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IAEA SAFETY STANDARDS SERIES

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

DRAFT SAFETY GUIDE - VERSION 4

DS 309

INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA

Supersedes 50-SG-G2

(Front inside cover)

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorised to establish standards of safety for protection against ionising 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 may 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.

Safety Guides (green lettering) recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

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.

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

www.iaea.org/ns/coordinet

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.

Reports on safety and protection in nuclear activities are issued in other series, in particular the **IAEA Safety Reports Series**, as informational publications. Safety Reports may describe good practices and give practical examples and detailed methods that can be used to meet safety requirements. They do not establish requirements or make recommendations.

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

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.

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 advisory 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 recognised 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 may 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.

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1. GENERAL INTRODUCTION

BACKGROUND

1.1. In order to obtain regulatory permission to build and then operate a nuclear power plant an authorisation shall be granted by the national regulatory body. The paragraphs 5.3 and 5.4 of the IAEA Requirements on Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport safety [1] state: " Prior to the granting of an authorisation, 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 …..". " The regulatory body shall issue guidance on the format and content of documents to be submitted by the operator in support of applications for authorisation. The operator shall be required to submit or make available to the regulatory body, in accordance with agreed time-scales, all information that is specified or requested." This information should be presented in a form of a report, herein after referred to as Safety Analysis Report (SAR).

1.2. The requirements for Safety Analysis Reports are heavily dependent upon the type of regulatory regime adopted by a Member State, which may affect the scope and depth of the information presented in the document. For countries with small nuclear power programmes or importing NPPs there may be a significant reliance on the vendor country practices or on international work which helps to demonstrate the safety of the design. In any event it is important that there is 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 agree the programme of submissions. Some Member States give very comprehensive guidance as to the contents of Safety Analysis Reports; a widely quoted and used document is the Standard Format produced by the USNRC [2]. This document considers the USNRC document and other relevant references. It draws heavily on the IAEA safety standards [1,3,4,5,6] and other documents to present one possible format and content option for a comprehensive Safety Analysis Report for any type of nuclear power plant. Alternative formats to that presented in this document may be used, in such cases the recommendations of this report should be regarded as potential ingredients for use in such alternative formats.

OBJECTIVE

1.3. The objective of this document is to provide guidance on the possible format and content of a Safety Analysis Report that supports a request to the regulatory body for authorisation to construct and/or operate a nuclear power plant. As such, this document details requirements contained in the IAEA standard GS-R-1 (paragraph 5.4, [1]), and in the related safety guide [3].

1.4. Guidance on the assessment and verification to be performed by the design and operating organisation when preparing the SAR is provided in an IAEA Safety Guide [4]. Guidance on the review and assessment to be performed by the regulatory body during the authorisation process is also provided in an IAEA Safety Guide [7].

SCOPE

1.5. This guide is aimed primarily at land based stationary thermal nuclear power plants but 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 NPP proposed, and this will determine how sections of this guide are included in the SAR. Although biased towards new plant the information presented would be also useful to existing nuclear power plants (NPPs) when operators are periodically reviewing their existing Safety Analysis Reports to identify any areas where further improvements may be appropriate and/or review the licensing basis. This guide covers at a same level of importance both technical and human factors aspects which have to be addressed adequately in a SAR in order to substantiate plant safety.

2. GENERAL CONSIDERATIONS

2.1. Safety Analysis Reports represent an important communication between the operator and the regulatory body which helps to form the basis for licensing an NPP and present the basis for the safe operation of the plant. They will therefore need to contain sufficiently precise information on the plant and its operating conditions and will typically include information such as safety requirements, design basis, site and plant characteristics, operational limits and conditions and safety analyses, in such a way that the regulatory body is able to evaluate independently the safety of the plant. In particular, it will be important to demonstrate that the interdependence between the technical and human factors safety aspects have been considered all along the report. The Safety Analysis Report ideally presents sufficient information on the plant so that for the purposes of nuclear and radiation safety assessment, the amount of additional documentation to allow the authorisation process to proceed is minimised. The Safety Analysis Report may refer out to more detailed supplementary information that should be made available to the regulatory body, if requested.

2.2. It is common practice in many Member States that the Safety Analysis Reports are issued in successive and complementary parts, which may include:

- An initial (Preliminary) Safety Analysis Report or Pre-Construction Safety Analysis Report (PCSAR) that supports the application for authorisation for siting and/or construction.
- An updated (Intermediate) Safety Analysis Report or Pre-Operation Safety Analysis Report (POSAR) that precedes an application for authorisation to operate within the licensing process.
- A finalised (Final) Safety Analysis Report or Station Safety Analysis Report (SSAR) which incorporates the revisions to the intermediate report prior to the NPP entering first routine operation within the licensing process.

2.3. From the regulatory position it is most desirable to be kept well informed about the process of site selection and the subsequent development of the selected site and plant. It is therefore recommended that sections of the Safety Analysis Report be submitted to the regulatory body at an early stage and in accordance with an agreed programme; this approach is likely to facilitate a smooth review process and should help prevent delays in construction and commissioning.

2.4. The initial report (PCSAR) may be of limited scope. Informal contacts before the pre-construction review stage are encouraged between those planning to build a reactor and the regulatory body in order to develop a mutual understanding of the nature of the project and the likely regulatory requirements. The report will include a statement of safety principles adopted and safety objectives set for the intended design. It should include the manner of complying with the fundamental safety principles and a statement of how the safety objectives have been met. It will 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 the life cycle of the plant. The safety features incorporated into the design together with the possible challenges to the plant which have been considered should be described having due regard to any site specific considerations. The amount of information provided in the preliminary report may be influenced by the extent to which the proposed reactor design is based on a generic type or standard design that has been through the licensing process previously, including the production of a SAR.

2.5. The intermediate report (POSAR) revises and provides more specific information on the topics outlined in the PCSAR and on any departure from or revisions to the safety provisions or design intent set out in the preliminary report. In essence the POSAR justifies the finalised detailed design of the plant and presents a demonstration of its safety. In addition, the POSAR deals in greater detail than the PCSAR with matters relating to the commissioning and operation of the plant during this phase of the life of the NPP. The POSAR will provide more recent information to update the licensing basis for the plant.

2.6. The final report incorporates any necessary revisions that have been required to the intermediate report (POSAR), following commissioning and in preparation for first entry into routine operation of the as built NPP and taken into consideration during the licensing process. The final report should clearly demonstrate that the plant meets its design intent. Systematic updating of such a Safety Analysis Report would then become a matter for the operator during the remaining life-cycle of the plant. This would normally be done periodically to reflect any operating experience feedback, plant modifications and improvements, new regulatory requirements or any proposed change to the licensing basis.

2.7. The SAR is prepared by the operator for submission to the regulatory body to enable them to assess the suitability of the plant for licensing. The following sections of this document set out a possible list and description of topics for inclusion in a comprehensive Safety Analysis Report for a Nuclear Power Plant. A standard format of the Safety Analysis Report is also discussed in the appropriate sections. However, in the application of this guide adaptations may be needed to reflect the differences in plant licensing phases and the licensing practices in Member States. Where required by Member States parts of the SAR may be made available to the public.

2.8. While the main purpose of the SAR is to provide necessary information to the regulatory body, it is also important that plant staff and management have an understanding of the main findings of the SAR. This may be aided by providing supplementary documentation which summarises the relevant sections of the SAR.

3. FORMAT AND CONTENTS OF A SAFETY ANALYSIS REPORT

I - INTRODUCTION

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

- the main purpose of the SAR;
- a description of the existing authorisation status;
- an identification of the designer, vendor, constructor and operator of the nuclear power plant;
- an identification of any similar (or identical) NPPs that the regulator has already reviewed and approved and what specific differences or improvements have been made since such approval was issued;

- the main information about the preparation of the SAR;

- a description of the SAR structure, the aims and scope of each of its sections, and the intended connections between them.

II - GENERAL PLANT DESCRIPTION

3.2. This chapter should present 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 obtain an adequate understanding of the overall 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 which 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 should be provided for their appropriateness. Any changes or deviations made to 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 to inform the regulatory body of such changes.

Basic technical characteristics

3.5. This section should present briefly (in table form where appropriate) the principal elements of the overall installation, including the number of plant units where appropriate, the type of plant, the principal characteristics of the plant, the primary protection system, the type of the nuclear steam supply system or gas turbine cycle, the type of containment structure, the core thermal power levels, the corresponding net electrical output for each thermal power level, etc., and any other characteristics that are necessary for understanding the main technological processes included in the unit design. It may be useful to compare the plant design with earlier similar designs already approved by the regulatory body, to identify the main differences and assist the justification for any modifications and improvements made. It is recommended that a list of the selected plant characteristics is included in an Appendix to the SAR.

Layout and other information

3.6. The basic technical, and schematic drawings of the main plant systems, and equipment should be incorporated here along with the physical and geographical location of the facility, connections with the electrical grid, means of physical access to the site by water, rail, and road. The operator should provide general layout drawings for the entire plant. The illustrations should be complemented with a brief description of the main items of plant and equipment, together with its objective and interactions. References to other SAR chapters that present detailed descriptions of specific systems and equipment should be made where necessary.

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

3.8. This section may, if required, also include information about the provisions made for physical protection of persons, plant, systems, equipment and access routes. In some Member States this may also include coverage of steps taken to provide protection in the event of malicious action on- or off-site.

Operating modes of the nuclear power unit

3.9. All possible operating modes of the unit should be described, including: start-up, power operation, shutdown, refuelling, and any other allowed modes. The permissible periods at different power level should be described in case of deviation from normal operation conditions. In this event the methods for restoration of the unit to the normal condition should be indicated.

Material incorporated by reference

3.10. This section should provide a tabulation of the topical reports that are incorporated by reference as part of the Safety Analysis Report. Results of tests and analyses (for example manufacturers' material test results and qualification data) may be submitted as separate reports. In such cases, these reports should be tabulated in this section, and referenced or summarised in the appropriate section of the SAR.

III - MANAGEMENT OF SAFETY

3.11. This chapter should describe and evaluate the operating organization's management structure, and the procedures and processes that achieve satisfactory control of all aspects of safety through the plant life cycle. This should include the role of on-site safety assessment organizations, and any off-site safety advisory committees that advise the operating organization's management. The aim is to demonstrate that the operator is able to fulfil its responsibility to operate the plant safely throughout its life cycle.

Specific Management Processes Aspects

3.12. This section should describe the site and corporate management structure and technical support organisation of the operator. The way in which management control of the design and operating organisations will be achieved to promote safety and the measures employed to confirm to the operator in the first instance, and then to the regulatory body, that implementation and observance of the management safety procedures is adequate should be presented. Further information on matters to be discussed in this section of the SAR may be found in the reference documents [8].

Consideration of Safety Culture

3.13. This section should contain the operator’s proposals to encourage the development, maintenance and improvement of a good safety culture throughout the plant life cycle. This section should present a demonstration that the necessary arrangements in respect of safety culture are adequate and in place at the NPP. The arrangements should be aimed at promoting good awareness of all aspects of safety on the plant and regularly reviewing with staff the level of safety awareness achieved on the site. The operator should where possible identify indicators of safety culture and develop a programme to monitor the safety culture against the indicators; the staff should be consulted on the indicators and kept informed of the outcome from the reviews, with action taken to reverse indications of declining safety levels.

Quality assurance (QA)

3.14. This section should describe the principal aspects of the QA system developed for the proposed plant and demonstrate that appropriate quality assurance provisions, including a QA programme, audit, review, and self-assessment functions, are implemented for all safety-related plant items, procedures and activities during the life cycle of the plant covered by the Safety Analysis Report . These activities should include design, procurement of goods and services (including use of contractors' organisations), plant construction and operation, maintenance, repair and replacement, in-service inspection, testing, refuelling, modification, commissioning and decommissioning. The QA arrangements presented in this section should cover safety matters relating to the plant, throughout its entire life cycle. Further information on matters to be included in this section of the SAR may be found in the reference documents [9].

Monitoring and review of safety performance

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

3.16. This section should also describe the way by which the operator intends to identify any evolution of the organization that could lead to safety performance degradation and justify the appropriateness of the measures planned to prevent such a degradation. Further information on matters to be included in this section of the SAR may be found in the reference documents [8].

IV - SITE EVALUATION

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

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

- Site specific hazard evaluation for external events (of human or natural origin);
- Design targets in terms of recurrence probability for external events;
- Definition of the design basis for external events;

[1] In some Member States some groups of information are collected in a so-called “Environmental Report”. However, they are addressed here as they are discussed in the Requirements for Siting [11] and in all IAEA relevant Safety Guides, being an important set of safety-related information.

- Collection of site reference data for plant design (geo-technical, hydrological, etc.);
- Evaluation of the impact of site related issues to be considered in the emergency planning and accident management sections of the SAR;
- Arrangements for monitoring of site related parameters throughout the plant life.

3.19. Considerations concerning the site exclusion/acceptance criteria applied for the preliminary screening on site suitability since 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 it may be one of the sources of uncertainty in the final safety evaluation. Therefore the SAR should address the measures employed to account for such uncertainty levels.

3.21. Further information to be considered in the preparation of this chapter of the SAR may be found in the reference documents [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 where there is a need for consultation on the control of activities with the potential to affect plant operation, including flight exclusion zones. This would include relevant data on population distribution and density around NPP site and the arrangement of public and private facilities (airports, harbours, rail transport centers, factories and other industrial sites, schools, hospitals, police, fire fighting, and municipal etc.). It should also cover uses of soil and water resources in the surrounding area, for example agriculture, and include an assessment of any possible interaction with the proposed NPP.

3.23. Site related data referring to geotechnical soil properties and groundwater hydrology should also be provided. The investigation campaigns aimed at the collection of data for foundation design, evaluation of soil structure interaction effects, construction of earth and buried structures and soil improvement at the site should also be described [17].

3.24. The SAR should present the relevant data and their associated range of uncertainty to be used in structural design and radioactivity dispersion studies. Reference should be made to the technical reports describing in details the conduct of the investigation campaigns, their extension, the origin of data collected on a regional basis and/or on a bibliographic basis. The design of earth structures and site protection measures [11], if applicable, should also be documented. A projection of anticipated developments to the above mentioned information should also be provided, and be updated as required.

Site specific hazard evaluation

3.25. This section should present the results from a detailed evaluation of natural and human induced hazards at the site. Where administrative measures are employed to mitigate the site hazard (particularly in case of human induced events), their implementation should be presented, together with roles and responsibilities involved in their enforcement.

3.26. The SAR should discuss the screening criteria used for each hazard (envelope, probability thresholds, incredible events etc.) and the expected impact of each of them in terms of originating source, potential propagation mechanisms and predicted effects at the site [11,13,14,15,16].

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

3.28. It should be demonstrated that appropriate arrangements are in place to periodically update site specific hazard evaluations with the results of updated evaluation methodologies, monitoring data and surveillance activities.

Proximity of industrial, transportation and military facilities

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

Activities at the NPP site that influence the unit's safety

3.30. Any production processes or related activities on the NPP site, which if incorrectly carried out might influence the safe operation of the unit should be presented and described, e.g. vehicular traffic in the plant area, storage and potential spillage of fuels, gas and other chemicals, intake (control room ventilation) or contamination by harmful particles/smoke/gases etc.

3.31. Site protection measures (dams, dikes, drainage, etc.) and any site modification (soil substitution, modifications to site elevation etc.) are usually considered part of the site characterisation stage and therefore their design basis assessment should be addressed in this section of the SAR. This assessment might be made on the basis of guidance documents and the general references [11,17,18].

Hydrology

3.32. This section should contain sufficient information to allow an evaluation of the potential effect of hydrological site conditions on the plant design, performance requirements and safe operation. These conditions should include phenomena, such as abnormally heavy rainfall, run off floods from watercourses, reservoirs, adjacent drainage areas and site drainage. It 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, seishos and the combined effects of tide and heavy wind should be evaluated. This section also impacts the assessment of the transport of radioactive materials to and from the site and the dispersion of radionuclides into the environment. Further information on matters to be included in this section of the SAR may be found in the reference documents [18].

Meteorology

3.33. This section should provide a description of the meteorological aspects relevant to the site and its surrounding areas, taking into account regional, and local climatic effects. To this aim, data from on-site meteorological monitoring programmes should be documented. Among others, the extreme values of temperature, humidity level, the rainfall, the wind speed from straight and rotational winds and the snow loads should be evaluated in relation to the design. The potential for lightning and “wind-borne debris” to affect plant safety should be addressed, where appropriate. The informatiom in this section will impact the assessment of the transport of radioactive materials to and from the site and the dispersion of radionuclides into the environment. Further information on matters to be included in this section of the SAR may be found in the reference documents [19].

Seismology

3.34. This section should provide information regarding the seismic, geological and tectonical characteristics of the site and the region surrounding the site. The seismic hazard evaluation should be based on a suitable geo-tectonic model substantiated by appropriate evidence and data. The results of this analysis, further used in other sections of the SAR which consider structural design, component seismic qualification and safety analysis, should be well identified. Further information on matters to be included in this section of the SAR may be found in the reference documents [16,17].

Radiological conditions due to external sources

3.35. The radiological conditions at the NPP site found in the environment, taking into account the radiological contribution from neighbouring units and other external sources, if any, should be described in sufficient detail to serve as an initial reference point and to permit a regulatory view of site radiological influences to be developed.

3.36. A short description may be presented of the available radiation monitoring systems and corresponding technical means for detection of radiation and radioactive contamination that may occur from these sources. If appropriate this section may reference other relevant sections of the SAR concerned with radiological aspects of licensing the NPP.

Site related issues in the emergency planning and accident management

3.37. The accident management strongly relies on the availability of adequate access and egress roads, sheltering, supply networks in the site vicinity. Many hazard scenarios for the site are expected to affect the site vicinity as well and therefore the possibility of personnel evacuation and access to the site. The availability of local transport networks and communications networks during and after an accident scenario is a key issue for the implementation of a suitable emergency plan. This section of the SAR should discuss the feasibility of emergency arrangements in terms of access to the plant and of transportation needs in case of a severe accident, showing that the requirements for infrastructures external to the site are met. The need for administrative measures should be identified together with the relevant responsibilities of administrations other than the operating utility.

Monitoring of site related parameters

3.38. The provisions to monitor site related parameters such as seismic, atmospheric, water, groundwater, demography, industrial activity and transportation developments should be presented in this section. This may be used to provide necessary information for emergency operator actions in case of external events, to support the periodic safety review at the site, to develop radiation dispersion modelling and as confirmation of the completeness of the site-specific hazards.

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

3.40. The SAR should describe in some detail the strategy for monitoring, and the use of the results in preventing, mitigating, and forecasting the effects from site related hazards.

V - GENERAL DESIGN ASPECTS

3.41. This chapter should briefly outline the general design concept and approach used to comply with the fundamental safety objectives [20,21], which should be relevant throughout the life cycle of the plant. The actual compliance of the design with the specific technical safety requirements should be provided 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 design principles used in the design should be presented in this section. This may be based on the objectives presented in para 2.2, 2.4 and 2.5 of the IAEA Requirements for Safety of Nuclear Power Plants, NS-R-1 [5], which refers to the general nuclear safety objective, the radiation protection objective and the technical safety objective, as defined by the IAEA.

Defence in-depth

3.43. This part of the SAR should describe in general terms the design approach used to incorporate the defence in depth concept into the design of the NPP. It should be demonstrated that the defence in depth concept has been considered for all safety related activities including organisational, behavioural and design related. The approach should ensure that multiple defence barriers exist within the design features to provide protection against operational occurrences and accidents regardless of 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 events and assist the performance of important safety functions should be included. Guidance on the implementation of defence in depth concept may be found in the reference documents [5].

Safety Functions

3.44. This part of the SAR should identify and justify the specific safety functions to be fulfilled by the specific plant design and the corresponding structures, systems and components that are necessary to fulfil these safety functions at the various times following a postulated initiating event. The IAEA Safety Standards Series No. NS-R-1 on Safety of Nuclear Power Plants: Design [5], specifies the fundamental safety functions which should be performed to ensure safety as:

- control of reactivity;
- removal of heat from the core; and
- confinement of radioactive material and control of operational discharges, as well as limitation of accidental releases.

3.45. It is important in addition to the fundamental safety functions to identify any other specific safety functions. For example the heat removal should be considered as a safety function which is needed not only for reactor core safety, but also for any other part of the plant containing radioactive materials, which need cooling, e.g. spent fuel pools and storage, etc. Guidance on the identification of specific safety functions for the light water reactor type of NPP can be found in the Annex of the IAEA Safety Standards Series No. NS-R-1 on Safety of Nuclear Power Plants: Design [5].

Deterministic design principles and criteria

3.46. The plant safety assessment may be considerably simplified if a design adopts conservative deterministic principles and criteria to deal with the issue of assuring the adequacy of safety margins when meeting a legal or regulatory requirement. Where aspects of the design are to be based on conservative deterministic principles such as embodied in internationally accepted industrial codes and standards, or in regulatory guidance documents issued by the regulatory body, the use of such approaches should be elaborated in this section of the SAR. The way in which the deterministic design principles are embodied in the design should be explained in this section.

3.47. In some cases a nuclear power plant design may not fully comply with a specific deterministic principle in a regulatory guidance document. In such a case it is necessary to demonstrate in the SAR that adequate safety margins are provided by another means or to justify those situations where the deterministic principles have not been entirely complied with and design changes or deviations are proposed. In these cases it is recommended that the regulatory body is consulted at an early stage.

Single failure criterion

3.48. It should be demonstrated in the SAR that the single failure criterion has been included in a systematic manner to ensure that plant safety functions are preserved. The need to ensure that systems, particularly systems important to safety, are not vulnerable to single failures should be demonstrated in the SAR. This should include provisions to employ redundancy, diversity and independence, to protect against common cause and common mode failures. Consideration should be given for the single failure to occur while a redundant train of a system is out for maintenance and/or impaired by hazards. Guidance on application of single failure criterion is provided in the IAEA Safety Series 50-P-1 publication on Application of the single failure criterion.

Other safety requirements

3.49. Consideration should also be given to including adequate safety margins, design simplification, passive safety features, equipment categorisation and classification, gradually responding plant systems, fault tolerant plant and systems, operator-friendly systems, leak-before-break concepts if appropriate and any other approaches which have the potential to avoid the likelihood of failures and enhance the safety of the design. Also, where possible, consideration should be given to incorporate 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 described in this chapter. The design compliance with these criteria should be briefly discussed here as well, however the results from the probabilistic safety assessment (PSA) of the final plant design should be provided in the safety analysis chapter.

Radiation Protection

3.51. This section should describe in general the design approach used to comply with the radiation protection objective and ensure that, in all operational states, radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and “as low as reasonably achievable” (ALARA), economic and social factors being taken into account (para 4.9 of [21]). It should be demonstrated that:

- the radiation exposure resulting from a practice is reduced by radiation protection measures to levels such that further expenditure on design, construction and operation would not be warranted by the corresponding reduction in radiation exposure; and

- the design takes into account issues such as avoiding the need for the workers to stay in areas exposed to radiation for long periods of time.

3.52. The design of the plant should itself be such that situations where higher operator doses might occur are reduced to an acceptable level based on appropriate national standards. In addition, the ALARA principle should be applied during operation to reduce the occupational exposure wherever practicable. This section may refer to other sections of the SAR that address in detail the matter of Radiation Protection.

Conformance with the design principles and criteria

3.53. This section should provide a brief but complete summary of the conformance of the plant design with the finalised design principles and criteria which themselves reflect the chosen safety objectives for the plant.

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

Classification of Structures, Systems, and Components

3.55. This section should include information on the approach used for categorisation and safety classification of structures, systems, and components, and on the methods used to ensure that they are suitable for the relevant design duty, remain fit for purpose and continue to perform any required safety function claimed in the design justification (in particular those claimed in the safety analyses and presented in the corresponding chapter of the SAR). Where there is a potential for structures or systems to interact, then details should be provided here of how it has been ensured within the design, that a lower class or category plant provision can not unduly impair the role of those with a higher classification. A list of safety relevant systems and main structure and components, with their classification and categorisation should be included as an annex or be referenced here. Guidance on options for the classification of structures, systems and components is provided in IAEA guide [4].

Civil Works and Structures

3.56. This section should present the relevant information about the design of civil works and structures as described at the beginning of this chapter. It should include a discussion on the design principles and criteria, codes and standards used in the design and briefly review the way in which 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 made to the requirements for the design should be clearly stated together with the way in which they have been addressed and justified.

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

- the range of anticipated structural loadings together with the defined duty of the buildings and structures and the consideration given to hazards in the design.

- a description of the extent to which load source interactions have been included with confirmation of the buildings' and structures' ability to withstand required load combinations whilst preserving safety function.

- where a safety and/or seismic classification system for buildings and structures has been used, the basis of the classification should be discussed for the design option outlined. It should be demonstrated that the safety classification of buildings enclosing equipment important to safety is commensurate with the classification of the components, equipment and systems that it contains.

- where a building structure or wall is to provide a separate function to its structural role, the additional requirements identified for this function should be briefly described and reference made to other sections of the SAR where appropriate e.g. radiation shielding, separation and containment.

Containment/Confinment Buildings

3.58. This subsection should present a description of the safety requirements for the containment building itself, including its leak-tightness, mechanical strength, pressure resistance and resistance to hazards. It should also describe the main design features of the building provided to comply with the relevant safety requirements. Where the design incorporates a secondary containment, this too should 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 remaining fit for purpose, when subjected to the identified range of individual or combined environmental challenges, throughout the lifetime of the plant. Where 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 plant disruptive influences, including internal and external hazard based events. A complete list of equipment with their environmental qualification should be included as an annex or be referenced here. Guidance on options for qualifying structures, systems and components, including the consideration of environmental factors, is provided in the IAEA safety guide on Seismic Design and Components Qualification [22].

Human Factors Engineering

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

3.61. This section should include a description of the human factors engineering principles used for taking into account all human performance shaping factors, that might have an impact on the reliability of operators' performance. The specific design features of systems and equipment which are aiming to promote successful operator actions, however should be addressed in “Plant system description and design conformance” chapter of SAR.

Protection Against Internal and External Hazards

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

VI - PLANT SYSTEM DESCRIPTION AND DESIGN CONFORMANCE

3.63. The information to be presented in this chapter of the SAR will inevitably be influenced by the particular type and design of reactor selected for construction. Therefore, for some types of reactor many of the sections discussed in this part of the guide will be entirely relevant, while for others they may not apply directly. For these later cases, it must be agreed between the operator and the appropriate regulatory body, which of the plant systems should be described in the SAR. However, as a general rule, all systems which have the potential to affect safety should be described in the SAR, and for these the following suggested general approach may be.

3.64. This chapter should contain a description of all plant structures, systems and components which may affect safety and a demonstration of their conformance with the design requirements. The level of detail of each system description should be commensurate with the safety importance of the system described.

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

- *System Description* – containing the functional requirements and detailed description of the system;

- *Engineering Evaluation* – containing a demonstration that all relevant functional requirements, industrial codes and standard requirements, and regulatory requirements have been addressed adequately. For safety-related systems, this demonstration is supported by the single failure assessments, failure modes and effects analysis, common cause/mode failure assessment, overall reliability assessment, and radiological assessment where applicable, with appropriate reference to more detailed documentation provided as considered necessary;

- *Safety Assessment* – For safety-related systems this subsection would contain a summary statement that the system has sufficient capacity to accomplish its safety function and that there are no credible single failures that can defeat performance of the safety function for which the system was designed. For non safety-related systems this subsection would contain a demonstration that the system is sufficiently separated and/or isolated from safety-related systems to preclude the possibility of impacting their performance.

3.66. As a minimum each *system description* subsection should contain the following information:

- The objective of the system, its safety, seismic, environmental and QA classification and how the system relates to the entire plant, including the degree of similarity to systems previously reviewed and approved by the regulatory body on similar units where appropriate;

- Functional Design Description of the system – including: functional requirements (postulated demands and required performance for all modes of plant operation); clarification on whether the system is normally in continuous, intermittent, or standby operation, specific requirements imposed by regulations, codes, and standards and dealing with: system reliability requirements, redundancy, interfaces with other systems, arrangements of electrical power supplies (instrumentation and control systems); specific requirements, if any, identified from Probabilistic

Safety Analyses; requirements resulting from operational feedback; main elements and their configuration; and simplified functional drawings;

- Human Factors considerations employed in the design including: human factors' considerations in the human-machine interfaces for normal start-up and shut-down and accident modes of operation; instrumentation provided to monitor system operation, physical control board layout of such instrumentation, physical location (accessibility) of equipment requiring testing, maintenance, surveillance, displays, alarms, physical interlocks, bypassed and inoperable status indication;

- Operational aspects – including interdependence with the operation of other systems, technical specification requirements regarding system operability, system testing provisions, system surveillance requirements, and system maintenance requirements;

- Detailed elements of the system design– including: main electrical single line diagram 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 (fluid systems), physical protection devices against internal and external hazards such as: water-tight seals, missile shields, insulation for high temperatures, electrical protection for short-to-ground or short-to-power faults (electrical, instrumentation, and control systems), voltage and frequency protection for electrical buses powering large rotating equipment, interfaces with support systems providing cooling, lubrication, fluid chemistry sampling, air cooling, and fire protection;

3.67. As a minimum each *engineering evaluation* subsection should contain the following information:

- Identification in table form of the specific technical requirements, industrial codes and standards requirements, regulatory requirements – and a demonstration how each of these requirements has been achieved by the system design.

- Summaries of supporting technical information (with references back to the original topical reports) to demonstrate compliance with technical, industrial code and standard, and regulatory requirements. Examples would include: summaries of materials strength and/or corrosion resistance reports, summaries of environmental qualification reports, summaries of flammability tests, summaries of seismic structural analyses, summaries of EMI/RFI interference tests, summaries of independent verification and validation analyses of software, etc.

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

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

- A demonstration that the physical separation, electrical and/or fluid isolation devices, 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/external floods, high winds and internally generated missiles;

- A single failure analysis documented in a Failure Modes and Effects Analysis consistent with the requirement of meeting the single failure criterion contained in the reference documents [5];

- A reliability and common cause failure analysis demonstrating that the system reliability is commensurate with the safety function of the system.

3.69. As a minimum each *safety assessment* subsection should contain a statement summarizing the technical bases by which the system in question is judged to be acceptably safe. This judgement should be based on a combination of: demonstrated compliance with all applicable regulatory criteria (via use of regulatory guidance documents and industrial codes and standards) and/or demonstration of the existence of sufficient design margins by analysis or testing. For non safety-related systems it is sufficient to demonstrate that a failure of the system in question cannot initiate an event more severe than already considered in the safety analysis or degrade the operation of safety-related systems.

3.70. The general points described above may need to be supplemented by more detailed information relating to the specific features or functions to be completed by each particular system. The information given in the following sections refers to these specific topics for each of the listed systems and may need to be adapted to reflect the particular plant type design.

Reactor

3.71. This section should present the relevant information about the reactor, where possible in a format as described at the beginning of this chapter of the SAR. In addition, the following information should be presented to demonstrate the capability of the reactor to perform its safety functions throughout its intended lifetime under all operational modes:

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

Fuel System Design

- description of the main fuel system elements with a safety substantiation for the selected design bases. The fuel system design bases justification should include, amongst other, a description of the fuel design limits and functional characteristics in terms of desired performance under stated conditions including normal operation, anticipated operational occurrences and accident conditions.

Reactor internals design

- a description of the reactor internals system defined as the general external details of the fuel, the structures into which the fuel has been assembled (for example fuel assembly, fuel bundle), related components required for fuel positioning and all supporting elements internal to the reactor, including if relevant separate provisions for moderation and fuel location. Reference to the other sections of the SAR which cover related aspects of the reactor fuel and also fuel handling and storage should be made;

- a description of the thermo-hydraulic, chemical, physical, structural and mechanical properties of the components including the expected response to static and dynamic mechanical loads, behaviour and the effects of irradiation on the ability of the reactor internals to adequately perform their safety functions over the lifetime of the plant;

- a description of any significant sub-system components including separate provisions for moderation and fuel location with corresponding design drawings and a consideration of the effects of service on performance of safety functions, and including reactor internals surveillance / inspection programs to monitor the effect of irradiation and ageing on the internal components;

- a description of the programme to monitor the behaviour and performance of the core, which should address provisions to monitor the core neutronics, dimensions and temperatures.

Nuclear design and core nuclear performance

- the nuclear design bases, including nuclear and reactivity control limits such as excess reactivity, fuel burn-up, reactivity coefficients, power distribution control, and reactivity insertion rates;

- a description of the nuclear characteristics of the lattice, including, core physics parameters, fuel enrichment distributions, burnable poison distributions, burn-up distributions, and refuelling schemes;

- a description of the analytical tools, methods, and computer codes (along with code verification and validation information and uncertainties) used to calculate the neutronics characteristics of the cores, including reactivity control characteristics;

- a description of the design basis power distributions within fuel elements, fuel assemblies, and the core as a whole; providing information on both axial and radial power distributions and overall reactivity control capability;

- a discussion of neutronics stability of the core throughout the fuel cycle, considering the possible normal and design basis operating conditions of the plant.

Thermal and hydraulic design

- the design bases, the description of thermal and hydraulic design for the reactor core and attendant structures, the interface requirements for the thermal and hydraulic design of the reactor coolant system;

- a description of analytical tools, methods, and computer codes (along with code verification and validation information and uncertainties) used to calculate thermal and hydraulic parameters;

- a description of flow, pressure and temperature distributions, with the identification of limiting values and their comparison with design limits;

- justification of core thermal and hydraulic stability.

Reactor materials

- justification of the materials used for the components of the reactor, including the primary pressure boundary materials, the materials providing a core support function and any separate moderation function. Information should also be provided on the materials 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.

Functional design of reactivity control systems

- Information justifying 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 information on matters to be included in this section of the SAR may be found in the reference documents [23,24].

Reactor Coolant and Associated Systems

3.73. This section should present the relevant information about the reactor coolant system and associated systems, where possible in a format as described at the beginning of this chapter. In addition, the following information should be presented to demonstrate that the reactor coolant system (RCS) will retain its required level of structural integrity under conditions imposed by reactor service under both operational states and accident conditions:

Integrity of reactor coolant pressure boundary

- description and justification of the results of the detailed analytical and numerical stress evaluations, engineering mechanics, and fracture mechanics studies of all components comprising the reactor coolant pressure boundary subjected to normal, including shut down conditions, and postulated accident loads. A list of all components should be provided with the corresponding applicable codes. The specific detailed stress analyses for each of the major components should be directly referenced to allow further evaluations if needed.

Reactor vessel

- Information detailed enough to demonstrate that the materials, fabrication methods and inspection techniques, load combinations used conform to all applicable regulations, industrial codes and standards. This concerns the reactor vessel materials, the pressure-temperature limits and the reactor vessel integrity including embrittlement considerations. Where the reactor design includes pre-stressed concrete components or vessel calandria these too need to be considered.

Component and subsystem design

- description and justification of the performance and design features implemented to ensure that the various components within the reactor coolant system and subsystems interfacing with the reactor coolant system meet the design safety requirements. This should include the reactor coolant pumps, gas circulators if present, the steam generators or boilers, the reactor coolant piping or ducting, the main steam line isolation system, the reactor core isolation cooling system, the main steam line and feedwater piping, the pressuriser, the pressuriser relief discharge system, the emergency cooling provisions, the residual heat removal system, the primary and secondary systems under pressure including all components such as pumps, valves and supports etc.

3.74. Further information on matters to be included in this section of the SAR may be found in the reference documents [25].

Engineered Safety Features

3.75. This section should present the relevant information about the engineered safety features and associated systems as described at the beginning of this chapter. Where necessary additional system specific information should be added as suggested bellow.

Emergency core cooling system

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

Containment (or Confinement) systems

3.77. This subsection should present the relevant information about the containment (or confinement) systems, included to localise the effects of accidents, and including amongst other things: the containment heat removal systems, the secondary containment functional design, the containment isolation system, the containment over and under pressure protection where provided, the combustible gas control in containment, the containment spray system and the containment leakage testing. Further information on matters to be included in this section of the SAR may be found in the reference documents [26].

Habitability systems

3.78. This subsection should present the relevant information about the habitability systems. The term habitability systems refers to the engineered safety features systems, equipment, supplies and procedures provided to ensure that essential plant personnel can remain in their positions, including in the main and supplementary control rooms, and take actions to operate the nuclear power unit safely under normal 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.

Fission product removal and control systems

3.79. This subsection should present the relevant information about the fission product removal and control systems. In addition the following specific information should be presented to demonstrate the performance capability of the fission product removal and control systems: a consideration of the coolant pH and chemical conditioning during all necessary conditions of system operation, the effects of postulated design bases fission product heat loads on filters; and the effects of design basis fission product release mechanisms on filter operability.

Other engineered safety features

3.80. This subsection should present the relevant information about any other engineered safety features implemented in the plant design as described at the beginning of this chapter. Some examples include, but are not limited to: the auxiliary feed-water system, and back-up cooling systems, etc. The list of these systems will depend very much on the type of the plant under consideration.

Instrumentation and Control

3.81. This section should present the relevant information about the instrumentation and control systems as described at the beginning of this chapter. The reactor instrumentation senses the various reactor parameters and transmits appropriate signals to the control systems during normal operation and to the reactor trip and engineered-safety-feature systems during anticipated operational occurrences and accident conditions. The information provided in this chapter should emphasise those instruments and associated equipment that constitute the protection systems, and those systems relied upon by operators to monitor plant conditions and to transfer and maintain the plant in a safe shutdown state after a postulated design basis accident. Information should also be provided on non-safety related instrumentation and control systems used to control the plant during normal operations. 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 information on matters to be included in this section of the SAR may be found in the reference document [27].

Protection systems

Reactor Trip System

3.82. This subsection should present the relevant information about the reactor trip systems as described at the beginning of this chapter. In addition the following specific information should be presented that is unique to the reactor trip system:

- the design bases for each of all individual reactor trip parameters with reference to the postulated initiating events (PIE) which the trip parameter is credited for mitigating;

- the identification of reactor trip system set-points, time delays in system operation, measurement uncertainties, and how these relate to assumptions made in the Safety Analysis Report;

- any interfaces with the engineered safety feature actuation system (including use of shared signals and parameter measurement channels);

- any interfaces with non-safety related instrumentation, control, or display systems along with provisions to assure independence;

- the means employed to assure separation of redundant reactor trip system channels and the means by which coincidence signals are generated from redundant independent channels;

- Provisions for manual actuation of the reactor trip system from both the main control room and the supplementary control room should be described;

- Where reactor trip logic is implemented using digital computers, the system description should include a discussion of the software design and quality assurance programs, and software verification and validation program. Further information on matters to be included in this section of the SAR may be found in the reference document [27].

Engineered safety feature actuation systems

3.83. This subsection should present the relevant information about the engineered safety features actuation systems as described at the beginning of this chapter. In some plant designs the reactor trip and engineered safety features actuation system are designed as a single system. In this case it is appropriate to have a single section describing the reactor trip and engineered safety features actuation as one system.

3.84 In addition the following specific information should be presented that is unique to the engineered safety features actuation system:

- the design bases for each of the individual engineered safety feature actuation system parameter with reference to the PIE which the parameter is credited for mitigating; interfaces with the reactor trip system (including use of shared signals and parameter measurement channels); interfaces with non-safety related systems along with provisions to assure proper electrical signal isolation; the means employed to assure separation of redundant engineered safety feature actuation system channels;

- where engineered safety features actuation logic is implemented using digital computers, the system description should include a discussion of the software design and quality assurance programs, and software verification and validation program. Further information on matters to be included in this section of the SAR may be found in the reference document [27];

- the identification of engineered safety features actuation system set-points, time delays in system operation, measurement uncertainties, and how these relate to assumptions made in the Safety Analysis chapter;

- provisions for equipment protective interlocks (eg. pump and valve interlocks) within the engineered safety feature actuation system should be described and a demonstration provided that such interlocks will not adversely impact engineered safety feature operation;

- provisions for manually initiating engineered safety features from the main control room and supplementary control room;

- any relevant remote operator/ automatic control, local control, on-off control, or modulating control, envisaged in the design and credited in the safety analysis.

- automatic and manually operated controls for safe shutdown systems that assure reactivity control, primary and secondary coolant system makeup, and decay heat removal from the core and containment. These systems may be used to permit both safe shutdown in the short term and hold-down in the longer term.

Safety-related display instrumentation

3.85. This section should present the relevant information about the safety-related display instrumentation systems and the computerized plant information system as described at the beginning of this chapter. In addition the following specific information should also be presented:

- A list of parameters which are measured and the physical location of the sensors and environmental qualification envelope;

- An identification of the parameters monitored by the plant computer and the characteristics of any computer software (scan frequency, parameter validation, cross-channel sensor checking, etc.) utilized for filtering, trending, generation of alarms, long-term storage of data, and displays available to the operators in the control room and supplementary control room. If data processing and storage is performed by multiple computers, the means to achieve synchronization of the different computers systems should be addressed.

All other instrumentation systems required for safety

3.86. This section should present the relevant information about any other diagnostic and instrumentation systems required for safety as described at the beginning of this chapter and should cover: any particular system needed for management of severe accidents, leak detection systems, vibration and loose parts monitoring systems, protective interlock systems which are credited in the safety analysis to prevent damage to safety related equipment to prevent certain types of accidents (examples include: valve interlocks at low pressure – high pressure fluid system interfaces that if operated could result in inter-system loss of coolant accidents), etc.

Control systems not required for safety

3.87. This section should present the relevant information about the control systems not required for safety in a format similar to the one described at the beginning of this chapter. In addition the following specific information should also be presented to demonstrate that postulated failures of control systems will not defeat the operation of safety related systems, or result in scenarios more severe than already postulated and analysed in the safety analysis:

- a brief description of non-safety related control systems used for normal plant operations;

- a description of any non-safety related limitation systems (control grade power reduction systems installed to avoid reactor trip by initiating a partial power reduction);

- A demonstration that such systems do not challenge the operation of safety related systems.

Main Control Room

3.88. This section should present a description of the main control room layout with an emphasis on human-machine interface. It is recognised that the electrical design standards for equipment located in the main control room are already described in previous sections and need not be repeated here. If a formal control room design review (human factors review) has been performed in developing or upgrading the layout, the results of this review should be summarised in this section.

Supplementary Control Room

3.89. This section should present an appropriate description of the supplementary control room, including layout, with an emphasis on human-machine interface. It is recognised that the electrical design standards for equipment signals routed to the supplementary control room are already described in previous sections and need not be repeated here. The means of physical and electrical isolation between the plant system and communication signals routed to the main control room and supplementary control room should be described in detail to demonstrate that the supplementary control room is redundant and independent from the main control room. The mechanisms for transfer of control and communications from the main control room to the supplementary control room should be described in detail to demonstrate how this would be done under accident conditions.

Electrical Systems

3.90. This section should present the relevant information about the electrical power systems as described at the beginning of this chapter. In addition the following specific to electrical systems information should also be presented:

- the divisions of electrical power systems, including the differing system voltages and which parts of the system are considered to be essential.

- substantiation of the functional adequacy of the safety-related electric power systems including breakers, and ensuring that these systems have adequate redundancy, physical separation, independence and testability in conformance with the design criteria. Further information on matters to be included in the report on this subject may be found in the reference document [28].

- a general description of the utility grid and its interconnection to other grids and the connection point to the on-site electric 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 along with communications provisions between the dispatch centre, remote major load centres, and generating plants. The principal means of external grid voltage and frequency regulation should be described. A simplified one-line drawing showing the main grid interconnections should be provided.

Off-site power system

3.91. This section should present the relevant information about the off-site electric power systems as described at the beginning of this chapter. This section should include a description of the off-site power system with emphasis on control and protective features (breaker arrangements, manual and automatic disconnect switches) at the interconnection to the on-site power system. Special emphasis should be provided on all design provisions used to protect the plant from off-site electrical disturbances and maintain power supply to in-plant auxiliaries. Information on grid reliability should also be provided and if needed, the design specific provisions to cope with frequent grid failures should also be described.

On-site power systems

A.C. Power System

3.92. This subsection should present the relevant information about the A.C. power system as described at the beginning of this chapter. This section should contain description of the on-site A. C. power systems including the diesel, or gas turbine driven, generator configuration and non-interruptible A.C. power system. The power requirements for each plant A.C. load should be identified, including - steady-state load; start-up kVA for motor loads; nominal voltage; allowable voltage drop (to achieve full functional capability within required time period); sequence and time to achieve full functional capability for each load; nominal frequency; allowable frequency fluctuation; number of trains, and minimum number of ESF trains to be energised simultaneously.

3.93. In addition relevant on-site A.C. power system information should also be presented to demonstrate that:

- during a design basis accident with subsequent loss of off-site power the required ESF loads can be sequenced on to the emergency diesel generators without overloading the diesel generators and in time frames consistent with the assumptions contained in the safety analysis chapter.

- on-site A.C. power system breakers are co-ordinated to assure reliable delivery of emergency power to ESF and non-interruptible A.C. power system loads

- non-interruptible A.C. power to essential safety and safety-related instrumentation and control systems is continuously provided during normal off-site A.C. power system availability and during postulated loss of off-site power events.
- the maximum frequency decay rate and limiting under-frequency value for reactor coolant pump coast-down, and minimum number of ESF trains to be energised simultaneously (if more than two trains provided) should be provided.

D.C. power systems

3.94. This subsection should present the relevant information about the D.C. power system as described at the beginning of this chapter. In addition the following specific D.C. power system information should also be presented: an evaluation of long-term battery discharge capacity (projected voltage decay as a function of time without charging when subjected to design loads), the major DC loads present (including the non-interruptible A.C. power system inverters, and any non-safety related D.C. loads such as turbine bearing lubrication oil pumps), and the description of the fire protection for D.C. battery vault area and cable systems.

3.95. The power requirements for each plant D.C. load should be identified, including - steady-state load; surge loads (including emergency conditions); load sequence; nominal voltage; allowable voltage drop (to achieve full functional capability within required time period); number of trains, and minimum number of ESF trains to be energised simultaneously (if more than two trains provided).

3.96. Further information on matters to be included in the report on on-site and off-site power systems may be found in the reference document [28].

Plant auxiliary systems

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

Water systems

3.98. This subsection should present the relevant information about the water systems associated with the plant as described at the beginning of this chapter. It should include the station service water system, the cooling system for reactor auxiliaries, the de-mineralised water makeup system, the ultimate heat sink, the condensate storage facilities, etc.

Process auxiliaries

3.99. This subsection should present the relevant information about the auxiliary systems associated with the reactor process system in a format as described at the beginning of this chapter. It should include for example information on: the compressed air systems, the process and post-accident sampling systems, the equipment and floor drainage system, the chemical and volume control system, the purification system and the system for controlling the usage of boric acid, etc.

Heating, ventilation and air conditioning systems (HVAC)

3.100. This subsection should present the relevant information about the heating, ventilation, air conditioning, and cooling systems in a format as described at the beginning of this chapter. It should include the control room area ventilation system, spent fuel pool area ventilation system, auxiliary and radioactive waste area ventilation system, turbine building area ventilation system (in BWRs) and engineered-safety-feature ventilation systems.

Other auxiliary systems

3.101. This subsection should present the relevant information about any other plant auxiliary system which operation may influence plant safety and which has not been cover in any other part of the SAR. For example: the communication systems, the lighting systems, the diesel generator cooling water system, the diesel generator starting system, the diesel generator lubrication system and the diesel generator combustion air intake and exhaust system, etc..

Power Conversion Systems

3.102. This section should present the relevant information about the plant power conversion system which is plant type/design dependent.

3.103. The following information specific to steam and power conversion systems should also be provided where appropriate:

- the performance requirements under normal operational states and accident conditions for the turbine generator(s).

- description of the main steam line piping and 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. The water chemistry program should be described here along with a discussion of the steam, feedwater, and condenser system materials.

3.104. For other types of power conversion systems equivalent alternative information should be provided to demonstrate the systems conformance with the relevant design requirements.

Fire protection systems

3.105. This section should present the relevant information about the fire protection systems as described at the beginning of this chapter. It should justify the provisions made to ensure that the plant design has adequate fire safety. This should include adequate provisions for defence in depth in the event of a fire, which should address fire prevention, fire detection, fire warning, fire suppression and fire containment. Consideration should be given to the selection of materials, physical separation of redundant systems, equipment seismic qualification and the use of barriers to segregate redundant trains.

3.106. The extent to which the design has been successful in providing adequate fire resistance should be assessed and this section may refer to other sections of the SAR for this information (e.g. safety analyses chapter). Where appropriate the provisions to ensure personnel fire safety may also be described in this section Further information on matters to be included in this section of the SAR may be found in the reference documents [14].

Fuel Handling and Storage Systems

3.107. This section should present the relevant information about the fuel handling and storage systems as described at the beginning of this chapter. It should include the details of the proposed arrangements for shielding, handling, storing, cooling, transfer and transporting nuclear fuel. Further information on matters to be included in this section of the SAR may be found in the reference documents [29].

Non-irradiated Fuel

3.108. This subsection should present the relevant information about the fuel handling and storage systems used for non-irradiated fuel as described at the beginning of this chapter. It should include the measures proposed to ensure that non-irradiated fuel is maintained in a safe condition at all times. This should include considerations such as packaging, accounting systems, storage, criticality prevention, fuel integrity and fuel security.

Irradiated Fuel

3.109. This subsection should present the relevant information about the fuel handling and storage systems used for irradiated fuel as described at the beginning of this chapter. It should include the measures proposed to ensure that irradiated fuel is maintained in a safe condition at all times, this should include considerations such as appropriate radiological protection provisions, criticality prevention, fuel integrity including the potential for leakage, fuel chemistry, cooling, fuel accounting, fuel security and fuel consignment and transport arrangements.

Radioactive Waste Treatment System

3.110. This section should present the relevant information about the radioactive waste treatment systems as described at the beginning of this chapter. It should include the design features of the plant which safely control, collect, handle, process, store and dispose of solid, liquid and gaseous forms of radioactive wastes arising from all activities on the site throughout the plant life cycle. This should include the structures, systems and components provided for these purposes, and also the instrumentation incorporated to monitor potential leaks or escapes of active waste material. The potential for waste materials to be adsorbed/absorbed should be considered when deciding on the measures necessary to address this hazard. Further information on matters to be included in this section of the SAR may be found in the reference documents [30].

3.111. This section should include a description of the sources of radioactivity that have been included in the design requirements for the active waste provisions and if needed reference to the chapter of the SAR that addresses decommissioning should be made.

3.112. This section may need to cross reference the section of the SAR that considers radiological protection issues on the plant. This section might also refer to other sections of the SAR where 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, might assist or support a safety system or might influence the performance of a safety system should be described here.

VII - SAFETY ANALYSES

3.114. This chapter should describe the results of the safety analyses performed to assess the safety of a NPP in response to postulated initiating events against safety criteria and radiological release limits. These include deterministic safety analyses used to support normal operation, analyses of anticipated operational occurrences, design basis events and beyond design basis events, selected severe accidents and PSA. The description may be supported by reference material where necessary. Additional guidance on the analyses to be performed by the designers and operators, in the support of the plant licensing process, is provided in chapter 4 of the IAEA Safety Guide on Safety Assessment and Verification for NPPs NS-G-1.2[4]. This guide should be considered as a reference in the preparation of this chapter of the SAR.

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

3.116. The information presented in the safety analysis chapter should be enough 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 prescribed and acceptable limits for radiation doses and releases for each plant condition category. The design, manufacture, construction, and commissioning should be integrated with the safety analysis to ensure that the design intent has been incorporated into the as-built plant.

Safety objectives and acceptance criteria

3.117. This section should refer to the nuclear safety, radiation protection and technical safety objectives and principles, applicable to the particular plant design as previously identified in chapter "Safety objectives and design principles".

3.118. In addition, detailed acceptance criteria, specific to structures, systems and components for different classes of events and type of analyses, should be identified.

These acceptance criteria should be such that frequent events should have minor consequences and that events that may result in severe consequences should be of very low probability.

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

Postulated Initiating Event identification and classification

3.120. The methodology used to identify Postulated Initiating Events[2] (PIEs) should be described. This may include among others the use of master logic diagrams, fault trees, hazard analysis, failure mode and effects analysis (FMEA), comparison with list of PIEs for similar plants and analysis of operating experience. Initiating events which can occur due to human errors should also be considered in the identification of the PIEs. Whatever method is used, it should be demonstrated that the PIE identification has been performed in a systematic way and resulted in the development of a comprehensive list of events.

3.121. Events should be classified in accordance with their anticipated frequency and type. The purpose of this classification is to:

- justify the basis for the spectrum of events under consideration;

- reduce the number of initiating events requiring detailed analysis to a set which exercises the most bounding cases in each of the various event groups credited in the safety analysis, but does not contain events with identical system performance (such as timing, plant systems response, and radiological release fractions).

[2] A PIE is defined in Ref [5] as an ”identified event that leads to anticipated operational occurrences or accident conditions”. PIEs include events such as equipment failure, human errors and human induced or natural events.

- allow for differing analysis acceptance criteria to be applied to differing event classes.

3.122. The basis for event classification should be described and justified. Typically the list of the postulated initiating events to be addressed in the SAR will cover anticipated operational occurrences and design basis accidents. It is recommended also to include results from the beyond design basis accident analysis performed [4, 5]. Some of the design basis or beyond design basis accidents may further develop, if additional faults are assumed, and lead to so called “severe accidents” involving significant core degradation and/or off-site radioactivity release. The results from severe accident analyses should also be included in the SAR as far as they are needed to develop the plant accident management programme and to support emergency preparedness. [4,5]. (The terminology used for the PIE types mentioned above is defined in [4].)

3.123. This process should lead to a list of different classes of plant specific events to be analyzed which considers all types of initiators, both internal and external to the plant, and all modes of operations, including normal operation, shutdown and refuelling. 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 total loss of off-site power should also be evaluated, taking into account possible interactions between plant manoeuvers and the grid and, when appropriate, possible interactions between different reactor units on the same site. Failures in other plant systems such as the irradiated fuel storage and radioactive gas storage tanks 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 postulated initiating events, such as: increase or decrease of heat removal; increase or decrease in reactor coolant flow; reactivity and power anomalies (including mis-positioning of a fuel bundle); increase or decrease in reactor coolant inventory; release of radioactive material from a subsystem or component. In addition, a set of internal PIEs derived from other considerations should be taken into account, such as the loss of support systems, internal floods, fires and explosions, internally generated missiles, collapse of structures and falling objects, pipe whip and jet effect, false containment isolation signals leading to loss of primary pump cooling, etc [15].

3.125. The set of external postulated events to be considered should include, where appropriate: fires; floods; earthquake; volcanism, extreme winds, extreme weather conditions; biological phenomena, human induced events such as aircraft crash, explosions, toxic or asphyxiant gases corrosive gas and liquids, electromagnetic interference, damage of water intakes [13], effects of nearby industrial plants transportation system explosions, etc..

3.126. The recommendations on PIE identification and classification included in reference [4] should be considered in the preparation of this section of the SAR.

Human actions

3.127. This section should present and justify in general the approaches used to consider human action in the different types of the safety analyses and the methodology selected to model these actions in each type of the analyses.

Deterministic Analyses

3.128. This part of the SAR should addressed all deterministic analyses performed to evaluate and justify plant safety. Deterministic safety analysis predicts the NPP response from specific pre-determined operational states to postulated initiating events. It applies specific rules and uses specific acceptance criteria. The analyses typically focus on neutronics, thermal-hydraulic, structural and radiological aspects that are often analysed with different computational tools. As it 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”.

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

3.130. A general summary of the validation and verification 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 computer program supporting documentation. Emphasis should be given to the substantiation of the applicability of the computer program to the particular event, and reference 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 information on matters to be included in the SAR on this subject may be found in the reference documents [4].

3.131. Any general analysis guidelines (such as but not limited to: choice of operating states of systems and/or support systems, conservative time delays, operator actions) used in setting up the deterministic safety analyses methods and models used to demonstrate acceptability should be described in this section. Guidance on performance of deterministic accident analyses for PWR, BWR, CANDU and RBMK type of reactors is provided in the IAEA Safety Report on Accident Analysis of Nuclear Power Plants [31].

Safety in Normal Operation

3.132. This section should demonstrate that the normal operation of the plant can be carried out safely and hence confirm that radiation doses to workers and members of the public are within acceptable limits and planned radioactive discharges/releases from the plant are within the acceptable limits [4].

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

- normal reactor start-up from shutdown through criticality to full power;
- power operation including full power and low power;
- changes in reactor power, including load follow modes if applicable;
- reactor shutdown from power operation;
- hot shutdown;
- cooldown process;
- refuelling during normal operation, for plants which do so;
- shutdown in a refuelling mode or other maintenance condition which opens the reactor coolant or containment boundary;
- fresh and irradiated fuel handling.

Anticipated Operational Occurrences and Design Basis Accidents

3.134. This section should describe the results from the analyses of the 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. This should be done by carrying out a conservative analysis which should take account of the uncertainties in the modelling. The analyses should cover all normal operational states, including low power and shut down regimes.

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 that are important to the outcome of the safety analysis should be identified. These would typically include reactor power and distribution, primary system pressures, temperatures and flows, reactivity coefficients, reactor kinetics parameters, and the worth of reactivity devices.

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

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

Analysis of Individual PIEs

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

- *Postulated Initiating Event* – a description of the PIE, the class to which the PIE belongs and the acceptance criteria or safety goals which are to be met.
- *Accident Boundary Conditions* - a detailed description of the plant operating configuration prior to the occurrence of the PIE, the model and event specific assumptions, and the computer model (and version number) used. System and operator actions which are credited in the analysis including:

    - normal operating plant instrumentation and controls;
    - plant and reactor protection systems;
    - normally operating plant systems and support systems;
    - engineered safety systems and their actuation set points; and
    - operator’s action, if any.

- *Initial Plant State* – Specific values of important plant parameters and initial conditions used in the analysis, this may be presented in table format. There should be an explanation of how these values have been chosen and the degree to which these are conservative for the specific PIE being analyzed.

- *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 it being selected as the limiting single failure [31].

- *Plant Response Assessment* – A discussion of the modelled plant behaviour, highlighting the timing of the main events (initial event, any subsequent failures, time at which various safety groups are actuated, and time at which a safe long term stable state is reached). Individual system actuation times and operator intervention should be included. 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 progression can be described within the context of the acceptance criterion that is being considered. For example, in evaluating fuel cladding temperatures, parameters such as power, heat flux, reactor coolant system pressure, reactor coolant system fluid inventories, fuel temperatures, ECCS flow rates should be given, where appropriate to the type and design of 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. It should be demonstrated that sufficient safety margins exist.

- *Radiological Dose Assessment* – Where applicable, results of the radiological consequences assessment should be presented. A discussion of the key results should be compared with the acceptance criteria or safety goals and conclusions on meeting the acceptance criteria should be clearly stated.

- *Sensitivity Studies and Uncertainty Analyses* – When appropriate, the results from sensitivity and uncertainty analyses, performed to demonstrate the robustness of the results and conclusions of the accident analyses, should be presented.

Consideration of Design Capability for Beyond Design Basis Accidents

3.139. In addition to the design basis events, analysis should also be performed which demonstrates the capability of the design to mitigate certain beyond design basis events. The choice of the events of this class to be analysed may be decided partly from national regulations, a PSA, or from any other fault analysis that identifies potential plant vulnerabilities. Events which may typically fall into this category are sequences involving more than one single failure (unless they are taken into account in the DBA at the design stage as for some new designs) such as: Station AC Blackout, ATWS, design basis events with degraded performance of the protection system or engineered safeguards systems, sequences that lead to containment or confinement bypass. The basis for the selection of events should be described and justified in this section.

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

3.141. The format and content of the analyses of Beyond Design Bases Events to be presented in this part of the SAR should be consistent with the presentation of the analyses for Anticipated Operational Occurrences and Design Bases Events with the following modifications:

- the objective of the beyond design bases events analysis and the specific safety goals or acceptance criteria should be stated;
- a discussion of the additional postulated failures in the accident scenario should be provided along with a discussion of the bases for their selection;
- 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, procedures to follow and have been trained. The key results should be compared with the specific acceptance criteria or safety goals and conclusions on meeting the acceptance criteria should be clearly stated.

Severe Accidents

3.142. Where required, this section should describe in sufficient detail the analysis performed to identify accidents which can lead to significant core damage and/or off-site radiological releases (severe accidents). The challenges to the plant that these events represent and the extent to which the design might be reasonably expected to mitigate these faults should be considered and referenced here.

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

Probabilistic Analyses

3.144. An integrated review of plant design and operation 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 Probabilistic Safety Assessment (PSA). This section should briefly describe the scope of the PSA study, the methodology used and the results obtained. If any quantitative probabilistic safety criteria or goals have been used in the development of plant design (as mentioned in section on ”Probabilistic design criteria” in SAR), those should also be referred to here.

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

- justification of the selected scope for the PSA study;
- accident sequence modeling including: event sequence and system modelling, human performance analysis, dependence analysis, classification of accident sequences into plant damage states;
- data assessment and parameter estimation including assessment of: frequency of initiating events, component reliability, common cause failure probabilities, human errors probabilities ;
- accident sequences quantification including uncertainty, importance and sensitivity analyses;
- source term and consequences assessment.

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 plant design and operation that are the most important contributors to those 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 the criteria should be presented to demonstrate compliance. These criteria may relate to both individual and societal risk measures to ensure that all aspects of assessing risk to the public from the NPP are adequately considered.

Summary of Results of Safety Analysis

3.148. This section should contain a summary of the overall results of the safety analysis, confirming that the requirements of the analysis have been met in every respect, providing justification if requirements have been varied and clearly justifying where requirements have not been met entirely or have been varied as a result of further considerations. In the latter case any compensatory measures taken to meet the safety requirements should be identified.

VIII - COMMISSIONING

3.149. It is necessary for the operator to demonstrate that the plant will be suitable for service prior to it entering the operational phase. The process that the operator has adopted to demonstrate this should be presented here. The operator should clearly describe the tests intended to validate the plant performance against the design, prior to the operation of the plant. The commissioning programme must, among other things, confirm that the separate plant items will perform within their specification and that in the various safety systems they work together to ensure that the system safety functions are reliably performed.

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 part of the Safety Analysis Report. A clear link from the NPP safety justification to the commissioning programme should be demonstrated.

3.151. This chapter should present also the details of the commissioning organisation, including the appropriate interfaces between design, construction and operating organisations during the commissioning period, which should include any provisions for additional personnel and their interaction with the commissioning organisation. 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 outcome from 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. 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 test programme time schedule should be presented, with a clear identification of tests that are considered pre-requisites for other tests. Further information on matters to be included in this section of the SAR may be found in the reference documents [32].

3.153. For older plants, possibly after a revision to the SAR, the section on commissioning may be reduced, with some information transferred to supporting references.

IX - OPERATIONAL ASPECTS

3.154. Depending on the regulatory practice of particular countries, some topics such as operational aspects may be included in 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 that are relevant to safety throughout the plant life cycle, and also present the operators proposals to adequately address the identified issues. Information on matters to be discussed in this chapter of the SAR may be found in the reference documents [8, 33,34,35,36].

Organisation

3.155. This section should present the operating organisational arrangements and identify the functions and responsibilities of the different components within it. The organisation and responsibilities of review bodies/(e.g. safety committees, advisory panels) should also be described. The description of the organisational structure should demonstrate that all the management functions for safe operation of power plant such as policy making functions, operating functions, supporting functions and reviewing functions are adequately addressed.

Administrative procedures

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

Operating procedures (normal and abnormal operation)

3.157. This section should describe the plant operating procedures. The information presented should be enough 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 and they provide instructions for the safe conduct of all modes of normal operation, such as starting up, power production, shutting down, cooldown, shutdown, load changes, process monitoring and fuel handling. It should be explicitly demonstrated that human factors engineering principles have been considered in the development and validation of the procedures [37].

Emergency operating procedures

3.158. This section should describe the procedures, whether event, outcome or symptom oriented, that will be used by the operators in emergency situations. Justification of the selected approach should be provided and where appropriate linked to the findings of the plant safety analyses. Whatever approach is selected it should be demonstrated that the required operator actions to diagnose and to deal with emergency conditions are addressed appropriately. The approach used for validation and verification should be presented, together with a list of the procedures involved. It should be demonstrated that human factors engineering principles have been considered in the development and validation of the procedures.

Accident Management Guidelines

3.159. This section should describe the selected approach to plant accident management. The corresponding severe accident management guidelines developed to mitigate the consequences of a severe accident should be described and justified. The information provided should make reference to the plant accident management programme, if appropriate. It should be demonstrated that all possible means, safety related or conventional, in the plant or from neighbouring units or external, with the aim of preventing the release of radioactive material to the environment, have been considered. It should also be demonstrated that the accident management guidelines have been developed taking into account in a systematic way the results from the severe accidents analysed presented in the SAR section on severe accidents, identified plant's vulnerabilities to such accidents, and strategies selected to deal with these vulnerabilities.

Maintenance, Surveillance, Inspection and Testing

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

3.161. In this section the Safety Analysis Report should describe and justify the arrangements that the operator intends to have in place to identify, control, plan, execute, audit and review maintenance, surveillance, inspection and testing practices which influence reliability and affect nuclear safety.

3.162. The operator must justify the appropriateness of the plant surveillance activities that have been identified as necessary to confirm the continued fitness for purpose of the plant and its retained operability. The surveillance programme should verify that provisions for safe operation that were made in the design and were checked during construction and commissioning continue to exist during the life of the plant and to supply data to be used for assessing the residual life of SS&C. In addition it should be demonstrated that the surveillance programme is adequately specified to ensure the inclusion of all relevant aspects of the operational limits and conditions.

Also, it should be demonstrated that the frequency of the surveillances 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, 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 inspected meets specified standards, satisfies adopted inspection criteria and remains capable of performing its required safety function. In particular the emphasis should be placed on the adequacy of the in-service inspections of the integrity of the primary and secondary coolant systems because of their importance to safety and the possible severity of the consequences of failure.

3.164. The operator should also identify all testing, which can affect the safety function of nuclear plant. This should include, in addition to a schedule of identified testing, a system for ensuring that testing is prompted, carried out and confirmed within the time-scales allowed. This section should also refer to methods for the audit and review of identified testing. Further information on matters to be included in this section of the SAR may be found in the reference documents [35].

Management of ageing

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

Control of modifications

3.166. The operator should describe the proposed method to identify, control, plan, execute, audit, review and document the necessary modifications to the plant throughout its life cycle. This should take account of the safety significance of proposed modifications to allow them to be graded and referred to the regulator where necessary. The modification control process should cover the changes incurred on the plant systems and components, operating limits and conditions, plant procedures and process software. It also should be demonstrated that the modification control covers the permanent and temporary changes at the plant. When a proposed modification impacts on performance of operators or organization, it should be demonstrated that provisions are in place to ensure that the human factor engineering principles are considered and implemented all along the design and implementation of the modifications. Records of all modifications should be retained, and where necessary all changes to documentation, procedures, instructions and drawings should be routinely revised. It should be also demonstrated that the configuration management requirements are observed in the implementation of the plant modifications. Further information on matters to be included in this section of the SAR may be found in the reference documents [33].

3.167. If applicable, this section should also provide a list of all safety relevant modifications that have been implemented since the last safety analysis report (or Periodic Safety Review) stating the reasons of these modifications.

Qualification and training of personnel

3.168. This section should justify that the plant staff qualification and training programme is adequate to acquire and maintain the required level of staff professional competence during the entire plant life cycle. Information should be provided to describe the initial qualification requirements and the staff-training program, including the refresher and retraining and also the applicable documentary system for recording the current position 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 might include a training programme based on analysis of the responsibilities and tasks involved in the work, and should apply to all personnel including management.

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

3.171. Further information on matters to be included in this section of the SAR may be found in the reference documents [34].

Human Factors

3.172. This part of the SAR should describe the operator's proposals for a programme to manage operational issues, which are affected by human factors considerations, 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, assessment of operator fitness for duty and other human factors related issues.

Operational Experience Feedback Programme

3.173. The operator should present proposals for an operational experience feedback program to be implemented. It should provide measures to ensure that plant incidents and events are identified, recorded, notified, investigated internally as appropriate and used to promote improved plant performance and safety culture by adoption of appropriate countermeasures to prevent recurrences, and allow the regulator to be informed where necessary. The program should consider technical, organisational, and human factors aspects. Where available, arrangements made for reporting and analysing of "low level events and near misses" should be described.

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

3.175. This part of the SAR should demonstrate the suitability of the proposed operational experience feedback system for analysing the root causes of equipment failures and human errors, for improving job descriptions and operational procedures and for assessing the need for back-fitting and modernisation of the plant, including organizational changes, if necessary. Further information on matters to be included in this section of the SAR may be found in the reference documents [36].

Documents and records

3.176. The operator should include details of the provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and deleting documents and records which relate to the life cycle activities of the NPP. This should include the operator's documentary provisions to address configuration management for the plant. Further information on matters to be included in this section of the SAR may be found in the reference documents [3].

Outages

3.177. The operator should describe here the relevant plant arrangements for conducting periodic shutdown of the reactor as the operating cycle and other factors require. This should include measures to ensure safety of the plant during the outage period, as well as measures to ensure the safety of temporary personnel working at the plant at that time. Particular attention should be paid on measures taken for taking into account the specific aspects of outages situation, such as: multiple activities, multiple actors from different fields and services, organisation and planning, time pressure, management of unforeseen events, feedback experience of outages and how this feedback experience is analysed and used to improve outages.

X - OPERATIONAL LIMITS AND CONDITIONS

3.178. Although the practices in the Member States differ concerning the explicit inclusion of the Operational Limits and Conditions (OLCs) in the SAR, it is recognized that the operational limits and conditions form an important part of the basis on which the operating organization is authorized to operate the plant. In some countries OLC are presented as part of the SAR, in others they are prepared as a separate document which is referenced in the SAR. Whatever approach is used it is important to demonstrate in the SAR that the OLC have been developed in a systematic way.

3.179. The licensing process generally includes a consideration of the OLC in the form of controls, limits, conditions, rules and required actions that are the formal derivation of the safe operating envelope, lying within the possible operating states covered in the establishment of the design basis. This is required to ensure that the operation of the station will not present an intolerable risk to the health and safety of workers and the public, operation being at all times within the safe operating regime established for the plant. The format of the means by which such control is exercised varies with member states and 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 a safety analysis of the individual 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 [37].

3.181. If included in the SAR, detailed Operational Limits and Conditions for Operation should contain numeric values of limiting parameters and operability conditions of systems and components. The corresponding surveillance requirements to ensure that parameters remain within specified limits and that systems and components are operable should also be specified. The associated actions to be taken in the event that limits and conditions are not fulfilled should also be clearly established. In some cases, essential administrative aspects, such as minimum shift composition and frequency of internal reviews are also covered by these conditions. Reporting requirements of operational events should also be covered. Further information on matters to be included in this section of the SAR may be found in the reference documents [37].

XI - RADIATION PROTECTION

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 presented. Further information on matters to be included in this section of the SAR may be found in the reference documents [21,38,39].

3.183. The details provided should either include a brief description of the ways in which adequate radiation protection provisions have been incorporated into the design or refer to other sections of the SAR where this information can be obtained. The consideration should explain how the basic protection measures of time, distance and shielding have been addressed. It should be demonstrated that appropriate design and operational arrangements have been made to reduce the amount of unwanted radiation sources as recommended in para 3.76-3.80 of the IAEA Safety guide NS-G-2.3 [39].

Application of ALARA principle

3.184. This section should describe the operator's policy and the operational application of ALARA principle. It should be demonstrated that as a minimum the recommendations for implementation of ALARA principle, as described in the IAEA Safety Guides on Design Aspects of Radiation Protection and Waste Safety for NPPs [38] and Radiation protection and Radioactive Waste Management in the Operation of NPPs [39], are met.

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

Radiation sources

3.186. This section should describe all site radiation sources taking into account both contained or immobile sources and potential sources of airborne radioactive material. It should also examine the possible pathways of exposures.

Radiation protection design features

3.187. This section should describe equipment and facility design features implemented to ensure radiation protection, provide information for the shielding for each of the radiation sources identified, describe the personnel protection features, describe the fixed area radiation and continuous airborne radioactivity monitoring instrumentation and the criteria for selection and placement and address design provisions for any equipment decontamination, if needed.

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

- no persons should receive doses of radiation in excess of statutory dose limits as a result of normal operation;
- the occupational exposures in the course of normal operation should be as low as reasonably achievable;
- dose constrains should be used to avoid inequity in the dose distributions;
- measures should be foreseen to avoid some workers being exposed near the dose limits year by year;
- all reasonably practicable steps should be taken to prevent accidents having radiological consequences;
- all reasonably practicable steps should be taken to minimise 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 to members of the public from the operation of the NPP throughout its life-cycle.

3.190. It should be demonstrated that for the overall design, suitable provision is made in the design, layout and use of the plant to reduce doses and contamination releases from all sources. Such provisions should include adequate design of systems, structures and components so that exposures during all activities throughout the life-cycle of the plant are reduced or where no significant benefit is accrued from the activity eliminated. Reference to the SAR chapter on “System description and design conformance” on this subject might be appropriate.

Radiation Monitoring

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

Radiation Protection Programme

3.192. This section should describe the administrative organisation, 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 [39] the plant radiation protection programme is “ based on a prior risk assessment which takes into account the location and magnitude of all radiation hazards, and covers:

(a) classification of working 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; and

(k) arrangements for response to emergencies “

XII - EMERGENCY PREPAREDNESS

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, plant and the staff could be carried out, and that the decision making process for the use of these actions would be timely, disciplined, co-ordinated, and effective. The emergency preparedness arrangements should address the full spectrum of accidents with effects on the environment including severe accidents and the off-site areas where preparations for implementation of protective measures are warranted. The information should include 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 [40].

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 process 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- and off-site exercises, to ensure appropriate emergency preparedness and response arrangements are in place before commissioning should be described. The frequency of ongoing exercises foreseen to maintain adequate emergency preparedness should be established and justified.

3.196. Further information on matters to be included in this chapter of the SAR may be found in the reference documents [40,41].

Emergency Management

3.197. This section should contain an appropriate description of the plant response to the occurance of an emergency.

3.198. A general description should be presented here of the emergency arrangements for protection of the personnel and the public, in the event of accidents, including measures 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-site, protecting emergency workers, assessing the initial phase, managing the medical response and keeping the public informed.

3.199. Measures employed to ensure physical protection of the plant staff and how this will be co-ordinated with other emergency response actions, should as well 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 presented about the particular plant capability to provide:

- an on-site emergency facility where response personnel will determine, initiate and manage all on-site measures, apart from detailed control of the plant, and to transmit data on plant conditions to the off-site emergency facility;
- appropriate measures to allow control of essential safety systems from a supplementary control room;

- an off-site emergency facility where response personnel will assess information 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 organisations in order to inform and, if necessary, to protect the public;
- off-site monitoring systems passing data and information to the regulatory body if appropriate or required by national arrangements.

3.201. These descriptions should include a description of any necessary equipment, communications and other arrangements needed to support the facilities assigned functions. The habitability of these facilities and provisions to protect the staff during accidents should be described and justified.

Capability for assessment of accident progression, radiological releases and consequences of accidents

3.202. It should be demonstrated in this section that the operator will have available measures for:

- early detection, monitoring and assessment of the conditions for which emergency response actions are warranted to mitigate the accident, protect on-site personnel and to recommend appropriate protective action to off-site officials. This assessment should include actual or predicted levels of core damage;
- prediction of the extent and significance of any release of radioactive materials if an accident occurred;
- prompt and continuous assessment of the on- and off-site radiological situation; and
- continuous assessment of facility and radiological conditions in order to adjust, as appropriate, on-going response actions.

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

XIII - ENVIRONMENTAL ASPECTS

3.204. Practices among member states may vary with respect to inclusion of information on environmental aspects in the Safety Analysis Report. If this is required, this section should briefly describe the approach taken to assessing the impact on the environment of the construction of the NPP and its operation under normal conditions and its decommissioning.

Radiological Impact

3.205. This section should describe the measures that will be taken to control the discharges to the environment of solid, liquid and gaseous radioactive effluents; which should comply with the ALARA principal. This section should:

- identify any regulatory limits and operational targets of solid; liquid and gaseous discharges and measures to comply with such limits;
- describe the off-site monitoring regime for contamination and radiation levels;
- identify methods to make, store and retain records of releases that will be routinely made from the site;
- describe dedicated environmental monitoring programme and alarm systems required to respond to unplanned radioactive releases;
- identify what measures will be taken to make appropriate data available to authorities and the public.

3.206. This section should address all aspects of site activity that have the potential to affect the radiological impact of the site throughout the life cycle of the NPP, including construction, operation and decommissioning.

Non Radiological Impact

3.207. This section should describe the measures that will be taken to control the discharge to the environment of any dangerous solid, liquid and gaseous non-radioactive effluents. This section should:

- identify the chemical and physical nature of the releases or discharges;
- identify any regulatory limits and operational targets of discharges;
- describe the off-site monitoring regime for pollution;
- describe alarm systems required to respond to unplanned releases;
- identify what measures will be taken to make appropriate data available to the public.
- This section should address all aspects of site activity that have the potential to affect the non-radiological impact of the site throughout the life cycle of the NPP, including construction, operation and decommissioning.

XIV - RADIOACTIVE WASTE MANAGEMENT

3.208. This chapter should justify the adequateness of the measures proposed for the safe management of radioactive waste of all types, which are generated throughout the plant life cycle. It should make reference to the plant radioactive waste treatment system design description, provided in the SAR chapter on “System description and design conformance”.

3.209. A short description of the main sources of solid, liquid and gaseous wastes and an estimate 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 rate and the quantities, condition and form of radioactive wastes with different states of aggregation and activity level for normal and abnormal conditions of operation and for accident conditions, methods and technical means for their processing/conditioning, storage and transportation. The consideration of wastes should include solid, liquid and gaseous wastes as appropriate in all stages of the development of measures to safely address radioactive wastes throughout the plant life cycle. This section should consider the options for safe pre-disposal management of wastes. Further information on matters to be included in this section of the SAR may be found in the reference documents [30,42,43].

Control of waste

3.211. Measures to control or contain the wastes produced at all stages during the life-cycle of the NPP should be described in this section. This section may also address the proposals to categorise and segregate wastes as necessary.

Handling of radioactive waste

3.212. Measures to safely handle all types of wastes produced at all stages during the life-cycle of the NPP should be described in this section. This section should include the provisions for safely handling the wastes produced during the movement of waste from the point of origin to the identified storage point. The text should include a consideration of the possible need to retrieve wastes at some stage in the future, including the decommissioning stage.

Minimizing waste accumulation

3.213. Measures to reduce the accumulation of wastes produced at all stages during the life-cycle of the NPP should be described in this section. This should include measures employed to reduce the waste arising to a level that is as low as reasonably practicable. The assessment should show that both the volume and the activity of the waste are minimised.

Conditioning of waste

3.214. Measures to condition the wastes produced at all stages during the life-cycle of the NPP should be described in this section. Where considered prudent, wastes may be processed in accordance with established procedures and the options considered should be identified here. However, consideration should also be given to establish the most suitable option that, to the extent possible, does not foreclose alternative options in the event that preferences for waste disposal change during the life cycle of the plant.

Storage of waste

3.215. Measures to store the wastes produced at all stages during the life-cycle of the NPP should be described in this section. This section should consider quantities, types and volumes of radioactive waste, and the need to categorise and separate wastes within the provisions for storage. The possible need for specialised systems to address long term storage issues such as cooling, containment, volatility, chemical stability, reactivity and criticality should also be addressed and any such system should be described.

Disposal of waste

3.216. Measures to dispose safely of the wastes produced at all stages during the life-cycle of the NPP should be described in this section. This should include the measures to ensure safe transport to another identified location for longer term storage, if necessary.

XV - DECOMMISSIONING AND END OF LIFE ASPECTS

3.217. Decommissioning of the plant will become necessary at either the end of the plant life cycle or earlier if the operator so decides. The NPP decommissioning capability should be demonstrated before initial criticality or plant operation commences. This chapter of the report should contain the anticipated proposals for decommissioning of the plant at this point. It should be periodically up-dated to allow an increasing level of detail and to reflect developments in the decommissioning strategy. Information on matters to be included in this section of the SAR may be found in the reference documents [44,45].

Decommissioning concept

3.218. This part of the SAR should briefly discuss the proposed decommissioning concept, taking into account the following aspects :

- design solutions which minimize the waste material produced and facilitate decommissioning;
- consideration of volume and activity of waste produced during the operational and decommissioning phases;
- identified options for decommissioning;
- planning, phasing or staging of the decommissioning process including appropriate surveillance requirements throughout the process;
- adequate documentary control and maintenance of suitable and sufficient records;
- anticipated organisational changes, including provisions in place to preserve constitutional knowledge which will be needed at the decommissioning phase.

Provisions for safety during decommissioning

3.219. A short description of the measures that will ensure safety during decommissioning should be made, based on the identified safety objectives and principles. Special attention should be paid to the following aspects:

- radioactive (airborne and liquid) discharges during the process should conform to ALARA principles and at least be kept within prescribed limits;
- the practicability of adherence to the concept of defence in depth against the possible radiological hazards during the decommissioning process should also be demonstrated .

Differing approaches to decommissioning

3.220. This section should present a description of the identified options and the chosen method for decommissioning, with corresponding justification. The main reasons for the differences between the alternative approaches should be explained (minimisation of the radiation impact on personnel, the public and the environment, optimisation of the technological, economical, social and other relevant indicators). Any options and their effect on the time scale 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 time-scale, containing the following basic activities (including the anticipated schedule of implementation):

- development of an engineering study for decommissioning, identifying policy and objectives;
- development of a rational strategy for decommissioning including the identification of a staged approach to decommissioning if appropriate;
- development of a Safety Analysis Report for decommissioning;
- development of a program for bringing the reactor to a safe condition with total or partial dismantling;
- development of a program for ensuring that services will be available to support the work (heat, electrical and water supply sources);
- development of a program for providing adequate facilities for sorting, processing, transportation and storage of the radioactive wastes during decommissioning;
- providing for the physical protection, monitoring and surveillance of the unit during the decommissioning stages identified;
- observation of the licensing process throughout decommissioning.

Documentation and Records

3.222. Throughout the decommissioning process the operator should make and implement specific arrangements to issue, store, retrieve, audit and review documents to control the decommissioning process and also to ensure that full, complete and accurate records are securely retained so that actions taken by the operator are recorded for the benefit of future considerations of the site use. This should include the possibility of opting for staged or deferred decommissioning.

4. REVIEW AND UPDATE OF THE SAFETY ANALYSIS REPORT

4.1. In the licensing of a new NPP the preliminary, intermediate and final Safety Analysis Reports are important documents compiled by the operator, that the regulator uses in assessing the adequacy of the NPP design and assessing the suitability of the licensing basis. It should be noted that the SAR might be only one source of information and that the final safety justification that the regulator accepts may consist of a far wider range of information.

FORM OF THE REPORT

4.2. The operator shall agree, with the regulator, about the form of presentation, storage and use of the SAR, in respect of electronic or paper form. This will need to take full account of the regulatory position in respect of the acceptability of alternative report forms, in accordance with national laws and regulations where appropriate.

ROUTINE REVISIONS TO THE SAFETY ANALYSIS REPORT

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

4.4. The need for routine review and update of the SAR depends upon the role of the report in the ongoing licensing process. Factors that a regulator should consider in determining the need for SAR routine update and the updating frequency are related to the way in which the overall safety justification and documentation is maintained; implementation of major plant modifications and revisions to the OLCs; frequency of plant periodic safety reviews (PSR), etc. Guidance on the performance of periodic safety reviews and periodic updating of SAR may be found in the reference document [10].

LIST OF ABBREVIATIONS AND ACRONYMS

AC	Alternate Current
ALARA	as low as reasonably achievable
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CANDU	heavy water reactor
DC	Direct Current
ESF	Engineering Safety Features
EMI	Electro Modulation Interference
FMEA	Failure Mode and Effects Analysis
HVAC	Heat, Ventilation and Air Conditioning
MS	Member State
NPP	Nuclear Power Plant
OLC	Operational Limits and Conditions
PCSAR	Pre-Construction Safety Analysis Report
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
POSAR	Pre-Operation Safety Analysis Report
PWR	Pressurized Water Reactor
QA	Quality Assurance
RBMK	Water graphite reactor
RCS	Reactor Coolant System
RFI	Radio Frequency Interference
SAR	Safety Analysis Report
SS&C	Structure, Systems and Components
US NRC	United States Regulatory Commission

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