PCSAR, should be submitted to the regulatory body in order to demonstrate the readiness of the operating organization to start trial testing operations of the plant before putting it into commercial operation.
(c) A finalized (final) SAR or station SAR (SSAR) that incorporates the revisions to the intermediate report prior to the plant entering first routine operation.

2.3. The regulatory body should be kept well informed about the process of site selection and the subsequent development of the selected site and plant. Parts of the SAR should be submitted to the regulatory body at an early stage and in accordance with an agreed timetable; this approach will permit a smoother review process and help prevent unnecessary delays.

2.4. The initial report (PCSAR) may be of limited scope. Informal contacts should be encouraged before the pre-construction review stage between those planning to build a reactor and the regulatory body to develop a common understanding of the nature of the project and the likely regulatory requirements. The PCSAR should include a statement of the safety principles adopted and the safety objectives set for the intended design. It should include a statement of the manner of conforming with the fundamental safety principles and a statement of how the safety objectives are being met. It should typically contain sufficiently detailed information, specifications and supporting calculations to enable those responsible for safety to assess whether the plant can be constructed and operated in a manner that is acceptably safe throughout its lifetime. The safety features incorporated into the design, together with the possible challenges to the plant that have been considered, should be described, with due regard to any site specific aspects. The amount of information to be provided in the preliminary report will depend on the extent to which the proposed reactor design is based on a generic type or a standard design for which the licensing process has been followed previously, including the production of a SAR.

2.5. The intermediate report (POSAR) should revise and provide more specific information on the topics outlined in the PCSAR and on any departure from or revisions to the safety provisions or the design intent as set out in the preliminary report. The POSAR should essentially justify the finalized detailed design of the plant and presents a demonstration of its safety. In addition, the POSAR should deal in greater detail than the PCSAR with matters relating to the commissioning and operation of the plant during this phase of its lifetime. The POSAR should provide more up to date information on the licensing basis for the plant.

2.6. The final report (SSAR) incorporates any necessary revisions to the intermediate report (POSAR) following the commissioning and licensing process for the first entry into routine operation of the as-built nuclear power plant. The final report should clearly demonstrate that the plant meets its design intent. Systematic updating of the SAR would then become a requirement for the operating organization during the remaining lifetime of the plant. This would usually be done periodically so as to reflect any feedback of operating experience, plant modifications and improvements, new regulatory requirements or changes to the licensing basis.

2.7. The SAR is prepared by the operating organization for submission to the regulatory body to enable it to assess the suitability of the plant for licensing. The SAR should also serve as a basis for the operating organization to assess the safety implications of changes to the plant or to operating practices. The following sections of this Safety Guide set out a list and provide a description of possible topics for inclusion in a comprehensive SAR for a nuclear power plant. A standard format for the SAR is also discussed in the appropriate sections. In the application of this Safety Guide, adaptation may be necessary to reflect the differences between the phases of plant licensing and the licensing practices in different States. Where required by States, the SAR or any parts thereof may be made available to the public.

2.8. One of the main purposes of submitting the SAR is to provide the necessary information to the regulatory body. Plant staff and management should also have an understanding of the main findings of the SAR. This may be aided by providing supplementary documentation that summarizes the relevant sections of the SAR.

3. FORMAT AND CONTENT OF A SAR

CHAPTER I: INTRODUCTION

General considerations

3.1. The SAR should start with an introduction, which should include:

(a) A statement of the main purpose of the SAR;
(b) A description of the existing authorization status;
(c) An identification of the designer, vendor, constructor and operating organization of the nuclear power plant;
(d) A statement of any similar (or identical) plants that the regulatory body has already reviewed and approved and a statement of the specific differences and improvements that have been made since such an approval was granted;
(e) The main information on the preparation of the SAR;
(f) A description of the structure of the SAR, the objectives and scope of each of its sections and the intended connections between them.

CHAPTER II: GENERAL PLANT DESCRIPTION

General considerations

3.2. This chapter should provide a general description of the plant, including a consideration of current safety concepts and a general comparison with appropriate international practices. It should enable the reader to gain an adequate general understanding of the facility without having to refer to the subsequent chapters.

Applicable regulations, codes and standards

3.3. This section should provide a list of all relevant regulations, codes and standards that provide the general and specific design criteria that have been used in the design. If these regulations, codes and standards have not been prescribed by the regulatory body, a justification of their appropriateness should be provided. Any changes made to or deviations from the requirements for the design should be clearly stated, together with the way in which they have been addressed and justified.

3.4. Wherever systems or components do not comply in full with any of the requirements of the relevant regulations, codes and standards, a separate and complete justification of any relaxation of a specific requirement should be provided.

Basic technical characteristics

3.5. This section should present briefly (in a table, where appropriate) the principal elements of the installation, including the number of units at the plant, where appropriate, the type of plant, the principal characteristics of the plant, the primary protection system, the type of nuclear steam supply system or gas turbine cycle, the type of containment structure, 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. It may be useful to compare the plant design with similar earlier designs already approved by the regulatory body, so as to identify the main differences and assist in the justification of any modifications and improvements made. A list of selected plant characteristics should be included in an appendix to the SAR.

Information on the layout and other aspects

3.6. Basic technical and schematic drawings of the main plant systems and equipment should be included in this section, together with details of the physical and geographical location of the facility, connections with the electricity grid and means of access to the site by rail, road and water. The operating organization should provide general layout drawings for the entire plant. The illustrations 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 SAR that present detailed descriptions of specific systems and equipment.

3.7. The main interfaces and boundaries between on-site equipment and systems provided by different design organizations should be described, together with interfaces with equipment and systems external to the plant (including, for example, the electricity grid), with sufficient detail of the way in which plant operation is co-ordinated.

3.8. This section may, if required, also include or refer to confidential information on the provisions made for the physical protection of the plant. In some States this may also include coverage of the steps taken to provide protection in the event of malicious action on or off the site.

Operating modes of the nuclear power unit

3.9. All possible operating modes of the unit should be described, including startup, normal power operation, shutdown, refuelling and any other allowable modes of operation. The permissible periods of operation at different power levels in the event of a deviation from normal operating conditions should be described. The methods for restoring the unit to normal operating conditions should also be specified.

Material incorporated by reference

3.10. This section should provide a listing of the topical reports that are incorporated by reference as part of the SAR. Results of tests and analyses (e.g. results of manufacturers' material tests and qualification data) may be submitted as separate reports. In such cases, the reports should be listed in this section and referenced or summarized in the appropriate section of the SAR.

CHAPTER III: MANAGEMENT OF SAFETY

General considerations

3.11. This chapter should describe and evaluate the operating organization's management structure and the procedures and processes that have been put in place to achieve satisfactory control of all aspects of safety throughout the lifetime of the plant. This should include the roles of on-site safety assessment organizations and any off-site safety advisory committees that will advise the operating organization's management. The aim is to demonstrate that the operating organization will be able to fulfil its responsibility to operate the plant safely throughout its operating lifetime.

Specific aspects of management processes

3.12. The site and the corporate management structure and technical support organization of the operator should be described in this section. The way in which effective management control of the design and operating organizations will be achieved so as to promote safety, as well as the measures employed to ensure the implementation and observance of the management safety procedures, should be presented and justified. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [8].

Consideration of safety culture

3.13. This subsection should present the operating organization's strategy to encourage the development, maintenance and enhancement of a strong safety culture throughout the lifetime of the plant. 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. The operating organization should, where possible, determine indicators of safety culture and should develop a programme to monitor such indicators. The staff should be consulted on the determination of these indicators and should be kept informed of the outcome of the reviews. Action should be taken in response to any indications of declining safety levels.

Quality assurance

3.14. The principal aspects of the quality assurance (QA) system developed for the plant should be described in this subsection. It should be demonstrated that appropriate provisions for QA, including a QA programme, and audit, review and self-assessment functions, have been implemented for all safety related plant activities. These activities should include design, procurement of goods and services (including the use of contractors' organizations), plant construction and operation, maintenance, repair and replacement, in-service inspection, testing, refuelling, modification, commissioning and decommissioning. The QA arrangements should cover safety matters relating to the plant throughout its lifetime. Further discussion on matters to be covered in this subsection of the SAR is provided in Ref. [9].

Monitoring and review of safety performance

3.15. The information presented in this section should demonstrate that an adequate audit and review system has been established to provide assurance that the safety policy of the operating organization is being implemented effectively and that lessons are being learned from its own experience and from that of others, to enhance safety performance. It should be shown that means for independent safety review are in place and that an objective internal self-evaluation programme supported by periodic external reviews conducted by experienced industry peers is established. It should also be shown that relevant measurable indicators of safety performance are used to enable senior management to detect and respond in a timely way to any shortcomings and deterioration in safety.

3.16. This section should include a description of the way in which the operating organization intends to identify any development of the organization that could lead to the degradation of safety performance and should justify the appropriateness of the measures planned to prevent such a degradation. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [8].

CHAPTER IV: SITE EVALUATION

General considerations

3.17. This chapter should provide information on the geological, seismological, volcanic, hydrological and meteorological characteristics of the site and the surrounding region in conjunction with the present and projected population distribution and land use that is relevant to the safe design and operation of the plant. Sufficient data should be included to permit an independent evaluation.

3.18. Site characteristics that may affect the safety of the plant should be investigated and the results of the assessment should be presented. The SAR should provide information concerning the site evaluation task[1] as support for the design phase, design assessment phase [4] and periodic safety review [10], and may include:

(a) Site specific hazard evaluation for external events (of human or natural origin);
(b) Design targets in terms of recurrence probability of external events (see also paras 3.26 and 3.27);
(c) Definition of the design basis for external events;
(d) Collection of site reference data for the plant design (geotechnical, seismological, volcanic, hydrological and meteorological);
(e) Evaluation of the impact of the site related issues to be considered in the parts of the SAR on emergency preparedness and accident management;
(f) Arrangements for the monitoring of site related parameters throughout the lifetime of the plant.

3.19. 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 phase [11] should be provided in this section of the SAR.

3.20. Site related information represents a very 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 SAR.

[1] In some States such information is collected in an environmental report.

3.21. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [11].

Site reference data

3.22. 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 on the control of activities with the potential to affect plant operation, including flight exclusion zones. Information on such activities would include relevant data on the population distribution 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, fire fighting services and municipal services) around the plant site. This section should also cover the uses of the land and water resources in the surrounding area, for example for agriculture, and should include an assessment of any possible interaction with the plant.

3.23. Site reference data relating to geotechnical soil properties and groundwater hydrology should also be provided. The investigation campaigns for 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.

3.24. The SAR should present the relevant data for the site and the associated ranges of uncertainty to be used in the structural design and the dispersion studies for radioactive material. 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. The design of earth structures and site protection measures [11], 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.

Evaluation of site specific hazards

3.25. This section should present the results of a detailed evaluation of natural and human induced hazards at the site. Where administrative measures are employed to mitigate these hazards (especially for human induced events), information should be presented on their implementation, together with the roles and responsibilities for their enforcement.

3.26. 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 [11–16] should be discussed in the SAR.

3.27. The definition of the target probability levels for design against external events and their consistency with the acceptable limits should be discussed in this section of the SAR.

3.28. It should be demonstrated 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.

Proximity of industrial, transport and military facilities

3.29. This section should present the results of a detailed evaluation of the effects of potential incidents at present or proposed industrial, transport or other installations in the vicinity of the site. Any identified threats to the plant should be considered for inclusion in the design basis events to help determine any additional design features considered necessary to mitigate the effects of the potential incidents identified. A description of projected developments relating to this information should also be provided and should be updated as required.

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

3.30. Any processes or activities on the plant site that if incorrectly carried out might 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.

3.31. Measures for site protection (including dams, dykes and drainage) and any modifications to the site (such as soil substitution or modifications to the site elevation) are usually considered part of the site characterization stage and their assessment in relation to the design basis should be considered in this section of the SAR. This assessment may be made on the basis of guidance documents and Refs [11, 17, 18].

Hydrology

3.32. This section should present sufficient information to allow an evaluation of the potential implications of the hydrological conditions at the site for the plant design, performance requirements and safe operation. These conditions should include conditions relating to phenomena such as abnormally 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, ice related flooding and seismically generated water based effects on and off the site. For coastal and estuary sites, tsunamis, seiches and the combined effects of tides and strong wind should be evaluated. The information given in this section will be relevant to the assessment of the transport of radioactive material to and from the site and the dispersion of radionuclides to the environment. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [18].

Meteorology

3.33. 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. To this end, data deriving from on-site meteorological monitoring programmes should be documented. The extreme values of meteorological parameters, including temperature, humidity levels, rainfall levels, wind speeds for straight and rotational winds, and snow loads, should be evaluated in relation to the design. The potential for lightning and windborne debris to affect plant safety should be considered, where appropriate. The information given in this section will be relevant to the assessment of the transport of radioactive material to and from the site and the dispersion of radionuclides to the environment. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [19].

Seismology

3.34. This section should provide information concerning the seismic and tectonic characteristics of the site and of the region surrounding the site. The evaluation of seismic hazards should be based on a suitable geotectonic model substantiated by appropriate evidence and data. The results of this analysis, to be used further in other sections of the SAR in which structural design, seismic qualification of components and safety analysis are considered, should be described in detail. Further discussion on matters to be covered in this section of the SAR is provided in Refs [16, 17].

Radiological conditions due to external sources

3.35. The radiological conditions in the environment at the plant site, with account taken of the radiological effects of neighbouring plant units and other external sources, if any, should be described in sufficient detail to serve as an initial reference point and to permit the regulatory body to develop a view of the radiological conditions at the site.

3.36. A short description may 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 SAR concerned with the radiological aspects of licensing the plant.

Site related issues in emergency planning and accident management

3.37. Accident management relies strongly on the availability of adequate access and egress roads, sheltering and supply networks in the vicinity of the site. Many hazard scenarios for the site are expected to include effects in the vicinity of the site also, and thus the possibility of the evacuation of personnel and of access to the site. The availability of local transport networks and communications networks during and after an accident is a key issue for the implementation of a suitable emergency plan. The feasibility of emergency arrangements in terms of access to the plant and of transport in the event of a severe accident should be discussed in this section of the SAR. It should be shown that the requirements for adequate infrastructures external to the site are met. The needs for any necessary administrative measures should be identified, together with the relevant responsibilities of bodies other than the operating organization.

Monitoring of site related parameters

3.38. The provisions to monitor site related parameters affected by seismic, atmospheric, water and groundwater related, demographic, industrial and transport related developments 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.

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

3.40. 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 some detail in the SAR.

CHAPTER V: GENERAL DESIGN ASPECTS

General considerations

3.41. This chapter should outline the general design concept and the approach adopted to meet the fundamental safety objectives [20, 21]. The compliance of the actual design with the specific technical safety requirements should be demonstrated in more detail in other sections of the SAR, which may be referenced here.

Safety objectives and design principles

3.42. The safety objectives and principles used in the design should be presented in this section. These should be based on the objectives presented in paras 2.2, 2.4 and 2.5 of Ref. [5], which refers to the general nuclear safety objective, the radiation protection objective and the technical safety objective, as defined by the IAEA. This subject is discussed further in paras 3.43–3.52.

Defence in depth

3.43. This part of the SAR should describe in general terms the design 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 for all safety related activities. The design approach adopted should ensure that multiple and (to the extent possible) independent levels of and barriers for defence are present in the design to provide protection against operational occurrences and accidents regardless of their origin. The selection of the main barriers should be described and justified. Particular emphasis should be placed on systems important to safety. Where appropriate, any proposed operator actions to mitigate the consequences of events and to assist in the performance of important safety functions should be included. Guidance on the defence in depth concept is given in Ref. [5].

Safety functions

3.44. The Safety Requirements publication Safety of Nuclear Power Plants: Design [5] specifies the fundamental safety functions required to be performed to ensure safety as: the control of reactivity; the removal of heat from the core; and the confinement of radioactive material and the control of operational discharges, as well as the limitation of accidental releases. This part of the SAR should identify and justify the fundamental safety functions to be fulfilled by the specific plant design. It should specify the corresponding structures, systems and components necessary to fulfil these safety functions at various times following a postulated initiating event (PIE).

3.45. In addition to the fundamental safety functions, any specific safety functions should be identified. For example, heat removal should be 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. Guidance on the identification of specific safety functions for plants with light water reactors is given in the annex of Ref. [5].

Deterministic design principles and criteria

3.46. This subsection should provide a general description of the way in which the selected deterministic design principles are embodied in the design. The safety assessment for the plant may be considerably simplified if conservative deterministic principles and criteria are adopted in the design to ensure the adequacy of safety margins in meeting a legal or regulatory requirement. Where aspects of the design are to be based on conservative deterministic principles, such as those embodied in international standards or internationally accepted industrial codes and standards, or in regulatory guidance documents, the use of such approaches should be elaborated in this subsection of the SAR.

3.47. In some cases the design of a nuclear power plant may not fully comply with a specific deterministic principle in a regulatory guidance document. In such cases, either it should be demonstrated that adequate safety margins have been provided by another means or else the proposed design changes or deviations should be justified. In either case the regulatory body should be consulted at an early stage.

Single failure criterion

3.48. In this part of the SAR it should be demonstrated that in general the single failure criterion has been included in a systematic manner to ensure that plant safety functions are preserved. This should include provisions to employ redundancy, diversity and independence, to protect against common cause and common mode failures. Consideration should be 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 hazards. Guidance on the application of the single failure criterion is provided in Ref. [22].

Other safety requirements or criteria

3.49. This part of the SAR should provide a specification of, and a general demonstration of the appropriateness of, any other safety requirements or criteria applied in the design. Consideration should be given to incorporating adequate safety margins; simplification of the design; passive safety features; gradually responding plant systems; fault tolerant plant and systems; operator friendly systems; leak before break concepts, if appropriate; and any other design approaches that have the potential to prevent failures and to enhance the safety of the design. Consideration should also be given to incorporating, where possible, aspects of system design that fail to a safe state.

Probabilistic design criteria

3.50. If probabilistic safety criteria have been used in the design process, these criteria should be specified in this subsection. The compliance of the design with these criteria should also be briefly discussed here; however, the results of the probabilistic safety assessment (PSA) of the final plant design should be provided in the chapter on safety analyses.

Radiation protection

3.51. This subsection should describe in general terms the design approach adopted to meet the radiation protection objective and to ensure that, in all operational states, radiation doses within the installation or due to any planned release of radioactive material from the installation are kept below authorized limits and as low as reasonably achievable (ALARA), as is required, economic and social factors being taken into account (see para. 4.9 of Ref. [21]). It should be demonstrated that:

(a) The radiation doses resulting from a practice are reduced by means of radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses;
(b) Issues such as avoiding the need for workers to be in areas where they are exposed to radiation for long periods of time have been duly taken into account in the design.

3.52. The design of the plant should be such that situations in which exposures of the operators might be high are kept to an acceptable number, with account taken of the appropriate international [23] or national standards. In addition, the ALARA principle should be applied in the operation of the plant to further reduce occupational exposure wherever practicable. This subsection may refer to other sections of the SAR that deal in detail with radiation protection.

Conformance with the design principles and criteria

3.53. This section should provide a brief but complete statement of the conformance of the plant design with the finalized design principles and criteria, which themselves will reflect the safety objectives adopted for the plant.

3.54. If the basic plant design has been modified to meet the criteria, this should be stated. Any deviations from the chosen criteria should be described and justified here. If the criteria have been developed during the evolution of the design, an outline of their development should also be presented here.

Classification of structures, systems and components

3.55. This section should provide information on the approach adopted for the categorization and safety classification of structures, systems and components. It should include information on the methods used to ensure that they 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 SAR). If there is a potential for structures or systems to interact, then details should be provided here 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. A list of safety relevant systems and main structures and components, with their classifications and categorization, should be included as an annex or referenced here. Guidance on options for the classification of structures, systems and components is provided in Ref. [4].

Civil engineering works and structures

3.56. This section should present relevant information on the design of civil engineering works and structures. It should include a discussion of the design principles and criteria and the codes and standards used in the design. It should also briefly review the way in which the necessary safety margins have been demonstrated for the construction of buildings and structures that are relevant to nuclear safety, including the seismic classification of buildings and structures. Any deviations from the requirements for the design should be clearly stated, together with the way in which they have been effected and their justification.

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

(a) Details of the range of anticipated structural loadings, together with the defined performance 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 safety functions.
(c) If a safety and/or seismic classification system for buildings and structures has 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 equipment important to safety is commensurate with the classification of the systems, components and equipment that it contains.
(d) If a building structure or wall is intended to provide separate 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 SAR, as appropriate.

Containment and/or confinement buildings

3.58. This subsection should specify the safety requirements for the containment building itself, including its leaktightness, mechanical strength, pressure resistance and resistance to hazards. It should also describe the main design features of the building provided to comply with the applicable safety requirements. If the design incorporates a secondary containment, this should also be described here.

Equipment qualification and environmental factors

3.59. This section should describe the qualification procedure adopted to confirm that the plant items important to safety 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, throughout the lifetime of the plant. If acceptance criteria are used for the qualification of plant items by testing or analysis, these should be described here. The qualification programme should take account of all identified and relevant potentially disruptive influences on the plant, including internal and external hazard based events. A complete list of equipment items, together with their environmental qualification, should be included as an annex or referenced here. Guidance on options for qualifying structures, systems and components, including the consideration of environmental factors, is provided in Ref. [24].

Human factors engineering

3.60. This section should demonstrate that human factors engineering[2] and human–machine interface issues have been adequately taken into consideration in the development of the design, in order to facilitate interaction between the operating personnel and the plant. This should be valid for all operational states and accident conditions and for all plant locations where such interactions are anticipated.

3.61. This section should include a description of the principles of human factors engineering used for taking into account all factors shaping human performance that may have an impact on the reliability of the operators' performance. The specific design features of systems and equipment that are intended to promote successful operator actions should be considered in the chapter of the SAR on the plant system description and design conformance.

[2] 'Human factors engineering' is engineering in which factors that could influence human performance are taken into account.

Protection against internal and external hazards

3.62. This section should provide a description of the general design measures provided to ensure that the essential structures, systems and components important to safety are adequately protected against the detrimental effects of all the internal and external hazards considered in the plant design.

CHAPTER VI: DESCRIPTION AND CONFORMANCE TO THE DESIGN OF PLANT SYSTEMS

General considerations

3.63. The information to be presented in this chapter of the SAR will inevitably depend on the particular type and design of reactor selected for construction. For some types of reactor many of the sections discussed in paras 3.64–3.113 will be entirely relevant, while for other types they may not apply directly. For these latter cases it should be agreed between the operating organization and the regulatory body which of the plant systems should be described in the SAR. However, as a general rule, all systems that have the potential to affect safety should be described in the SAR, and for such systems the following general approach should be considered.

3.64. This chapter should provide a description of all plant structures, systems and components that are important to safety and should provide a demonstration of their conformance to the design requirements. The level of detail of each description should be commensurate with the safety importance of the item described.

3.65. As stated above, the detailed contents of this chapter are likely to depend on the particular type and design of reactor selected; however, regardless of the reactor type and design, the sections for each particular plant system should be organized into three basic subsections:

(a) *System description:* This should specify the functional requirements and provide a detailed description of the system.
(b) *Engineering evaluation:* This should provide a demonstration that all relevant functional requirements, requirements of industrial codes and standards, and regulatory requirements have been considered adequately. For systems important to safety, this demonstration is supported by assessments of single failures, failure modes and effects analysis, the

assessments of common cause failures and common mode failures, the assessment of overall reliability, and the radiological assessment, where appropriate, with appropriate reference to more detailed documentation to be provided as considered necessary.

(c) *Safety assessment:* For systems important to safety, this subsection should provide a summary statement that the system has sufficient capacity to fulfil its safety function and that there is no credible single failure or operator error that could defeat the performance of the safety function for which the system was designed. For systems not important to safety this subsection should provide a demonstration that the system is sufficiently separated and/or isolated from the systems important to safety to preclude the possibility of its affecting their performance.

3.66. As a minimum, each subsection on system description should provide the following information:

(a) The objective of the system; its safety, seismic, environmental and QA classifications; and the way in which the system relates to the entire plant, including the degree of similarity to systems previously reviewed and approved by the regulatory body for similar units, where appropriate.

(b) A functional design description of the system, including: functional requirements (postulated demands and required performance for all modes of plant operation); a clarification of whether the system is normally in continuous, intermittent or standby operation; specific requirements imposed by regulations, codes and standards and dealing with system reliability, redundancy and interfaces with other systems (including isolation devices on pipes crossing the containment); arrangements for electrical power supplies and instrumentation and control systems; specific requirements, if any, identified on the basis of probabilistic safety analyses; requirements resulting from operational feedback; main elements and their configuration; and simplified functional drawings.

(c) Human factor considerations in the design, including: human factor considerations relating to the human–machine interface for normal startup and shutdown and accident related modes of operation; instrumentation provided to monitor the system operation; the physical location (accessibility) of equipment requiring testing, maintenance and surveillance; displays; alarms; physical interlocks; and an indication of bypassed or inoperable status.

(d) Operational aspects, including: interdependence with the operation of other systems; requirements for technical specifications regarding system operability; provisions for system testing; requirements for system surveillance; and requirements for system maintenance.

(e) Detailed elements of the system design, including: main single line electrical diagrams and other selected schematics according to the safety importance of the system (for electrical and instrumentation and control systems); piping and instrument drawings (for fluid systems); physical location or isometric drawings; precautions against overpressure, such as interlock devices and local overpressure protection (for fluid systems); and protective devices against internal and external hazards, such as watertight seals, missile shields, insulation for high temperatures, electrical protection for short to ground or short to power faults (electrical and instrumentation and control systems); voltage and frequency protection for electrical buses powering large rotating equipment; and interfaces with support systems providing cooling, lubrication, fluid chemistry sampling, air cooling and fire protection.

3.67. As a minimum, each subsection on engineering evaluation should provide the following information:

(a) Identification in a table of the specific technical requirements, requirements of industrial codes and standards, and regulatory requirements, and a demonstration of how each of these requirements has been met by the system design.

(b) Summaries of supporting technical information (with references back to the original topical reports) to demonstrate compliance with technical and industrial codes and standards and regulatory requirements. Examples would include summaries of: reports on material strength and/or corrosion resistance; environmental qualification reports; flammability tests; seismic structural analyses; electromagnetic interferometry and radiofrequency interferometry interference tests; and independent verification and validation analyses of software.

3.68. For any system that is credited (or which supports a system credited) in the safety analyses, the following additional information should be provided in the engineering evaluation:

(a) An assessment of the functions of the system that are directly credited in the safety analysis, including, but not limited to: the timing of system operation; the minimum system performance to meet safety analysis assumptions; any unusual abnormal environmental scenario in which the system is credited with performing.

(b) A demonstration that the physical separation, the electrical and/or fluid isolation devices and the environmental qualification requirements provide sufficient capacity to deliver reliably those safety functions required during and following external events and internal hazards such as seismic events, fires, internal or external floods, high winds and internally generated missiles.
(c) A single failure analysis that is documented in a failure modes and effects analysis and is consistent with the requirement for meeting the single failure criterion established in Ref. [5].
(d) A reliability analysis (including common cause and common mode failures) demonstrating that the system's reliability is adequate to ensure the fulfilment of the intended safety function of the system.

3.69. As a minimum, each subsection on safety assessment should include a statement summarizing the technical basis on which the system in question is judged to be capable of fulfilling its intended function. This judgement should be based on a combination of demonstrated compliance with all applicable regulatory criteria (by the use of regulatory guidance documents and industrial codes and standards) and/or demonstration by means of analysis or testing that sufficient design margins are available. For non-safety-related systems it is sufficient to demonstrate only that a failure of the system in question cannot initiate an event more severe than has already been considered in the safety analyses and cannot degrade the operation of safety related systems.

3.70. The general points described above should in some cases be supplemented by more detailed information relating to the specific features of or functions to be fulfilled by each particular system. The information given in the following sections refers to these specific topics for each of the systems listed. It should if necessary be adapted to suit the design of the plant type concerned.

Reactor

3.71. This section should provide relevant information on the reactor, if possible in a format as described in paras 3.65–3.70. In addition, the following information should be provided to demonstrate the capability of the reactor to perform its safety functions throughout its intended lifetime in all operational modes.

(a) A summary description of the mechanical, nuclear, thermal and hydraulic behaviour of the designs of the various reactor components, including the fuel, reactor vessel internals and reactivity control systems and the related instrumentation and control systems.

(b) Design of the fuel system:
   (i) A description should be provided of the main elements of the fuel system with a safety substantiation for the selected design bases. The justification for the design bases of the fuel system should include, among other things, a description of the design limits for the fuel and the functional characteristics in terms of the desired performance under stated conditions, including normal operation, anticipated operational occurrences and accident conditions.

(c) Design of the reactor internals. Descriptions of the following should be provided:
   (i) The systems of reactor internals, 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. Reference should be made to the other sections of the SAR that cover related aspects of the reactor fuel and also fuel handling and storage.
   (ii) The physical and chemical properties of the components, including thermohydraulic, structural and mechanical aspects, the expected response to static and dynamic mechanical loads, and their behaviour, and a description of the effects of irradiation on the ability of the reactor internals to perform their safety functions adequately over the lifetime of the plant.
   (iii) Any significant subsystem components, including any separate provisions for moderation and fuel location, with corresponding design drawings, and a consideration of the effects of service on the performance of safety functions, including surveillance and/or inspection programmes for reactor internals to monitor the effects of irradiation and ageing on the internal components.
   (iv) The programme to monitor the behaviour and performance of the core, which should cover provisions to monitor the neutronics, dimensions and temperatures of the core.

(d) Nuclear design and core nuclear performance. Descriptions of the following should be provided:
   (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, burnable poison distributions, burnup distributions, control rod locations and refuelling schemes.
(iii) The analytical tools, methods and computer codes (together with information on code verification and validation and uncertainties) used to calculate the neutronics characteristics of the core, including reactivity control characteristics.
(iv) The design basis 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 neutronics stability of the core throughout the fuel cycle, with consideration given to the possible normal and design basis operating conditions of the plant.

(e) Thermal and hydraulic design. Descriptions of the following should be provided:
(i) The design bases, the thermal and hydraulic design for the reactor core and attendant structures, and the interface requirements for the thermal and hydraulic design of the reactor coolant system.
(ii) The analytical tools and methods and computer codes (together with codes for verification and validation information and uncertainties) used to calculate thermal and 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 and hydraulic stability of the core.

(f) Reactor materials:
(i) A justification should be provided of the materials used for the components of the reactor, including the materials of the primary pressure boundary and the materials providing the core support function and any separate moderation function. Information should also be provided on the 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.

(g) Functional design of reactivity control systems:
(i) 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.

3.72. Further discussion on matters to be covered in this section of the SAR is provided in Refs [25, 26].

Reactor coolant and associated systems

3.73. This section should provide relevant information on the reactor coolant system and its associated systems, where possible in the format described in paras 3.65–3.70. In addition, the following information should be provided so as to demonstrate that the reactor coolant system will retain its required level of structural integrity in both operational states and accident conditions.

(a) Integrity of the reactor coolant pressure boundary:
    (i) A description and justification should be provided of the results of the detailed analytical and numerical stress evaluations and studies of engineering mechanics and fracture mechanics of all components comprising the reactor coolant pressure boundary subjected to normal conditions, including shutdown conditions, and postulated accident loads. A list of all components 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.
(b) Reactor vessel:
    (i) Information should be provided 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. This includes the reactor vessel materials, the pressure–temperature limits and the integrity of the reactor vessel, including embrittlement considerations. If the reactor design includes prestressed concrete components or vessel calandria, similar information on these components should also be presented.
(c) Design of the reactor coolant system:
    (i) A description and justification should be provided of the performance and design features that have been implemented to ensure that the various components of the reactor coolant system and the subsystems interfacing with the reactor coolant system meet the safety requirements for design. This should include, where applicable, the reactor coolant pumps, the gas circulators, the steam

generators or boilers, the reactor coolant piping or ducting, the main steamline isolation system, the isolation cooling system of the reactor core, the main steamline and feedwater piping, the pressurizer, the pressurizer relief discharge system, the provisions for main and emergency cooling, and the residual heat removal system, including all components such as pumps, valves and supports.

3.74. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [27].

Engineered safety features

3.75. This section should present relevant information on the engineered safety features and associated systems as described in paras 3.65–3.70. Where necessary, additional system specific information should be added as indicated below.

Emergency core cooling system

3.76. This subsection should present relevant information on the emergency core cooling system and associated fluid systems. The actuation logic should be described subsequently in the section on protection systems and need not be described here.

Containment (or confinement) systems

3.77. This subsection should present relevant information on the containment (or confinement) systems incorporated to localize the effects of accidents, and should include, among other things: the heat removal systems of the containment, the functional design of the secondary containment, the containment isolation system, the protection of the containment against overpressure and underpressure, where provided, the control of combustible gas in the containment, the containment spray system and the containment leakage testing system. Further discussion on matters to be covered in this subsection of the SAR is provided in Ref. [28].

Habitability systems

3.78. This subsection should present relevant information on the habitability systems. The habitability systems are the engineered safety features, systems, equipment, supplies and procedures provided to ensure that essential plant personnel can remain at their posts, including those in the main and supplementary control rooms, and can take actions to operate the plant safely in operational states and to maintain it in a safe condition under accident conditions. The habitability systems for the control room should include shielding, air purification systems, control of climatic conditions and storage capacity for food and water as may be required.

Systems for the removal and control of fission products

3.79. This subsection should provide relevant information on the systems for the removal and control of fission products. 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

3.80. This subsection should present relevant information on any other engineered safety features implemented in the plant design, as described in paras 3.65–3.70. Examples include, but are not limited to: the auxiliary feedwater system, the steam dump to the atmosphere and backup cooling systems. The list of these systems will depend very much on the type of plant under consideration.

Instrumentation and control

3.81. This section should provide relevant information on the instrumentation and control systems as described in paras 3.65–3.70. The reactor instrumentation senses the various reactor parameters and transmits appropriate signals to the control systems during normal operation and to the reactor trip systems and engineered safety features and systems during anticipated operational occurrences and in accident conditions. The information provided in this section should emphasize those instruments and their associated equipment that constitute the protection systems and those systems relied upon by operators to monitor plant conditions and to shut the plant down and maintain it in a safe shutdown state after a design basis accident. Information should also be provided on the non-safety-related instrumentation and control systems used to control the plant in normal operation. These should be described for the purpose of demonstrating that their failure will not impair the proper operation of the safety related instrumentation and control systems or create challenges not already considered in the safety analysis of the plant. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [29].

Protection systems

Reactor trip system

3.82. This subsection should provide relevant information on the reactor trip systems as described in paras 3.65–3.70. In addition, specific information on the following, which are unique to the reactor trip system, should be provided:

(a) The design bases for each individual reactor trip parameter with reference to the PIEs 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 both the main control room and the supplementary control room.
(g) Where reactor trip logic is implemented by means of digital computers, a discussion of the software design and QA programmes, and the software verification and validation programme. Further discussion on matters to be covered in this subsection of the SAR is provided in Ref. [29].

Actuation systems for engineered safety features

3.83. This part of the SAR should provide relevant information on the engineered actuation systems for safety features as described in paras 3.65–3.70. In some plant designs the actuation systems for reactor trip and engineered safety features are designed as one single system. In this case it is appropriate to have a single section describing the actuation system for reactor trip and engineered safety features as one system.

3.84. In addition, specific information on the following, which are unique to the actuation system for engineered safety features, should be provided:

(a) The design bases for each individual actuation system parameter for an engineered safety feature with reference to the PIE whose consequences the parameter is credited with mitigating; interfaces with the reactor trip system (including the use of shared signals and parameter measurement channels); interfaces with non-safety-related systems, together with provisions to ensure the proper isolation of electrical signals; and the means employed to ensure the physical separation of redundant actuation system channels for engineered safety features.
(b) Where the actuation logic for engineered safety features is implemented by means of digital computers, a discussion of the software design and QA programmes, and the software verification and validation programme. Further discussion on matters to be covered in this part of the SAR is provided in Ref. [29].
(c) The specification of actuation system set points for engineered safety features, time delays in system operation and measurement uncertainties and how these relate to the assumptions made in the safety analyses chapter of the report.
(d) Provisions for equipment protective interlocks (e.g. pump and valve interlocks and motor protection) within the actuation system for engineered safety features, together with a demonstration that such interlocks will not adversely affect the operation of engineered safety features.
(e) Provisions for manually initiating engineered safety features from the main control room and the supplementary control room.
(f) Any relevant remote operator and/or automatic control, local control, on–off control or modulating control envisaged in the design and credited in the safety analysis.

Safety related display instrumentation

3.85. This subsection should provide relevant information on the systems for safety related display instrumentation and the computerized plant information system as described in paras 3.65–3.70. In addition, specific information on the following should also be provided.

(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 operators in the control room and the supplementary control room. If data processing and storage are performed by multiple computers, the means of achieving the synchronization of the different computer systems should be described.

All other instrumentation systems required for safety

3.86. This subsection should provide relevant information on any other diagnostic and instrumentation systems required for safety as described in paras 3.65–3.70, and should cover: 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).

Control systems not required for safety

3.87. This subsection should provide brief information on the control systems not required for safety. Specific information on the following should be provided to demonstrate that postulated failures of control systems will not defeat the operation of safety related systems or result in scenarios more severe than those already postulated and analysed in the safety analyses:

(a) A brief description of non-safety-related control systems used for normal plant operations;
(b) A description of any non-safety-related limitation systems (e.g. control grade power reduction systems installed to avoid a reactor trip by initiating a partial power reduction);
(c) A demonstration that such systems do not challenge the operation of safety related systems.

Main control room

3.88. This subsection should provide a description of the general philosophy followed in the design of the main control room. This should include a description of the layout of the main control room, with an emphasis on the human–machine interface. The electrical design standards for equipment located in the main control room have already been described in previous sections and need not be repeated here. If a formal design review (human factors review) for the control room has been performed in developing or upgrading the layout, the results of this review should be summarized in this section.

Supplementary control room

3.89. This subsection should provide an appropriate description of the supplementary control room, including the layout, with an emphasis on the human–machine interface. The electrical design standards for equipment signals routed to the supplementary control room have already been described in previous sections and need not be repeated here. 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. The mechanisms for the transfer of control and communications from the main control room to the supplementary control room should be described in detail so as to demonstrate how this transfer would occur under accident conditions.

Electrical systems

3.90. This section should provide relevant information on the electrical power systems as described in paras 3.65–3.70. In addition, information on the following, specific to electrical systems, should also be presented:

(a) The plant specific divisions of electrical power systems, including the differing system voltages and which parts of the system are considered to be essential.
(b) Substantiation of the functional adequacy of the safety related electrical power systems, including breakers, and assurance that these systems have adequate redundancy, physical separation, independence and testability in conformance with the design criteria. Electrical equipment protection, including the provisions to bypass this protection under accident

conditions. Further discussion on matters to be covered in this part of the SAR is provided in Ref. [30].

(c) A general description of the utility grid and its interconnection to other grids and the connection point to the on-site electrical system (or switchyard). The stability and reliability of the grid should be reviewed in relation to the safe operation of the plant. The physical location of the load dispatching centre controlling the grid should be described, together with the provisions for communications between the dispatch centre, the remote major load centres and the generating plants. The principal means of regulating the voltage and frequency of the external grid should be described. A simplified line drawing showing the main grid interconnections should be provided.

Off-site power systems

3.91. This subsection 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. Special emphasis should be put on 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 should also be described.

On-site power systems

AC power systems

3.92. 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 diesel or gas turbine driven systems, the generator configuration and the non-interruptible AC power system. The power requirements for each plant AC load should be identified, including: the steady state load; the startup kilovolt-amperes for motor loads; the nominal voltage; the allowable voltage drop (to achieve full functional capability within the required time period); the sequence and time necessary to achieve full functional capability for each load; the nominal frequency; the allowable frequency fluctuation; the number of trains, and the minimum number of trains of engineered safety features to be energized simultaneously.

3.93. In addition, information on relevant on-site AC power systems should also be provided to demonstrate that:

(a) In a design basis accident with a subsequent loss of off-site power the required engineered safety feature loads can be sequenced onto the emergency diesel generators without overloading the diesel generators and in time frames consistent with the assumptions presented in the chapter on safety analyses.
(b) 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.
(c) Non-interruptible AC power is continuously provided to essential safety systems and safety related instrumentation and control systems while normal off-site AC power systems are available and during postulated loss of off-site power events.
(d) The maximum frequency decay rate and the limiting underfrequency value for coastdown of the reactor coolant pumps are justified and the minimum number of engineered safety feature trains to be energized simultaneously (if more than two trains are provided) is ensured.

DC power systems

3.94. This part of the SAR should provide relevant information on the DC power system as described in paras 3.65–3.70. In addition, the following information on specific DC power systems should be provided: an evaluation of the long term discharge capacity of the battery (the projected voltage decay as a function of time without charging when subjected to design loads); the major DC loads present (including the non-interruptible AC power system inverters and any non-safety-related DC loads such as the lubrication oil pumps for the turbine bearings); and a description of the fire protection measures for the DC battery vault area and cable systems.

3.95. The power requirements for each plant DC load should be specified, including: the steady state load; surge loads (including emergency conditions); the load sequence; the nominal voltage; the allowable voltage drop (to achieve full functional capability within the required time period); the number of trains; and the minimum number of engineered safety feature trains to be energized simultaneously (if more than two trains are provided).

3.96. Further discussion on matters to be covered in this subsection of the SAR is provided in Ref. [30].

Plant auxiliary systems

3.97. This section should provide relevant information on plant specific auxiliary systems.

Water systems

3.98. This subsection should provide relevant information on the water systems associated with the plant as described in paras 3.65–3.70. It should include, for example, the station service water system, the cooling system for reactor auxiliaries, the makeup system for demineralized water, the ultimate heat sink and the condensate storage facilities.

Process auxiliaries

3.99. This subsection should provide relevant information on the auxiliary systems associated with the reactor process system in a format as described in paras 3.65–3.70. It should include, for example, information on the compressed air systems, the process and post-accident sampling systems, the equipment drainage and floor drainage systems, the chemical control and volume control system, the purification system and the system for controlling the use of boric acid.

Heating, ventilation and air conditioning systems

3.100. This subsection should provide relevant information on the heating, ventilation, air conditioning and cooling systems in a format as described in paras 3.65–3.70. It should include the ventilation systems for the control room area, the spent fuel pool area, the auxiliary and radioactive waste area and the turbine building (in boiling water reactors) and the ventilation systems for engineered safety features.

Other auxiliary systems

3.101. This subsection 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 SAR; for example, the communication systems, the lighting systems, the cooling water system, the starting system, the lubrication system and the combustion air intake and exhaust system for the diesel generator.

Power conversion systems

3.102. This section should provide relevant information on the plant power conversion system, which will depend on the plant type and/or design.

3.103. Information specific to steam and power conversion systems should also be provided on the following, where appropriate:

(a) The performance requirements for the turbine generator(s) in normal operational states and under accident conditions.
(b) A description of the main steamline 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 cleanup system, the condensate and feedwater system, and, where applicable, the steam generator blowdown system. Also to be provided is a description of the water chemistry programme, together with a discussion of the materials of the steam, feedwater and condenser systems.

3.104. For other types of power conversion systems, equivalent alternative information should be provided to demonstrate the systems' compliance with the applicable design requirements.

Fire protection systems

3.105. This section should provide relevant information on the fire protection systems as described in paras 3.65–3.70. It should justify the provisions made to ensure that the plant design provides adequate fire protection. The design should include adequate provisions for defence in depth in the event of a fire, and should provide fire prevention, fire detection, fire warning, fire suppression and fire containment. Consideration should be given to the selection of materials, the physical separation of redundant systems, the seismic qualification of equipment and the use of barriers to segregate redundant trains.

3.106. 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 SAR for this information (e.g. the chapter on safety analyses). Where appropriate, the provisions to ensure the fire safety of personnel may also be described in this section. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [14].

Fuel handling and storage systems

3.107. This section should provide relevant information on the fuel handling and storage systems as described in paras 3.65–3.70. It should include details of the proposed arrangements for the shielding, handling, storage, cooling, transfer and transport of nuclear fuel. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [31].

Fresh fuel

3.108. This subsection should provide relevant information on the fuel handling and storage systems used for fresh fuel as described in paras 3.65–3.70. It should include details of the measures proposed to ensure that fresh fuel is maintained in a safe condition at all times. This should include considerations such as packaging, fuel accounting systems, storage, criticality prevention, fuel integrity control and fuel security.

Irradiated fuel

3.109. This subsection should provide relevant information on the fuel handling and storage systems used for irradiated fuel as described in paras 3.65–3.70. It should include details of the measures proposed to ensure that irradiated fuel is maintained in a safe condition at all times. This should include considerations such as appropriate provisions for radiological protection, criticality prevention, fuel integrity control, including special provisions to deal with failed fuel, fuel chemistry, fuel cooling, fuel accounting systems, fuel security and arrangements for fuel consignment and transport.

Radioactive waste treatment system

3.110. This section should provide relevant information on the radioactive waste treatment systems as described in paras 3.65–3.70. It should include the design features of the plant that safely control, collect, handle, process, store and dispose of solid, liquid and gaseous forms of radioactive waste arising from all activities on the site throughout the lifetime of the plant. This should include the structures, systems and components provided for these purposes and also the instrumentation incorporated to monitor for possible leaks or escapes of radioactive waste. The potential for radioactive waste to be adsorbed and/or absorbed should be considered in deciding on the measures necessary to deal with this hazard. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [32].

3.111. This section should provide a description of the sources of radioactive material that have been covered in the design requirements for the provisions for radioactive waste; if necessary, reference to the chapter of the SAR on decommissioning should be made.

3.112. This section may need to cross-reference the section of the SAR in which radiation protection issues for the plant are considered. This section may also refer to other sections of the SAR in which the operational aspects of radioactive waste management are considered in detail.

Other safety relevant systems

3.113. Any other systems that are claimed to have a safety function, that may assist or support a safety system or that may influence the performance of a safety system should be described under this heading.

CHAPTER VII: SAFETY ANALYSES

General considerations

3.114. This chapter should provide a description of the results of the safety analyses performed to assess the safety of a plant in response to PIEs on the basis of safety criteria and authorized limits on radioactive releases. These analyses include deterministic safety analyses used in support of normal operation, analyses of anticipated operational occurrences, design basis events, beyond design basis events and selected severe accidents, and PSAs. The description may be supported by reference material, where necessary. Additional guidance on the analyses to be performed by the designers and by the operating organization in support of the plant licensing process is provided in chapter 4 of Ref. [4]. This Safety Guide should be used as a reference in the preparation of this chapter of the SAR.

3.115. The safety analyses should proceed in parallel with the design process, with iteration between the two activities. The scope and the level of detail of the analyses should increase as the design progresses so that the final safety analyses reflect the final plant design. Further analyses may be necessary for the justification of proposed design modifications, for taking into account more sophisticated tools or methods for periodic safety reviews.

3.116. The information provided in the chapter on safety analyses 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 authorized limits for radiation doses and radioactive releases for each category of plant conditions. The design, manufacture, construction and commissioning processes should be integrated with the safety analyses so as to ensure that the design intent has been achieved in the as-built plant.

Safety objectives and acceptance criteria

3.117. This section should refer to the principles and objectives of nuclear safety, radiation protection and technical safety applicable to the particular plant design, as previously identified in the chapter on general design aspects [20, 21].

3.118. In addition, detailed acceptance criteria specific to structures, systems and components for different classes of events and types of analyses should be specified. These acceptance criteria should be such that frequent events should have minor consequences and events that may result in severe consequences should be of a very low probability.

3.119. The specification of the detailed acceptance criteria should be well justified and documented in this part of the SAR. The recommendations on the specification of acceptance criteria provided in Ref. [4] should be considered in the preparation of this section of the SAR.

Identification and classification of PIEs

3.120. The methods used to identify PIEs[3] should be described. This may include, among other things, the use of analytical methods such as master logic diagrams, hazard and operability analysis, and failure mode and effects analysis (FMEA). Initiating events that can occur owing to human error should also be considered in the identification of PIEs. Whichever method is used, it should be demonstrated that the identification of PIEs has been performed in a systematic way and has led to the development of a comprehensive list of events.

[3] A PIE is an event identified during design as capable of leading to anticipated operational occurrences or accident conditions. The primary causes of PIEs may be credible equipment failures and operator errors (both within and external to the facility), human induced events or natural events.

3.121. Events should be classified in accordance with their anticipated frequencies and types. The purpose of this classification 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 that includes the most bounding cases in each of the various event groups credited in the safety analyses, but that does not contain events with identical system performance (such as in terms of timing, plant systems response and radiological release fractions).
(c) To allow for differing acceptance criteria for the safety analyses to be applied to differing event classes.

3.122. The basis for event classification should be described and justified. Typically the list of PIEs to be addressed in the SAR will cover anticipated operational occurrences and design basis accidents. It should also include results from the analysis of beyond design basis accidents performed [4, 5]. Some of the design basis accidents or beyond design basis accidents may further develop, if additional faults are assumed, and lead to severe accidents involving significant core degradation and/or off-site radioactive releases. The results of severe accident analyses should also be included in the SAR to the extent that they are needed for plant or system design or to develop the plant accident management programme and to support emergency preparedness [4, 5]. (The terminology used for the PIE types mentioned here is explained in Ref. [4].)

3.123. This process of event classification, in which initiators of all types, both internal and external to the plant, and all modes of operation, including normal operation, shutdown and refuelling, are considered, should lead to a list of different classes of plant specific events to be analysed. Different plant conditions, such as manual control or automatic control, should be investigated. Different site conditions, such as the availability of off-site power or the total loss of off-site power, should also be evaluated, with account taken of the possible interactions between plant manoeuvres and the grid and, where appropriate, possible interactions between different reactor units on the same site. Failures in other plant systems, such as the storage for irradiated fuel and storage tanks for radioactive gas, should also be considered.

3.124. The list of the plant specific events to be analysed and presented in the SAR should include, among others, internal PIEs such as: increase or decrease of heat removal; increase or decrease of reactor coolant flow; reactivity and power anomalies (including mispositioning of a fuel bundle); increase or decrease of the reactor coolant inventory; and the release of radioactive material from a subsystem or component. In addition, a set of internal PIEs, such as loss of support systems, internal floods, fires and explosions, internally generated missiles, the collapse of structures and falling objects, pipe whip and jet effects, and false containment isolation signals leading to the loss of primary pump cooling, derived from other considerations should be taken into account [15].

3.125. The set of external PIEs to be considered should include those due, where appropriate, to: fires; floods; earthquakes; volcanism; extreme winds and other extreme weather conditions; biological phenomena; human induced events such as aircraft crashes and explosions; toxic and asphyxiant gases and corrosive gases and liquids; electromagnetic interference; damage to water intakes [13]; and the effects of explosions at nearby industrial plants and parts of transport networks.

3.126. The recommendations on the identification and classification of PIEs provided in Ref. [4] should be considered in the preparation of this section of the SAR.

Human actions

3.127. This section should describe and justify in general the approaches adopted to take into account human actions in the different types of safety analyses and the methods selected to model these actions in each type of analysis.

Deterministic analyses

3.128. In this section of the SAR all the deterministic analyses performed to evaluate and justify plant safety should be considered. Deterministic safety analysis predicts the plant response to PIEs in specific predetermined operational states. It applies specific rules and uses specific acceptance criteria. The analyses typically focus on neutronics and thermal-hydraulic, structural and radiological aspects that are analysed with different computational tools. As is stated in para. 4.19 of Ref. [4]: “In general, the deterministic analysis for design purposes should be conservative. The analysis of beyond design basis accidents is generally less conservative than that of design basis accidents.” It is acceptable that best estimate codes are used for deterministic analyses provided that they are either combined with a reasonably conservative selection of input data or associated with the evaluation of the uncertainties of the results.

3.129. Deterministic analysis is usually performed by means of the calculation of plant parameters using complex computer codes. The models and the computer codes used for the deterministic analyses as well as the general assumptions made concerning plant parameters, the operability of systems, including control systems, and the operators' actions (if any) in the events should be described. Important simplifications made should be justified. The set of limiting assumptions for safety analysis used in the deterministic safety analyses performed for the different types of PIEs should be described in this section. The methods used to ensure that these assumptions demonstrate that sufficient safety margins are achieved for PIEs of each type should also be described.

3.130. A general summary of the verification and validation processes used for the computer codes should be presented, with reference to more detailed topical reports. Any computer programs used should be identified with reference to the supporting documentation. Emphasis should be given to the substantiation of the applicability of the computer program to the particular event, and reference should be made to the validation documentation, which should refer to relevant supporting experimental programmes and/or actual plant operating data. The validation status of the plant model should also be presented. Further discussion on matters to be covered in this part of the SAR is provided in Ref. [4].

3.131. Any general guidelines for the analysis (such as on the choice of operating states of systems and/or support systems, conservative time delays and operator actions) used in setting up the methods and models used to demonstrate acceptability in the deterministic safety analyses should be described in this section. Guidance on the performance of deterministic accident analyses for pressurized water reactors, boiling water reactors, pressurized heavy water reactors and graphite moderated boiling water cooled pressurized tube reactors is provided in Ref. [33].

Safety in normal operation

3.132. This subsection should demonstrate that the normal operations of the plant can be carried out safely and hence confirm that radiation doses to workers and members of the public and planned discharges and/or releases of radioactive material from the plant are within the authorized limits [4, 23].

3.133. All possible conditions of normal operation should be analysed. Typically these should include conditions such as:

(a) Normal reactor startup from shutdown, to criticality, to full power;
(b) Power operation, including full power and low power operation;
(c) Changes in reactor power, including load follow modes and return to full power after an extended period at low power, if applicable;
(d) Reactor shutdown from power operation;
(e) Hot shutdown;
(f) The cooling down process;
(g) Refuelling during normal operation, where applicable;
(h) Shutdown in a refuelling mode or another maintenance condition that opens the reactor coolant or containment boundary;
(i) Handling of fresh and irradiated fuel.

Anticipated operational occurrences and design basis accidents

3.134. This subsection should provide a description of the results of the analyses of anticipated operational occurrences and design basis accidents performed to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety systems. The analyses should cover all normal operational states, including low power and shutdown modes.

3.135. For each class of PIE it may be sufficient to analyse only a limited number of bounding initiating events that can then represent a bounding response for a group of events. The basis for these selected bounding events should be described in this section. Those plant parameters important to the outcome of the safety analysis should be identified. These would typically include: reactor power and its distribution; core temperature; cladding oxidation and/or deformation; pressures in the primary and secondary system; containment parameters; temperatures and flows; reactivity coefficients; reactor kinetics parameters; and the worth of reactivity devices.

3.136. Those characteristics of the protection system, including operating conditions in which the system is actuated, any time delays and the system capacity after actuation claimed in the design, should be specified and demonstrated to be consistent with the overall functional requirements of the system as described in the chapter on the description of and conformance to the design plant systems of the SAR.

3.137. In some cases different analyses may be necessary for a single PIE in order to demonstrate that different acceptance criteria are met. It should be demonstrated that all the relevant acceptance criteria for a particular PIE are met, and results from as many analyses as necessary should be explicitly included in the SAR.

Analysis of individual groups of PIEs

3.138. For each individual group of PIEs analysed, a separate subsection should be included that provides the following information:

(a) *PIE:* A description of the PIE, the class to which the PIE belongs and the acceptance criteria to be met.
(b) *Accident boundary conditions:* A detailed description of the plant operating configuration prior to the occurrence of the PIE, the model specific and event specific assumptions, and the computer codes used. A description should also be included of systems and operator actions that are credited in the analysis, such as:
    (i) Normally operating plant systems and support systems;
    (ii) Normally operating plant instrumentation and controls;
    (iii) Plant and reactor protection systems;
    (iv) Engineered safety systems and their actuation set points;
    (v) Operator action, if any.
(c) *Initial plant state:* Specific values of important plant parameters and initial conditions used in the analysis; these may be presented in a table. An explanation should be provided of how these values have been chosen and the degree to which they are conservative for the specific PIE being analysed.
(d) *Identification of additional postulated failures:* A discussion of any additional single failure postulated to occur in the accident scenario and a justification of the basis for selecting it as the limiting single failure [33].
(e) *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 graphically 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. For example, in evaluating fuel cladding temperatures, parameters such

as power, heat flux, pressure of the reactor coolant system, fluid inventories of the reactor coolant system, fuel temperatures and flow rates in the emergency core cooling system should be given, where appropriate to the type and design of the reactor. The results should present the relevant plant parameter 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.

(f) *Assessment of radiological consequences:* The results of the assessment of radiological consequences, if applicable, should be presented. The key results should be compared with the acceptance criteria, and conclusions on meeting the acceptance criteria should be clearly stated.

(g) *Sensitivity studies and uncertainty analyses:* The results of sensitivity and uncertainty analyses, if applicable, performed to demonstrate the robustness of the results and the conclusions of the accident analyses should be presented.

Consideration of design capability for beyond design basis accidents

3.139. In addition to the analysis of design basis events, analysis should also be performed to demonstrate the capability of the design to mitigate certain beyond design basis accidents. The choice of the events of this class to be analysed may be made partly on the basis of national regulations, a PSA or any other fault analysis that identifies potential vulnerabilities of the plant. Events that may typically fall into this category are sequences involving more than one single failure (unless they are taken into account in the design basis accident at the design stage), such as: plant AC blackout; anticipated transient without scram; design basis events with degraded performance of the protection system or engineered safety features; and sequences that lead to containment bypass and/or confinement bypass. The basis for the selection of events should be described and justified in this subsection.

3.140. The analyses should use best estimate models and assumptions and may take credit for realistic system action and performance, non-safety-related systems and realistic operator actions. Where this is not possible, reasonably conservative assumptions should be made in which the uncertainties in the understanding of the physical processes being modelled are taken into account.

3.141. The format and content of the analyses of beyond design basis accidents to be presented in this part of the SAR should be consistent with the presentation of the analyses for anticipated operational occurrences and design basis events, with the following modifications:

(a) The objective of the analysis of beyond design basis events and/or the specific acceptance criteria should be stated.
(b) A discussion of the additional postulated failures in the accident scenario should be provided, together with a discussion of the basis for their selection.
(c) Whenever operator action is taken into account, it should be demonstrated that the operators will have reliable information, sufficient time to perform the required actions and procedures to follow, and will have been trained. The key results should be compared with the specific acceptance criteria, and the conclusions on meeting the acceptance criteria should be clearly stated.

Severe accidents

3.142. Where required, this part of the SAR should provide a description in sufficient detail of the analysis performed to identify accidents that can lead to significant core damage and/or off-site releases of radioactive material (severe accidents). The challenges to the plant that such events represent and the extent to which the design may reasonably be expected to mitigate their consequences should be considered, justified and referenced here.

3.143. Detailed analysis of some severe accident sequences should be performed, including, for example, hydrogen fire, steam explosion and molten fuel–coolant interaction. The results of the most relevant severe accident analyses used in the development of the accident management programmes and emergency preparedness planning for the plant should be specified and presented in this section. The accident management measures that could be carried out to mitigate the accidents' effects, and also to provide input for emergency planning and preparedness, should have been identified and optimized in the severe accident analysis. Reference should be made to those relevant chapters of the SAR in which these results are used.

Probabilistic analyses

3.144. An integrated review of the plant design and operational safety should be used to complement the results of the deterministic analyses and to give an indication of the success of the deterministic design in achieving the design objectives. One possible means of undertaking an integrated review is through the use of a PSA. This section should provide a brief description of the scope of the PSA study, the methods used and the results obtained. If any quantitative probabilistic safety criteria or goals have been used in the development of the plant design (as mentioned in the section of the SAR on probabilistic design criteria), these should also be referred to here.

3.145. Topics that should be considered for inclusion in the discussion on the methods and scope of the PSA may include:

(a) Justification of the selected scope of the PSA study;
(b) Accident sequence modelling, including event sequence and system modelling, human performance analysis, dependence analysis and classification of accident sequences into plant damage states;
(c) Data assessment and parameter estimation, including the assessment of the frequency of initiating events, component reliability, common cause failure probabilities and human error probabilities;
(d) Quantification of accident sequences, including uncertainty, importance and sensitivity analyses;
(e) Source term analysis and assessment of off-site consequences.

3.146. The summary results of the probabilistic analyses should be described in this part of the SAR. These results should be presented in such a manner that they clearly convey the quantitative risk measures and the aspects of the plant design and operation that are the most important contributors to these risk measures. This section should refer to the completed plant PSA study being documented as a separate report. The PSA study itself should be made available for review as a separate report to the regulatory body, if required.

3.147. If quantitative probabilistic safety criteria have been used in the development of the plant design, a comparison of the main PSA results with these criteria should be provided to demonstrate compliance. These criteria may relate to both individual and societal risk measures to ensure that all aspects of assessing the risks to the public due to the plant have been adequately considered.

Summary of results of the safety analyses

3.148. This section should provide a summary of the overall results of the safety analyses, confirming that the requirements of the analyses have been met in every respect, providing justification if requirements have been changed, and clearly justifying where requirements have not been met entirely 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 VIII: COMMISSIONING

General considerations

3.149. The operating organization should demonstrate that the plant will be suitable for service prior to its entering the operational phase. The process that the operating organization has adopted to demonstrate this suitability should be presented here. The operating organization should describe the tests intended to validate the plant's performance against the design prior to the operation of the plant. The commissioning programme should, among other things, confirm that the separate plant items will perform within their specifications and that in the various safety systems they function together to ensure that the system's safety functions are reliably performed. In addition, the operating procedures should be validated to the extent practicable as part of the commissioning programme, with the participation of the future operating personnel (see Ref. [6], para. 4.1).

3.150. For this purpose a well planned, controlled and properly documented commissioning programme should be prepared and made ready for implementation. The proposal of the commissioning programme should be presented in this chapter of the SAR. A clear link from the plant safety justification to the commissioning programme should be demonstrated.

3.151. 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. It should also be shown that sufficient numbers of qualified operating personnel at all levels will be directly involved in the commissioning process [6]. 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 process to be followed when the initial outcomes of the tests do not fully meet the design requirements.

3.152. A cursory list of tests to be carried out in the different commissioning phases should be presented. In particular, a summary description of pre-operational and/or startup testing planned for each unique or first of a kind principal design feature should be included in the pre-operational SAR. The summary test descriptions should include the test method and test objectives. Test acceptance criteria, where appropriate, may be presented in the SAR or may be part of the detailed test procedures and referenced separately. A tentative time schedule for the test programme should be presented, with a clear identification of the tests considered prerequisites for other tests. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [34].

3.153. For older plants, possibly after a revision of the SAR, the section on commissioning may be reduced, with some information transferred to supporting references. If relevant, extra information specific to the commissioning of an upgraded plant may be added.

CHAPTER IX: OPERATIONAL ASPECTS

General considerations

3.154. Depending on regulatory practices of States, some topics, such as operational aspects, may be included in either the SAR or in separate documents submitted to the regulatory body. Where this chapter is included it should contain a description of the important operational issues relevant to safety throughout the lifetime of the plant and should also present the operating organization's proposals to address the identified issues adequately. Further discussion on matters to be covered in this chapter of the SAR is provided in Refs [8, 35–38].

Organization

3.155. This section should provide a description of the arrangements of the operating organization and specify the functions 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 demonstrate 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.

Administrative procedures

3.156. This section should provide a description of the general administrative procedures used by the operating organization to ensure the safe management of the plant. The processes to develop, approve, revise and implement plant procedures should be described. A list of the main plant administrative procedures should be provided, together with a brief description of their objective and contents.

Operating procedures

3.157. This section should provide a description of the plant operating procedures. The information presented should be sufficient to demonstrate that the operating procedures for normal operation have been developed to ensure that the plant is operated within the operational limits and conditions (OLCs). It should also demonstrate that the operating procedures provide instructions for the safe conduct of normal operation in all modes, such as starting up, power production, shutting down, cooldown, shutdown, load changes, process monitoring and fuel handling. It should be explicitly demonstrated that the principles of human factors engineering have been considered in the development and validation of the procedures [39].

Emergency operating procedures

3.158. This section should provide a description of the procedures, whether event or symptom orientated, that will be used by the operators in emergencies. A justification of the approach selected should be provided and, where appropriate, linked to the findings of the plant safety analyses. Whichever approach is selected, it should be demonstrated that the required operator actions to diagnose and deal with emergency conditions are covered appropriately. The approach used for verification and validation should be presented, together with a list of the procedures to be followed. It should be demonstrated that the principles of human factors engineering have been considered in the development and validation of the procedures [39].

Guidelines for accident management

3.159. This section should provide a description of the selected approach to plant accident management. The corresponding accident management guidelines developed to prevent severe accidents, and to mitigate their consequences if they do occur, should be described and justified. The information provided should make reference to the accident management programme at the plant, if appropriate. It should be demonstrated that all possible means, safety related or conventional, available at the plant or at neighbouring units or externally, for preventing the release of radioactive material to the environment have been considered. It should also be demonstrated that accident management guidelines have been developed in a systematic way, with account taken of: the results from severe accidents analysed and presented in the SAR; the identified vulnerabilities of the plant to such accidents; and the strategies selected to deal with these vulnerabilities.

Maintenance, surveillance, inspection and testing

3.160. The SAR should specify which safety related plant items will require any form of monitoring to ensure that they remain fit for their purpose and that their operation is within the identified operational limits for reliable and safe operation.

3.161. In this section the SAR should provide a description and justification of the arrangements that the operating organization intends to have in place to identify, control, plan, execute, audit and review maintenance, surveillance, inspection and testing practices that influence reliability and affect nuclear safety.

3.162. The surveillance programme should be such as to verify that the provisions for safe operation that were made in the design and were checked during construction and commissioning continue to be in place throughout the lifetime of the plant, and to provide data to be used for assessing the remaining service life of structures, systems and components. In addition, it should be demonstrated that the surveillance programme is adequately specified to ensure the inclusion of all relevant aspects of the OLCs. It should also be demonstrated that the frequency of surveillance is based on a reliability analysis, including, where available, a PSA and a study of experience gained from previous surveillance results or, in the absence of both, is based on the recommendations of the supplier.

3.163. This section should also include information justifying the appropriateness of the plant inspections, including in-service inspections, required to help demonstrate that the plant meets the specified standards, satisfies the inspection criteria adopted and remains capable of performing the required safety functions. In particular, emphasis should be placed on the adequacy of the in-service inspections of the integrity of the primary and secondary coolant systems, owing to their importance to safety and the severity of the possible consequences of failure.

3.164. The operating organization should also identify all testing that can affect the safety functions of a nuclear power plant. This should include, in addition to a schedule of identified testing, a system for ensuring that testing is initiated, carried out and confirmed within the timescales allowed. This section should also refer to methods for the audit and review of the testing identified. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [37].

Core management and fuel handling

3.165. The SAR should demonstrate that 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 batch refuelling, tests are performed to confirm that the core performance meets the design intent. It should also be shown that the core conditions are monitored and compared with predictions to determine whether they are as expected and are within operational limits. In addition, it should be shown that criteria have been established and procedures established for dealing with failures 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.

Management of ageing

3.166. The operating organization should identify all parts of the plant that can be affected by ageing and should present the proposals made for addressing the issues identified. This includes, among others, the operating organization's proposals for appropriate material monitoring and sampling programmes where it is found that ageing or other forms of degradation may occur that may affect the ability of components, equipment and systems to perform their safety function throughout the lifetime of the plant. Appropriate consideration should be given to analysing the feedback of operational experience with respect to ageing.

Control of modifications

3.167. The operating organization should describe the proposed method of identifying, controlling, planning, executing, auditing, reviewing and documenting the necessary 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. The modification control process should cover the changes made to the plant systems and components, OLCs, plant procedures and process software. It should also be demonstrated that the modification control covers permanent and temporary changes to the plant. When a proposed modification would affect the performance of the operators or the operating organization, it should be demonstrated that provisions are in place to ensure that the principles of human factors engineering are considered and applied throughout the design and implementation of the modifications. Records of all modifications should be retained, and, where necessary, all documentation, procedures, instructions and drawings should be routinely revised to reflect these changes. It should also be demonstrated that the requirements for configuration management are met in the implementation of the plant modifications. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [35].

Qualification and training of personnel

3.168. This section should provide justification that the qualification and training programme for plant staff is adequate to achieve and maintain the required level of professional competence of staff throughout the lifetime of the plant. Information should be provided to describe the initial qualification requirements and the staff training programme, including refresher training and retraining, and also the applicable documentation system for recording the present 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.

3.169. It should be demonstrated that a systematic approach to training is to be adopted. This may include a training programme based on an analysis of the responsibilities and tasks involved in the work, and should apply to all personnel, including managers.

3.170. Where the licensing regime includes provision for the licensing of operators, the report should describe the system and explain the provisions that will be put in place to comply with these licensing requirements.

3.171. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [36].

Human factors

3.172. This section of the SAR should provide a description of the operating organization's proposals for a programme to manage operational issues that are affected by considerations of human factors, including the continuing review and development of the measures in place. This programme should describe the organizational provisions in place to ensure that operators are able to perform effectively in the main control room as well as in other parts of the plant as required, under all operational circumstances, including proposed shift schemes and rotations, the assessment of operators' fitness for duty and other issues related to human factors.

Programme for the feedback of operational experience

3.173. The operating organization should present proposals for a programme for the feedback of operational experience to be implemented. The programme should provide measures to ensure that plant incidents and events are identified, recorded, notified, investigated internally, as appropriate, and used to promote enhanced plant performance and safety culture through the adoption of appropriate countermeasures to prevent recurrences, and should permit the regulatory body to be informed, where necessary. The programme should include consideration of technical, organizational and human factor aspects. Where relevant, arrangements made for reporting and analysing low level events and near misses should be described.

3.174. The programme for feedback of operational experience should also address the provisions for the evaluation of experience gained from operational events at similar plants, the identification of generic problems and the implementation of measures for improvement, if necessary.

3.175. This section of the SAR should demonstrate the suitability of the proposed system for feedback of operational experience for the purpose of analysing the root causes of equipment failures and human errors, improving job descriptions and operational procedures, and assessing the need for backfitting and modernization of the plant, including organizational changes, if necessary. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [38].

Documents and records

3.176. The operating organization should provide details here of the provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and deleting documents and records that relate to the operational activities over the lifetime of the plant. In particular, this should include the operator's documentary provisions for the management of plant configuration, as well as the management of waste and decommissioning of the plant. Further discussion on matters to be covered in this section of the SAR is provided in Ref. [3].

Outages

3.177. The operating organization should provide a description here of the relevant arrangements for conducting periodic shutdowns of the reactor as the operating cycle and other factors require. This should include measures to ensure the safety of the plant during the outage period, as well as measures to ensure the safety of temporary personnel working at the plant at the time. Particular attention should be paid to measures taken to ensure safety during specific circumstances of outage, such as multiple activities, multiple actors from different fields and services, organization and planning, time pressure, management of unforeseen events, feedback of experience of outages and how this experience is analysed and used to improve the management of outages.

CHAPTER X: OPERATIONAL LIMITS AND CONDITIONS

General considerations

3.178. Although practices concerning the explicit inclusion of the OLCs in the SAR differ among States, it is recognized that the OLCs form an important part of the basis on which the operating organization is authorized to operate the plant. In some States OLCs are presented as part of the SAR; in others they are prepared as a separate document that is referenced in the SAR. Whichever approach is used, it should be demonstrated in the SAR that the OLCs have been developed in a systematic way.

3.179. The licensing process should generally include a consideration of the OLCs in the form of controls, limits, conditions, rules and required actions that are formally derived from the safe operating envelope. The safe operating envelope is encompassed by the possible operating states included in the establishment of the design basis. This is to ensure that the operation of the plant will not present an intolerable risk to the health and safety of workers or the public, operation being at all times within the safe operating regime established for the plant. The means by which such control is exercised varies among States and with different reactor types, but generally all options should provide clear and unambiguous instructions to operators that are clearly linked to the safety justification for the plant.

3.180. The OLCs should be based on the safety analyses of the plant and its environment in accordance with the provisions made in the design. The OLCs should be determined with due account taken of the uncertainties in the process of safety analysis. 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 should be incorporated as necessary as a result of testing carried out during commissioning [39].

3.181. If included in the SAR, detailed OLCs for operation should contain numerical values of limiting parameters and operability conditions of systems and components. The corresponding requirements for surveillance, maintenance and repair to ensure that these parameters remain within acceptable limits and that systems and components are operable should also be specified and, where appropriate, justified by means of a PSA. The actions to be taken in the event that operational limits and conditions are not fulfilled should also be clearly established. In some cases, essential administrative aspects, such as the minimum shift composition and the frequency of internal reviews, are also covered by these conditions. Reporting requirements for operational events should also be covered. Further discussion on matters to be covered in this chapter of the SAR is provided in Ref. [39].

CHAPTER XI: RADIATION PROTECTION

General considerations

3.182. This chapter should provide information on the policy, strategy, methods and provisions for radiation protection. The expected occupational radiation exposures during normal operation and anticipated operational occurrences, including measures to avoid and restrict exposure, should also be described. Further discussion on matters to be covered in this chapter of the SAR is provided in Refs [21, 23, 40, 41].

3.183. The description provided should either include a brief description of the ways in which adequate provisions for radiation protection have been incorporated into the design or refer to other sections of the SAR where this information can be obtained. It should be explained how the basic protection measures of time, distance and shielding have been considered. It should be demonstrated that appropriate design and operational arrangements have been made to reduce the amount of unnecessary radiation sources, as recommended in paras 3.76–3.80 of Ref. [41].

Application of the ALARA principle

3.184. This section should provide a description of the operating organization's policy and the operational application of the ALARA (as low as reasonably achievable) principle. It should be in line with the conceptual description, as outlined in para. 3.51, and should demonstrate that as a minimum the recommendations for the application of the ALARA principle, as described in Refs [40, 41], have been followed.

3.185. This section should provide the estimated annual occupancy of the plant's radiation areas during normal operation and in anticipated operational occurrences. In order to reduce radiation doses to workers, the necessity of their presence in certain plant areas where radiation levels are high should be investigated (in order to limit working hours in those areas).

Radiation sources

3.186. This section should provide a description of all on-site radiation sources, with account taken of both contained and immobile sources and potential sources of airborne radioactive material. It should also cover the possible pathways of exposures.

Design features for radiation protection

3.187. This section should provide a description of the design features of the equipment and the facility that ensure radiation protection. It should provide information on the shielding for each of the radiation sources identified, describe the features for occupational radiation protection, describe the instrumentation for fixed area monitoring of radiation and continuous monitoring of airborne radioactive material, and the criteria for their selection and placement, and address design provisions for any decontamination of equipment, if necessary.

3.188. The principles of radiation protection applied in the design should be stated. Examples are:

(a) No person shall receive doses of radiation in excess of the authorized dose limits as a result of normal plant operation;
(b) The occupational exposures in the course of normal operation shall be ALARA;
(c) Dose constraints shall be used to avoid inequities in the dose distributions;
(d) Measures shall be taken to prevent any workers from receiving doses near the dose limits year by year;
(e) All practicable steps shall be taken to prevent accidents with radiological consequences;
(f) All practicable steps shall be taken to minimize the radiological consequences of any accident.

3.189. Where radiation dose targets are included in the design specification, these should be stated here. If relevant, this section should also include any radiation dose targets that relate to the dose levels expected for members of the public from the operation of the plant throughout its operating lifetime [23].

3.190. It should be demonstrated, for the overall design, that suitable provision is made in the design, layout and use of the plant to reduce doses and radioactive releases from all sources. Such provisions should include the adequate design of systems, structures and components so that exposures in all activities throughout the lifetime of the plant are reduced or, where no significant benefit accrues from the activities concerned, eliminated. Reference to the chapter of the SAR on description and conformance to the design of plant systems on this subject may be appropriate.

Radiation monitoring

3.191. This section should provide relevant details of the arrangements for the monitoring of all significant radiation sources, in all activities throughout the lifetime of the plant. This should include adequate provisions for monitoring to cover operational states, design basis and beyond design basis accidents and, where appropriate, severe accidents.

Radiation protection programme

3.192. This section should describe the administrative organization, the equipment, instrumentation and facilities, and the procedures for the radiation protection programme. It should be demonstrated that, as recommended in para. 2.2 of Ref. [41], 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:

(a) Classification of work areas and access control;
(b) Local rules and supervision of work;
(c) Monitoring of individuals and the workplace;
(d) Work planning and work permits;
(e) Protective clothing and protective equipment;
(f) Facilities, shielding and equipment;
(g) Health surveillance;
(h) Application of the principle of optimization of protection;
(i) Source reduction;
(j) Training;
(k) Arrangements for response to emergencies.

CHAPTER XII: EMERGENCY PREPAREDNESS

General considerations

3.193. 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 the full range of accidents (in particular beyond design basis accidents and severe accidents) that would have effects on the environment and the off-site areas where preparations for the implementation of protective measures are warranted. 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 [42].

3.194. 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 protective measures may be warranted in the event of a severe accident.

3.195. 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 exercises to maintain adequate emergency preparedness should be established and justified.

3.196. Further discussion on matters to be covered in this chapter of the SAR is provided in Refs [42, 43].

Emergency management

3.197. This section should contain an appropriate description of the operating organization's response to an emergency.

3.198. A general 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; performing mitigatory actions; taking urgent protective actions on and off the site; protecting emergency workers; assessing the initial phase; managing the medical response; and keeping the public informed.

3.199. Measures for ensuring the protection of the plant staff and how these will be co-ordinated with other emergency response actions should also be described in this section. Where necessary, reference to other sections of the SAR where this issue is discussed should be made.

Emergency response facilities

3.200. Information should be provided about the particular capability of the plant to provide:

(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.

3.201. Descriptions 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 for the assessment of accident progression, radioactive releases and the consequences of accidents

3.202. This section should provide a demonstration that the operator will have measures available for:

(a) The early detection, monitoring and assessment of conditions for which emergency response actions 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 and radiological conditions, in order to modify, as appropriate, ongoing response actions.

3.203. It should be demonstrated that the response of the necessary instrumentation or systems at the plant under abnormal conditions is adequate to ensure the performance of the required safety functions. (A reference to other chapters of the SAR justifying the equipment qualification required may also be acceptable.)

CHAPTER XIII: ENVIRONMENTAL ASPECTS

General considerations

3.204. Practices among States may vary with respect to the inclusion of information on environmental aspects in the SAR. If this is required, this chapter should provide a brief description of the approach taken to assess the impact on the environment of the construction of the plant, its operation under normal conditions and its decommissioning.

Radiological impacts

3.205. This section should provide a description of the measures that will be taken to control discharges to the environment of solid, liquid and gaseous radioactive effluents; these discharges should be in accordance with the ALARA principle. This section:

(a) Should specify any authorized limits and operational targets for solid, liquid and gaseous discharges and measures to comply with such limits;
(b) Should describe the off-site monitoring regime for contamination levels and radiation levels;
(c) Should identify methods to make, store and retain records of radioactive releases that will routinely be made from the site;
(d) Should describe the dedicated environmental monitoring programmes and alarm systems that are required to respond to unplanned radioactive releases and the automatic devices to interrupt such releases, if applicable;
(e) Should identify the measures that will be taken to make appropriate data available to the authorities and the public.

3.206. This section should cover all aspects of site activity that have the potential to affect the radiological impacts of the site throughout the lifetime of the plant, including construction, operation under normal conditions and decommissioning.

Non-radiological impacts

3.207. This section should cover all aspects of site activity that have the potential to affect the non-radiological impacts of the site throughout the lifetime of the plant, including construction, operation and decommissioning. In particular, this section should provide a description of the measures that will be taken to control discharges to the environment of any dangerous solid, liquid and gaseous non-radioactive effluents. This section also:

(a) Should identify the chemical and physical nature of the releases or discharges;
(b) Should identify any authorized limits and operational targets for discharges;
(c) Should describe the off-site monitoring regime for pollution;
(d) Should describe the alarm systems required to respond to unplanned releases;
(e) Should identify the measures that will be taken to make appropriate data available to the public.

CHAPTER XIV: RADIOACTIVE WASTE MANAGEMENT

General considerations

3.208. This chapter should justify 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. It should make reference to the design description of the plant systems for the treatment of radioactive waste provided in the SAR chapter on description and conformance to the design of plant systems.

3.209. A short description of the main sources of solid, liquid and gaseous waste and estimates of their generation rate in compliance with the design requirements should be provided.

3.210. This chapter should also provide information on the characteristics of the accumulation rates and the quantities, conditions and forms of radioactive waste with different states of aggregation and activity level, for normal and abnormal conditions of operation and for accident conditions, and on the methods and technical means for its processing and/or conditioning, storage and transport. 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. Further discussion on matters to be covered in this chapter of the SAR is provided in Refs [32, 44, 45].

Control of waste

3.211. Measures to control or contain the waste produced at all stages of the lifetime of the plant should be described in this section. This section may also deal with proposals to categorize and separate waste, as necessary.

Handling of radioactive waste

3.212. Measures to handle safely waste of all types produced at all stages of the lifetime of the plant should be described in this section. This should include the provisions for the safe handling of the generated waste while transporting it from the point of origin to the specified storage point. The section should include a consideration of the possible need to retrieve waste at some time in the future, including during the decommissioning stage.

Minimizing the accumulation of waste

3.213. Measures to minimize the accumulation of waste produced at all stages of the lifetime of the plant should be described in this section. This should include measures taken to reduce the waste arising to a level that is as low as practicable. The assessment should show that both the volume and the activity of the waste are minimized in such a way as to meet any specific requirements that may be posed by the design of the waste storage facility.

Conditioning of waste

3.214. Measures to condition the waste produced at all stages of the lifetime of the plant should be described in this section. Where it is considered prudent, waste may be processed in accordance with established procedures, and the options considered should be described here. However, consideration should also 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.

Storage of waste

3.215. Measures to store the waste produced at all stages of the lifetime of the plant should be described in this section. This section should consider the quantities, types and volumes of radioactive waste and the need to categorize and separate waste within the provisions for storage. 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.

Disposal of waste

3.216. Measures to dispose safely of the waste produced at all stages of the lifetime of the plant should be described in this section. This should include the measures for ensuring the safe transport of waste to another specified location for longer term storage, if necessary.

CHAPTER XV: DECOMMISSIONING AND END OF LIFE ASPECTS

General considerations

3.217. Decommissioning of the plant will become necessary either at the end of the lifetime of the plant or earlier if the operator so decides. The capability for decommissioning the plant should be demonstrated before initial criticality or before plant operation commences. This chapter of the SAR should contain the proposals anticipated at this point for the eventual decommissioning of the plant. It should be periodically updated to allow for an increasing level of detail and to reflect developments in the strategy for decommissioning. Further discussion on matters to be covered in this chapter of the SAR is provided in Refs [46, 47].

Decommissioning concept

3.218. This section of the SAR should briefly discuss the proposed decommissioning concept, with the following aspects taken into account:

(a) Design solutions that minimize the amount of waste material produced and that facilitate decommissioning;
(b) Consideration of the type, volume and activity of radioactive waste produced during the operational and decommissioning phases;
(c) Identified options for decommissioning;
(d) Planning, phasing or staging of the decommissioning process, including appropriate surveillance requirements throughout the process;
(e) Adequate documentary control and maintenance of suitable and sufficient records;

(f) Anticipated organizational changes, including provisions in place to preserve the institutional knowledge that will be needed at the decommissioning phase.

Provisions for safety during decommissioning

3.219. 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. Special attention should be paid to the following aspects:

(a) Radioactive (airborne and liquid) discharges during the process should be in accordance with the ALARA principle and should be kept at least within authorized limits;
(b) The practicability of adherence to the concept of defence in depth against radiological hazards during the decommissioning process should also be demonstrated.

Differing approaches to decommissioning

3.220. This section should present a description of the options identified and the method chosen for decommissioning, with corresponding justification. 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). Any options and their effects on the timescale for the decommissioning process should also be discussed.

Planning of the preliminary work

3.221. This section should present a tentative programme of decommissioning work, 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 development of a rational strategy for decommissioning, including the identification of a staged approach to decommissioning, if appropriate;
(c) The development of a SAR for decommissioning;

(d) The development of a programme for bringing the reactor to a safe condition for total or partial dismantling;
(e) The development of a programme for ensuring that services (heating, electricity and water supply) will be available to support the work;
(f) The development of a programme for providing adequate facilities for the sorting, processing, transport and storage of the radioactive waste arising during decommissioning;
(g) Providing for the physical protection, monitoring and surveillance of the unit during the decommissioning stages identified;
(h) The observation of the licensing process throughout decommissioning.

4. REVIEW AND UPDATING OF THE SAR

GENERAL CONSIDERATIONS

4.1. In the licensing of a new plant the preliminary, intermediate and final SARs are important documents, compiled by the operating organization, that the regulatory body uses in assessing the adequacy of the plant design and the suitability of the licensing basis. It should be noted that the SAR may be only one of several sources of information and that the final safety justification that the regulatory body accepts may consist of a far wider range of information.

FORM OF THE SAR

4.2. The operating organization should agree with the regulatory body on the form of the presentation, storage and use of the SAR (e.g. electronic or paper form). In taking this decision full account should be taken of the regulatory position on the acceptability of alternative forms of reports, in accordance with national laws and regulations, where appropriate.

ROUTINE REVISIONS TO THE SAR

4.3. Since the SAR is part of the overall justification of plant safety, it should reflect the current state and the licensing basis of the plant and should be kept up to date accordingly (this is sometimes referred to as a ‘living’ SAR).

4.4. The need for routine review and updating of the SAR depends on the role of the report in the ongoing licensing process. Factors that a regulatory body should consider in determining the need for the routine updating of a SAR and the updating frequency include: the way in which the overall safety justification and documentation are maintained; the implementation of major modifications to the plant and revisions to the OLCs; and the frequency of plant periodic safety reviews. Guidance on the performance of periodic safety reviews and the periodic updating of the SAR is provided in Ref. [10].

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GLOSSARY

operation. All activities performed to achieve the purpose for which a facility was constructed. For nuclear power plant, this includes maintenance, refuelling, in-service inspection and other associated activities.

operational limits and conditions. A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.

plant states

operational states		**accident conditions**			
				beyond design basis accidents	
normal operation	anticipated operational occurrences	a	design basis accidents	b	severe accidents
				Accident management	

a = Accident conditions which are not explicitly considered design basis accidents but are encompassed by them.

b = Beyond design basis accidents without significant core degradation.

accident conditions. Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

accident management. The taking of a set of actions during the evolution of a beyond design basis accident:

— to prevent the escalation of the event into a severe accident;
— to mitigate the consequences of a severe accident; and
— to achieve a long term safe stable state.

anticipated operational occurrence. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a nuclear power plant but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

design basis accident. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

normal operation. Operation within specified operational limits and conditions.

operational states. States defined under normal operation and anticipated operational occurrences.

severe accident. Accident conditions more severe than a design basis accident and involving significant core degradation.

postulated initiating event. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions.

protection system. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

safety function. A specific purpose that must be accomplished for safety.

safety system. A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

single failure. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

single failure criterion. A criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure.

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