Státní úřad
pro jadernou bezpečnost

jaderná
bezpečnost

PROPOSED TABLE OF CONTENTS FOR SAFETY ANALYSIS REPORTS

bezpečnostní návod JB-1.12

SÚJB
prosinec 2010

Jaderná bezpečnost
PROPOSED TABLE OF CONTENTS FOR SAFETY ANALYSIS REPORTS
Vydal: Státní úřad pro jadernou bezpečnost, prosinec 2010
Účelová publikace bez jazykové úpravy

Content

List of acronyms

AC	Alternate Current
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
DBA	Design Basis Accident
DC	Direct Current
EIA	Environmental Impact Assessment
EOPs	Emergency Operating Procedures
ESF	Engineered Safety Features
FMEA	Failure Mode and Effects Analysis
FSRA	Final Safety Analysis Report
HFE	Human Factor Engineering
HSI	Human-System Interfaces
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, Air Conditioning
IAEA	International Atomic Energy Agency
ISAR	Initial Safety Analysis Report
I&C	Instrumentation and Control
LOCA	Loss of Coolant Accident
MCR	Main Control Room
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OLCs	Operational Limits and Conditions
QER	Operating Experience Review=
P&IDs	Piping and Instrumentation Diagrams
PIE	Postulated Initiating Event
POSAR	Pre-operational Safety Analysis Report
PSA	Probabilistic Safety Analysis
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
SAR	Safety Analysis Report
SSC	Structures, Systems, Components
TG	Turbine Generator
USNRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation
WENRA	Western European Nuclear Regulators Association

General part

Introduction and objective of the Guide

The Safety Analysis Report (SAR) is a basic licensing document, compiled by the operating organization, that the regulatory body uses in assessing the adequacy of the plant design and the suitability of the licensing basis. The SAR should provide adequate justification that a nuclear power plant (NPP) meets all appropriate safety requirements, and, at later stages of the plant implementation, that the plant has been built and commissioned as intended, and that all design, construction and commissioning changes have been properly addressed. In addition to providing a documented justification that the plant has been designed to appropriate safety standards, the SAR also demonstrates that the plant will be operated safely and provides reference material for the safe operation.

In general, the SAR contains the following information:

- general description of the plant,
- site characteristics,
- requirements on design of systems, structures and components (SSCs); including criteria for mechanical components, civil structures, electrical systems, instrumentation, and control systems,
- description of individual systems, structures and components; a common structure is proposed to follow to the extent possible for all sections dealing with systems or equipment,
- justification of safe operation by safety analysis,
- justification of operational and administrative measures.

The objective of this Guide is to provide guidance on the possible content and structure of a SAR as the most important document for licensing of siting, construction, commissioning and operation of a NPP. In this way the Guide is intended to facilitate development of the SAR by the operating organization and checking completeness and adequacy of the SAR by the regulatory body. However, the structure proposed should not be interpreted as a strict requirement to be followed verbatim. In each specific case the operating organization should agree with the regulatory body on the content, structure, form of the presentation, storage and use of the SAR. In taking this decision full account should be taken of the regulatory position on the acceptability of alternative forms of reports.

The current Guide has two main parts: this general part, and other part directly devoted to the description of the content and structure of the SAR.

This general part in next sections addresses the following:

- Applicability of the current guide to various stages of the Safety Analysis Report as required during the NPP implementation project,
- Applicability of the guide to different nuclear facilities, including existing and new NPPs,

- General approach to specification of the structure of the Safety Analysis Report, summarizing the reasons and way for development of the current Guide,
- Structure of the Safety Analysis Report, specifying chapters of the SAR,
- Use and updating of SAR during plant operation,
- Formal aspects of the Safety Analysis Report,
- Relation of the Safety Analysis Report to other licensing documents.

Second part of this Guide devoted to the description of the content and structure of the SAR is further subdivided into the main body and the Appendix. The content of each chapter is briefly described in the main body. An example of the detailed list of content is in the Appendix. Although this example is not meant as the obligatory list of content, it is advisable to follow the example to the reasonable extent in order to facilitate and accelerate the review of the SAR by the regulatory body.

In development of this Guide the following documents have been broadly used:

- INTERNATIONAL ATOMIC ENERGY AGENCY, Format and Content of the Safety Analysis Report for Nuclear Power Plants, Safety Standards Series No. GS-G-4.1, IAEA, Vienna, (2004)
- INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, General Safety Requirements No. GSR Part 4, Vienna (2009)
- NUCLEAR REGULATORY COMMISSION, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Regulatory Guide 1.70, Rev. 3, NRC, Washington, DC (1978)
- NUCLEAR REGULATORY COMMISSION, Combined License Applications for Nuclear Power Plants, Regulatory Guide 1.206, NRC, Washington, DC, (2007)
- WESTERN EUROPEAN NUCLEAR REGULATORS ASSOCIATION, Reactor Harmonization Group, WENRA Reactor Safety Reference Levels, Issue N, January 2008.

Key guidance documents regarding the format and content of the SAR are IAEA Safety Guide No. GS-G-4.1 and US NRC Regulatory Guides 1.70 and 1.206. The IAEA Safety Guide as compared to US Regulatory Guides (in addition to some changes in SAR content) is structured in a different way, which could be interpreted as a proposal for a different structure of the SAR (see Annex 3 for comparison of structures, taking into account that similar title not necessarily corresponds to the similar content). In the current Guide an attempt was made to combine the both IAEA and US NRC key guidance documents in order to reflect both recent trends established by major reactor vendors and new safety requirements introduced in the IAEA Safety Standards.

Applicability of the current guide to various stages of the Safety Analysis Report

In the Czech Republic, in accordance with common practice, several subsequent Safety Analysis Reports should be developed for authorizations of the different licensing stages:

- Initial Safety Analysis Report (ISAR), which is the basis for site approval,
- Preliminary Safety Analysis Report (PSAR), which is the basis for construction permit,
- Pre-operational Safety Analysis Report (PoSAR), which is the basis for authorization of NPP commissioning and operation. During the NPP operation the PoSAR can be further complemented by additional information, so forming operational or Final Safety Analysis Report (FSAR).

The structure of the SAR proposed in the current Guide fits best to the POSAR and FSAR. Nevertheless, the SAR through its subsequent stages should preferably be a continuously updated document that reflects the current NPP implementation stage. It also means that the similar structure of the SAR should be maintained from the ISAR up to PoSAR as far as reasonable. It should be however taken into account that the level of information is substantially developing during the implementation of the NPP project. Guiding principle is that any subsequent SAR should revise and provide more specific information on the topics outlined in the previous SAR and explain and justify any departure from previous safety provisions.

In particular, at the stage of the ISAR the information about the plant may be limited, and even the future reactor design may not be selected. While information about the site itself should be reasonably complete, effects of the future NPP on the site and its environment should be based on a reasonable estimate, using e.g. bounding (enveloping) approach. Rather than describing safety features of the future NPP, the ISAR should describe relevant safety principles and requirements and to some extent also to indicate, how these requirements will be complied with. Due to given reasons not all the subchapters as specified in the current Guide should be included in the ISAR. Since in many cases the text should consist of summary of requirements and these are often not existing in much details, it may be practicable to combine several subsections into one overwhelming section.

At the next stage the PSAR should already contain sufficiently detailed information, specifications and supporting calculations to enable those responsible for safety to assess whether the plant can be constructed and operated in a manner that is acceptably safe throughout its lifetime. The PSAR should convincingly demonstrate that the requirements specified in ISAR are complied with. The safety features incorporated into the design, together with the possible challenges to the plant that have been considered, should be described, with due regard to any site specific aspects. The amount of information to be provided in the preliminary report will depend on the extent to which the proposed reactor design is based on a generic type or a standard design for which the licensing process has been followed previously, including the production of a SAR.

The PoSAR should revise and provide more specific information on the topics outlined in the PSAR, taking into account all modifications implemented during the NPP construction and justifying any departure from or revisions to the safety provisions or the design intent as set out in the PSAR. The PoSAR should essentially justify the finalized detailed design of the plant and presents a demonstration of its safety. In addition, the PoSAR should deal in greater detail than the PSAR with matters relating to the commissioning and operation of the plant during this phase of its lifetime. The PoSAR should provide more up to date information on the licensing basis for the plant.

The level of information required in different chapters in different stages of SAR is specified in the Annex 1.

Applicability of the guide to different nuclear facilities

This Guide is intended for use with NPPs but it may, in parts, have a wider applicability to other nuclear facilities. The particular contents of the SAR will depend on the specific type and design of the NPP proposed, and this will determine how different sections as in this Guide will be included in the SAR.

Although intended mainly for use with new plants, the guidance presented here could also be useful for existing NPPs when operating organizations periodically review their existing SARs to identify any areas in which improvements may be appropriate and/or to review the licensing basis. However, for existing NPPs it would be meaningless or even counterproductive to safety to devote too much effort and resources just rearranging the text of the SAR according the newly proposed structure.

General approach to specification of the structure of the Safety Analysis Report

Development of this Guide introducing certain modifications to previous SARs is justified by the need to reflect current safety requirements (in particular IAEA Safety Standards) and recent progress in development of safety documentation for new reactor designs. While there is a general trend in available safety documentation developed by all major reactor vendors to follow the USNRC Regulatory Guide 1.206 regarding the content and level of details, there is also a need to harmonize approaches and to ensure consistency with recently developed national legislation and/or published IAEA Safety Standards on siting, design, operation and in particular on safety assessment of NPPs. Consistency with the IAEA Safety Standards which sometimes goes beyond the intent of the RG 1.206 is also useful in order to comply with the intent of the WENRA Reference Levels for existing reactors.

The proposed table of content is based mainly on the USNRC Regulatory Guides 1.70 and 1.206 with certain modifications made due to intent of achieving closer consistency with current IAEA Safety Standards. These modifications are introduced in those chapters which are in some way associated with current safety requirements and approaches in particular in the following areas:

- defence in depth,
- management of safety,
- description of plant systems, using common structure in description,

- civil structure engineering,
- emergency preparedness,
- decommissioning,
- environmental aspects,
- deterministic safety analysis of all plant states, including beyond design basis accidents and severe accidents,
- probabilistic safety analysis.

The intent in preparing this Guide was to minimize subjective formulations, but rather to rely on internationally recognized descriptions. Therefore, the text of the current Guide was to the extent possible taken (preferably by verbatim) from the IAEA Safety Guide No. GS-G-4.1, properly restructured in accordance with newly established structure. In some instances, when adequate description was not found in GS-G-4.1, the USNRC Regulatory Guide 1.206 was consulted regarding proper description. However, due to large difference in sizes of both reference documents (GS-G-4.1 and RG 1.206) only short parts of the RG 1.206 was utilized in order to maintain reasonable size of the current Guide. The user of this Guide are nevertheless encouraged to use RG 1.206 for further reference, with attention paid to potential differences indicated in the list of bullets above.

Structure of the Safety Analysis Report

In accordance with the proposal in the current Guide the SAR is structured into following 21 chapters:

Chapter 1. Introduction and general considerations,

Chapter 2. Site characteristics,

Chapter 3. Design of structures, systems and components,

Chapter 4. Reactor,

Chapter 5. Reactor coolant and connected systems,

Chapter 6. Engineered safety features,

Chapter 7. Instrumentation and control,

Chapter 8. Electric power,

Chapter 9. Auxiliary systems and civil structures,

Chapter 10. Steam and power conversion systems,

Chapter 11. Radioactive waste management,

Chapter 12. Radiation protection,

Chapter 13. Conduct of operations,

Chapter 14. Plant commissioning,

Chapter 15. Safety analysis,

Chapter 16. Operational limits and conditions (technical specifications),

Chapter 17. Management systems,

Chapter 18. Human factors engineering,

Chapter 19. Emergency preparedness,

Chapter 20. Environmental aspects,

Chapter 21. Decommissioning and end of life aspects.

The appendix of the current guide provides quite detailed logically arranged structure of individual chapters of the SAR. Objective of this guidance is to indicate the expected comprehensiveness of information provided in the SAR. However, the applicant may consider a modified structure, taking into account various factors such as specifics of the design or “traditional” arrangement of information provided.

Special issue of the proposed structure is introducing into the SAR several new chapters, which were previously either missing in the SAR or covered by another documents. Examples of such chapters include management systems, emergency preparedness, environmental aspects, decommissioning and end of life aspects. While it is acceptable to complement the SAR by separate documents, in view of maximum comprehensiveness of the SAR it is advisable at least for new NPPs to provide summary of such documents or at least to make reference to them.

Use and updating of SAR during plant operation

Although the SAR has its importance mainly as licensing document, its use should not be limited for the licensing prior the operation. The SAR can be also a useful means of providing public assurance regarding the safety of the plant. However, possibly the most important purpose of the SAR is its use by the licensee to manage safety. The SAR itself does not guarantee safety; it is therefore essential that the plant operator implement the safety intent embodied in the SAR by developing appropriate safety management, procedures and instruction. The SAR identifies the limits and conditions for safe plant operation and these provide the basis for the development of operating procedures and instructions including maintenance.

Since the SAR is part of the overall justification of plant safety, it should reflect the current state and the licensing basis of the plant and should be kept up to date accordingly. The updating of the SAR is initiated by safety related activities including:

- plant hardware modifications,
- plant inspections,
- procedural changes,
- maintenance findings (ageing),
- periodic safety review,
- analysis of operational events,

- analysis of experiences from other similar plants,
- changes to analytical techniques, standards and criteria,
- interventions by the regulatory body.

Ideally the SAR should reflect the current plant status at all times. Since such ideal situation is difficult to achieve, it is considered as a good practice to update the SAR at the time of refuelling outages or statutory shutdowns. As minimum, updating of the SAR should be a part of the Periodic Safety Review usually scheduled every ten years.

It is however essential that all activities that could impact on the validity of SAR are clearly identified and controlled by procedures that include a requirement to review the impact of each event. Between the SAR updates, the full impact of any plant modification on the plant safety should be evaluated before implementation and any necessary changes made to safety management system and plant operating procedures. Documentation that initiates the change should be available for inspection by the regulatory body.

Formal aspects of the Safety Analysis Report

The SAR should document the safety level of a NPP in sufficient scope and detail to support the conclusions reached and to provide an adequate input into independent verification and regulatory review. Depth of description in the SAR is determined by the requirement that the SAR is a basic reference material, thus the report shall be understandable by itself.

It is required that information is presented in the SAR in a clear and concise way. Each subject should be treated in sufficient depth and should be documented to permit a reviewer to evaluate the safety level independently. Tables, drawings, plots and figures should be used wherever they contribute to the clarity and brevity of the report.

The information contained in the SAR should be self-sufficient to the extent reasonable. The most important supporting materials should be supplemented to the SAR. These materials are enhancing the review process and the later usability of the SAR. Some less essential external references are usually not submitted together with the SAR for the regulatory review, but they should be made available on request.

User friendly format of the SAR significantly facilitates its use and review. At present the SAR should be available in electronic form. Use of internal reference links in electronic form is very useful. Use of external references and their extended use is almost inevitable (detailed design documents, references to standards, detailed analysis reports, code validation reports, source material for PSA , etc.). References to lower level documents are also useful (e. g. operational procedures and EOPs)

Relation of the Safety Analysis Report to other licensing documents

It should be noted that the SAR may be only one of several documents used in the licensing process possibly even governed by different legislation. Typical example is the Environmental Impact Assessment study, PSA study, emergency plans or decommissioning plans. Some of the information in the SAR may be the same as required for other licensing documents. In such cases, the required information should be in parallel incorporated in all relevant documents to the appropriate extent. The reason is that these documents are responsive to different legislative requirements and each of them should be essentially self-contained.

Consistency and continuity of information provided in different licensing documents as well as in subsequent stages of the SAR should be ensured. Any significant differences between information provided these documents should be explained and justified.

Description of the content of the SAR

Chapter 1. Introduction and general considerations

1.1 Introduction

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

(a) A statement of the main purpose of the SAR;

(b) A description of the existing authorization status;

(c) The main information on the preparation of the SAR;

(d) A description of the structure of the SAR, the objectives and scope of each of its sections and the intended connections between them.

1.2 Identification of Stakeholders

The primary agents or contractors for the design, construction, and operation of the NPP should be specified in this section. The principal consultants and outside service organizations (such as those providing audits of the quality assurance programme) should be also identified. The division of responsibility between the reactor/facility designer(s), architect-engineer(s), constructor(s), and plant operator should also be delineated.

1.3 General plant description

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

The section should present briefly (in a table, where appropriate) the principal elements of the installation, including the number of units at the plant, where appropriate, the type of plant, the principal characteristics of the plant, the primary protection system, the type of nuclear steam supply system, the type of containment structure, the thermal power levels in the core, the corresponding net electrical power output for each thermal power level, and any other characteristics necessary for understanding the main technological processes included in the design.

1.4 Comparison with Other Facilities

It may be useful to compare the plant design with similar earlier designs already approved by the regulatory body, so as to identify the main differences and assist in the justification of any modifications and improvements made. A list of selected plant characteristics should be included in an appendix to the SAR.

A statement of any similar (or identical) plants that the regulatory body has already reviewed and approved and a statement of the specific differences and improvements that have been made since such an approval was granted;

1.5 Additional information concerning new safety features

This section should describe information or provide references to the location of information that demonstrates the performance of new safety features for NPPs that differ significantly from previous light-water reactor designs such as use simplified, inherent, passive, or other innovative means to accomplish their safety functions.

1.6 Operating modes of the plant

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

1.7.Principles of safety management

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

1.8 Additional documents considered as a part of the safety analysis report

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

1.9 Information on the layout and other aspects

Basic technical and schematic drawings of the main plant systems and equipment should be included in this section, together with details of the physical and geographical location of the facility, connections with the electricity grid and means of access to the site by rail, road and water. The operating organization should provide general layout drawings for the entire plant. The illustrations should be complemented with a brief description of the main plant systems and equipment, together with their purposes and interactions. References should be made, where necessary, to other chapters of the SAR that present detailed descriptions of specific systems and equipment.

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

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

1.10 Conformance with applicable regulations, codes and standards

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

Chapter 2. Site characteristics

Chapter 2 should provide information on the geological, seismological, volcanic, hydrological and meteorological characteristics of the site and the surrounding region in conjunction with the present and projected population distribution and land use that is relevant to the safe design and operation of the plant. Sufficient data should be included to permit an independent evaluation.

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

(a) Site specific hazard evaluation for external events (of human or natural origin);

(b) Design targets in terms of recurrence probability of external events;

(c) Definition of the design basis for external events;

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

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

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

A discussion of considerations concerning the site exclusion and/or acceptance criteria applied for the purposes of preliminary screening of the site for suitability after the site survey phase should be provided in this section of the SAR.

Site related information represents a very important input to the design process and may be one of the sources of uncertainty in the final safety evaluation. The measures employed to account for such uncertainty levels should be considered in the SAR.

2.1 Geography and demography

This section should specify the site location, including both the area under the control of the licensee and the surrounding area in which there is a need for consultation on the control of activities with the potential to affect plant operation, including flight exclusion zones. Information on such activities would include relevant data on the population distribution and density and on the disposition of public and private facilities (airports, harbours, rail transport centres, factories and other industrial sites, schools, hospitals, police services, fire fighting services and municipal services) around the plant site. This section should also cover the uses of the land and water resources in the surrounding area, for example for agriculture, and should include an assessment of any possible interaction with the plant.

2.2 Evaluation of site specific hazards

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

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

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

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

2.3 Nearby industrial, transportation, and military facilities

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

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

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

Measures for site protection (including dams, dykes and drainage) and any modifications to the site (such as soil substitution or modifications to the site elevation) are usually considered part of the site characterization stage and their assessment in relation to the design basis should be considered in this section of the SAR.

2.5 Hydrology

This section should present sufficient information to allow an evaluation of the potential implications of the hydrological conditions at the site for the plant design, performance requirements and safe operation. These conditions should include conditions relating to phenomena such as abnormally heavy rainfall and runoff floods from watercourses, reservoirs, adjacent drainage areas and site drainage. This section should also include a consideration of flood waves resulting from dam failures, ice related flooding and seismically generated water based effects on and off the site. For coastal and estuary sites, tsunamis, seiches and the combined effects of tides and strong wind should be evaluated. The information given in this section will be relevant to the assessment of the transport of radioactive material to and from the site and the dispersion of radionuclides to the environment.

2.6 Meteorology

This section should provide a description of the meteorological aspects relevant to the site and its surrounding area, with account taken of regional and local climatic effects. To this end, data deriving from on-site meteorological monitoring programmes should be documented. The extreme values of meteorological parameters, including temperature, humidity levels, rainfall levels, wind speeds for straight and rotational winds, and snow loads, should be evaluated in relation to the design. The potential for lightning and windborne debris to affect plant safety should be considered, where appropriate. The information given in this section will be relevant to the assessment of the transport of radioactive material to and from the site and the dispersion of radionuclides to the environment.

2.7 Geology, seismology, and geotechnical engineering

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

Site reference data relating to geotechnical soil properties and groundwater hydrology should also be provided. The investigation campaigns for the collection of data for the design of foundations, the evaluation of the effects of soil–structure interaction, the construction of earth structures and buried structures, and soil improvements at the site should be described.

The SAR should present the relevant data for the site and the associated ranges of uncertainty to be used in the structural design and the dispersion studies for radioactive material. Reference should be made to the technical reports describing in detail the conduct of the investigation campaigns, and their extension, and the origin of the data collected on a regional basis and/or on a bibliographic basis. The design of earth structures and site protection measures, if relevant, should also be documented. A description of projected developments relating to the above mentioned information should also be provided and should be updated as required.

2.8 Radiological conditions due to external sources

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

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

2.9 Site related issues in emergency planning and accident management

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

2.10 Monitoring of site related parameters

The provisions to monitor site related parameters affected by seismic, atmospheric, water and groundwater related, demographic, industrial and transport related developments should be described in this section. This may be used to provide necessary information for emergency operator actions in response to external events, to support the periodic safety review at the site, to develop dispersion modelling for radioactive material and as confirmation of the completeness of the set of site specific hazards taken into account.

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

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

Chapter 3. Design of structures, systems, and components

Chapter 3 should outline the general design concepts, requirements, codes and standards, and the approach adopted to meet the fundamental safety objective. The compliance of the actual design with the specific technical safety requirements should be demonstrated in more detail in other sections of the SAR, which may be referenced here.

3.1 General safety design basis

The overall safety philosophy, safety objectives and high level principles used in the design should be presented in this section. These should be based on the fundamental safety objective presented in the IAEA Safety Fundamentals. Five relevant subjects are discussed further in the following subsections: defence in depth, safety functions, general design basis and plant states, radiation protection and radiological acceptance criteria, and deterministic and probabilistic design principles and criteria.

Defence in depth

This subsection should describe in general terms the design approach adopted to incorporate the defence in depth concept into the design of the plant. It should be demonstrated that the defence in depth concept has been considered for all safety related activities. The design approach adopted should ensure that multiple and (to the extent possible) independent levels of and barriers for defence are present in the design to provide protection against operational occurrences and accidents regardless of their origin. The selection of the main barriers should be described and justified. Particular emphasis should be placed on systems important to safety. Where appropriate, any proposed operator actions to mitigate the consequences of events and to assist in the performance of important safety functions should be included.

Safety functions

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

In addition to the fundamental safety functions, any specific safety functions should be identified. For example, heat removal should be considered a safety function necessary not only for the safety of the reactor core but also for the safety of any other part of the plant containing radioactive material that needs to be cooled, such as spent fuel pools and storage areas.

General design basis and plant states

This subsection should describe the plant capabilities to cope with a specified range of operational states and accident conditions. Plant operating modes should be described.

Plant states considered in the design should be listed and grouped into categories. Basis for categorization of plant states should be explained. Initiating events (postulated initiating events, both of internal and external origin) specifically considered in the plant design should be listed.

Radiation protection and radiological acceptance criteria

This subsection should describe in general terms the design approach adopted to meet the radiation protection objective and to ensure that, in all plant states, radiation doses within the installation or due to any release of radioactive material from the installation are kept below authorized limits and as low as reasonably achievable (ALARA), as is required, economic and social factors being taken into account. Relevant radiological acceptance criteria should be assigned for each category of plant states (normal operation, AOOs (AOOs), design basis accidents and beyond design basis accidents) and they should be specified in this subsection.

Deterministic and probabilistic design principles and criteria

This subsection should provide a general description of the deterministic design principles and criteria embodied in the design. Where aspects of the design are to be based on conservative deterministic principles, such as those embodied in international standards or internationally accepted industrial codes and standards, or in regulatory guidance documents, the use of such approaches should be elaborated in this subsection of the SAR.

The single failure criterion should have special place in these considerations. This should include provisions to employ redundancy, diversity and independence, to protect against common cause and common mode failures. Consideration should be given to the possibility of a single failure occurring while a redundant train of a system is out for maintenance and/or is impaired by hazards.

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

If probabilistic safety criteria have been used in the design process, these criteria should be specified in this subsection.

3.2 Classification, load combinations, and allowable stresses

This section should provide information on the approach adopted for the categorization and safety classification of structures, systems and components. It should include information on the methods used to ensure that they are suitable for their design duty, remain fit for purpose and continue to perform any required safety function claimed in the design justification (in particular those functions claimed in the safety analyses and presented in the corresponding chapter of the SAR). If there is a potential for structures or systems to interact, then details should be provided here of the way in which it has been ensured in the design that a plant provision of a lower class or category cannot unduly impair the role of those with a higher classification. A list of safety relevant systems and main structures and components, with their classifications and categorization, should be included as an annex or reference here.

Combinations of various kinds of loads including loads from randomly occurring individual events and corresponding allowable stresses should be also described here.

3.3 Protection against external hazards

This section should provide a list of external hazards considered in the design, quantitative design parameters of individual hazards, relevant design criteria, codes and standards, methods of assessment and a description of the general design measures provided to ensure that the essential structures, systems and components important to safety are adequately protected against the detrimental effects of all the hazards considered in the plant design. External hazards to be considered include:

- Seismic events,
- Extreme winds,
- External flooding,
- Extreme ambient temperature,
- Missiles (missiles generated by extreme winds or explosion, aircraft cash),
- Any other relevant site specific external hazards,

3.4 Protection against internal hazards

This section should provide a list of internal hazards considered in the design, quantitative design parameters of individual hazards, relevant design criteria, codes and standards, methods of assessment and a description of the general design measures provided to ensure that the essential structures, systems and components important to safety are adequately protected against the detrimental effects of all the hazards considered in the plant design. Internal hazards to be considered include:

- Internal fires,
- Internal flooding,
- Internal missiles,
- Dynamic effects associated with high energy pipe ruptures,
- Any other relevant design specific internal hazards.

3.5 Civil works and structures

This section should present relevant information on the design of civil engineering works and structures. It should include specification of the design principles and criteria and the codes and standards used in the design. It should also briefly review the way in which the necessary safety margins have been demonstrated for the construction of buildings and structures that are relevant to nuclear safety, including the seismic classification of buildings and structures.

In addition to general design principles for structural and civil engineering, more specific information should be provided on foundations and on buildings. In particular, this section should specify the safety requirements for the containment building itself, including its leaktightness, mechanical strength, pressure resistance and resistance to hazards. The specific information should be composed of the following items:

- Applicable codes, standards, and specifications,
- Loads and load combinations,
- Design and analysis procedures,
- Structural acceptance criteria,
- Materials, quality control, and special construction techniques,
- Testing and in-service inspection requirements.

3.6 Mechanical systems and components

Relevant information on design principles and criteria, and the codes and standards used in the design of mechanical components should be included in this section. Information should be provided concerning the design transients and resulting loads and load combinations with appropriate specified design and service limits for components and supports. The methods, assumptions, computer programmes or experimental verification used in dynamic and static analyses to determine the structural and functional integrity of the mechanical components should be presented. Other information items to be presented in this section include:

- Dynamic testing and analysis of systems, components, and equipment,
- ASME Code Class 1, 2, and 3 components, component supports, and core support structures,
- Control rod drive systems (design requirements and assessment methods only),
- Reactor pressure vessel internals (design requirements and assessment methods only),
- Functional design, qualification, and in-service testing programmes for pumps, valves, and dynamic restraints,
- Piping design,
- Threaded fasteners.

3.7 Instrumentation and control systems and components

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

- Performance,

- Design for reliability,
- Independence,
- Failure modes,
- Control of access to equipment,
- Set points,
- Quality,
- Testing and testability,
- Maintainability,
- Documentation,
- Identification of items important to safety.

3.8 Electrical systems and components

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

- Redundancy,
- Independence,
- Diversity,
- Controls and monitoring,
- Identification,
- Capacity and capability,
- Sharing of components in multiunit plants,
- Operating modes,
- Control of access to the emergency power system.

3.9 Equipment qualification

This section should describe the qualification procedure adopted to confirm that the plant items important to safety are capable of meeting the design requirements and of remaining fit for purpose when subjected to the range of individual or combined environmental challenges identified, throughout the lifetime of the plant. Seismic, environmental and electromagnetic environmental factors should be considered. If acceptance criteria are used for the qualification of plant items by testing or analysis, these should be described here. The qualification programme should take account of all identified and relevant potentially disruptive influences on the plant, including internal and external hazard based events. A complete list of equipment items, together with their environmental qualification, should be included as an annex or referenced here.

3.10 In-service monitoring, tests, maintenance and inspections

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

3.11 Compliance with national and international regulations

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

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

Chapter 4. Reactor

This chapter should provide relevant information on the reactor to demonstrate the capability of the reactor to perform its safety functions throughout its intended lifetime in all operational modes. Reactor pressure vessel as the pressure boundary is described separately in chapter 5 of the SAR.

4.1 Summary description

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

For each of the reactor components, the more detailed description below should also include (see the Annex 2 for further details):

- The component function and safety design bases,
- Description of the component itself including design drawings,
- Material specifications, including nuclear, radiological, chemical, physical and mechanical properties,
- Interfaces with the other relevant components,
- Monitoring, inspection, testing and maintenance provisions,
- Demonstration that all relevant functional requirements, requirements of industrial codes and standards, and regulatory requirements have been considered adequately,
- Demonstration that the component has sufficient capacity to fulfil its safety function.

4.2 Fuel system design

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

4.3 Nuclear design

Descriptions of the following should be provided:

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

(ii) The nuclear characteristics of the lattice, including core physics parameters, fuel enrichment distributions, burnable poison distributions, burnup distributions, control rod locations and refuelling schemes.

(iii) The analytical tools, methods and computer codes (together with information on code verification and validation and uncertainties) used to calculate the neutronics characteristics of the core, including reactivity control characteristics.

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

(v) The neutronics stability of the core throughout the fuel cycle, with consideration given to the possible normal and design basis operating conditions of the plant.

(vi) Changes in nuclear design from prior design practices, if relevant.

4.4 Thermal and hydraulic design

Descriptions of the following should be provided:

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

(ii) The analytical tools and methods and computer codes (together with codes for verification and validation information and uncertainties) used to calculate thermal and hydraulic parameters.

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

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

(v) Instrumentation requirements.

4.5 Reactivity control systems:

All reactivity control systems should be described. A demonstration should be provided that the reactivity control systems, including any essential ancillary equipment and hydraulic systems, are designed and installed to provide the required functional performance and are properly isolated from other equipment.

4.6 Evaluation of combined performance of reactivity control systems

This section should describe the relevant situations and evaluate the combined functional performance for accidents where two or more reactivity systems are used. This section should include failure analyses to demonstrate that the reactivity control systems are not susceptible to common-mode failures when used redundantly. These failure analyses should consider failures originating within each reactivity control system as well as those originating from plant equipment other than reactivity systems and should be provided in tabular form with supporting discussion and logic.

4.7 Reactor internal structures.

Descriptions of the following should be provided:

(i) The systems of reactor internals, defined as the general external details of the fuel, the structures into which the fuel has been assembled (e.g. the fuel assembly or fuel bundle), related components required for fuel positioning and all supporting elements internal to the reactor, including any separate provisions for moderation and fuel location (description of interfaces). Reference should be made to the other sections of the SAR that cover related aspects of the reactor fuel and also fuel handling and storage.

(ii) The physical and chemical properties of the materials used for the reactor internal components, as well as nuclear, thermohydraulic, structural and mechanical aspects of the components, the expected response to static and dynamic mechanical loads, and their behaviour, and a description of the effects of irradiation and corrosion on the ability of the reactor internals to perform their safety functions adequately over the lifetime of the plant.

(iii) Any significant subsystem components, including any separate provisions for moderation and fuel location, with corresponding design drawings, and a consideration of the effects of service on the performance of safety functions, including surveillance and/or inspection programmes for reactor internals to monitor the effects of irradiation and ageing on the internal components.

(iv) The programme to monitor the behaviour and performance of the core, which should cover provisions to monitor the neutronics, dimensions and temperatures of the core.

Chapter 5. Reactor coolant and connected systems

Chapter 5 should provide relevant information on the reactor coolant system (RCS) and its associated systems, where possible in the format described in Annex 2. In addition, the following information should be provided so as to demonstrate that the RCS will retain its required level of structural integrity in both operational states and accident conditions.

(a) Integrity of the reactor coolant pressure boundary:

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

(b) Reactor vessel:

Information should be provided that is detailed enough to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards. This includes the reactor vessel materials, the pressure–temperature limits and the integrity of the reactor vessel, including embrittlement considerations.

(c) Design of the RCS:

A description and justification should be provided of the performance and design features that have been implemented to ensure that the various components of the RCS and the subsystems interfacing with the RCS meet the safety requirements for design. This should include, where applicable, the reactor coolant pumps (RCPs), the steam generators or boilers, the reactor coolant piping or ducting, the main steamline isolation system, the isolation cooling system of the reactor core, the main steamline and feedwater piping, the pressurizer, the pressurizer relief discharge system, the provisions for main and emergency cooling, and the residual heat removal system, including all components such as pumps, valves and supports.

5.1 Summary description

This section should provide a summary description of the RCS and its various components. This description should indicate the independent and interrelated performance and safety functions of each component and should include a tabulation of important design and performance characteristics.

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

5.2 Reactor coolant system and reactor coolant pressure boundary

This section should focus on measures implemented to ensure integrity of the RCS throughout the plant lifetime as described in paragraph (a) above. In addition, the section should provide information on overpressure protection of the reactor coolant pressure boundary and the secondary side of steam generators, including all pressure-relieving devices (safety and relief valves). Coolant leakage detection provisions should be described too.

5.3 Reactor vessel

This section should contain pertinent data in sufficient detail to provide assurance of the reactor vessel integrity under all relevant plant states (see the paragraph (b) above).

5.4 Reactor coolant pumps

A description and justification should be provided of the performance and design features that have been implemented to ensure that the reactor coolant pumps meet the safety requirements for design. The information should discuss the provisions taken to preclude rotor overspeeding of the RCP in the event of a design-basis loss of coolant accident (LOCA).

5.5 Primary heat exchangers (steam generators)

A description and justification should be provided of the performance and design features that have been implemented to ensure that the steam generators meet the safety requirements for design. Estimates of design limits for radioactivity levels in the secondary side of the steam generators during normal operation should be provided, including the bases for those estimates and the potential effects of tube ruptures. The design criteria to prevent unacceptable tube damage should be specified, including

(1) design conditions and transients that will be specified in the design of the steam generator tubes and the operating condition category selected (i.e., upset, emergency, or faulted) that defines the allowable stress intensity limits to be used and the justification for this selection

(2) extent of tube-wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design-basis pipe break in the reactor coolant pressure boundary or a break in the secondary piping during reactor operation.

5.6 Reactor coolant piping

A description and justification should be provided of the performance and design features that have been implemented to ensure that the reactor coolant piping meets the safety requirements for design. The description should include the design, fabrication, and operation provisions to control those factors that contribute to stress-corrosion cracking.

5.7 Reactor pressure control system

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

5.8 Reactor coolant system component supports and restraints

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

5.9 Reactor coolant system and connected system valves

A description and justification should be provided of the performance and design features that have been implemented to ensure that the valves interfacing with the RCS meet the safety requirements for design.

5.10 Access and equipment requirements for in-service inspection and maintenance

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

- Accessibility,
- examination categories and methods,
- inspection intervals,
- provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects,
- system pressure tests

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

5.11 Reactor auxiliary systems

A description and justification should be provided of the performance and design features that have been implemented to ensure that the various subsystems interfacing with the RCS meet the safety requirements for design. These subsystems include:

- Chemical and volume control system,
- Reactor coolant make-up system,

- Residual heat removal system,
- RCS high point vents,
- Reactor water cleanup system (BWRs only).

Chapter 6. Engineered safety features

This chapter should present relevant information on the engineered safety features (ESFs) and associated systems. ESFs are provided to mitigate the consequences of postulated accidents (including severe accidents). The ESF systems provided in plant designs may vary. The ESF systems explicitly discussed in this chapter are those that are commonly used to limit the consequences of postulated accidents in light-water-cooled power reactors, and should be treated as illustrative of the ESF systems and of the kind of informative material that is needed. For these latter cases it should be agreed between the operating organization and the regulatory body which of the plant systems should be described in the SAR. The discussions on ESF designs should identify functional requirements, demonstrate how the functional requirements comply with regulatory requirements, and demonstrate how the ESF design meets or exceeds the functional requirements.

For each of the systems, the more detailed description should also include (see the Annex 2 for further details):

- The system function and safety design bases,
- The description of the system itself including design drawings,
- Material specifications, including nuclear, radiological, chemical, physical and mechanical properties,
- Interfaces with the other relevant systems,
- Monitoring, inspection, testing and maintenance provisions,
- Demonstration that all relevant functional requirements, requirements of industrial codes and standards, and regulatory requirements have been considered adequately,
- Performance and safety evaluation including demonstration that the system has sufficient capacity to fulfil its safety function.

6.1 Emergency core cooling system

This section should present relevant information on the emergency core cooling system and associated fluid systems. The actuation logic should be described subsequently in the section on protection systems in chapter 7 and need not be described here.

6.2 Containment systems

This section should present relevant information on the containment systems incorporated to localize the effects of accidents, and should include, among other things:

- the heat removal systems of the containment,
- the functional requirements of the primary and the secondary containment,
- the fission product removal and control systems,
- the containment isolation system,

- the protection of the containment against overpressure and underpressure, where provided,
- the control of combustible gases in the containment,
- the containment spray system, and
- the containment leakage testing system.

6.3 Habitability systems

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

6.4 Systems for the removal and control of fission products

This section should provide relevant information on the systems for the removal and control of fission products (if not already described as a part of the containment systems). In addition, the following specific information should be presented to demonstrate the performance capability of these systems: considerations of the coolant pH and chemical conditioning in all necessary conditions of system operation; effects on filters of postulated design basis loads due to fission products; and the effects on filter operability of design basis release mechanisms for fission products.

6.5 Other engineered safety features

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

Chapter 7. Instrumentation and control

This chapter should provide relevant information on the instrumentation and control (I&C) systems as described in Annex 2. The instrumentation senses the various plant parameters and transmits appropriate signals to the control systems during normal operation and to the reactor trip systems and engineered safety features and systems during AOOs and in accident conditions. The information provided in this chapter should emphasize those instruments and their associated equipment that constitute the protection systems and those systems relied upon by operators to monitor plant conditions and to shut the plant down and maintain it in a safe shutdown state after a design basis accident (DBA). Information should also be provided on the non-safety-related instrumentation and control systems used to control the plant in normal operation. These should be described for the purpose of demonstrating that their failure will not impair the proper operation of the safety related instrumentation and control systems or create challenges not already considered in the safety analysis of the plant.

7.1 I&C system architecture, functional allocation, and design bases

This introductory section should list all instrumentation, control, and supporting systems that are safety-related, including alarm, communication, and display instrumentation and should specify functions allocated to individual systems. Further on this sub-section should describe:

- Classification,
- I&C system design basis,
- Defence in depth and diversity strategy.

.7.2 Reactor Protection System

This section should provide relevant information on the reactor protection system following the structure described in Annex 2. In addition, specific information on the following specific aspects should be provided:

(a) The design bases for each individual reactor trip parameter with reference to the PIEs whose consequences the trip parameter is credited with mitigating.

(b) The specification of reactor trip system set points, time delays in system operation and uncertainties in measurement, and how these relate to the assumptions made in the chapter of the report on safety analyses.

(c) Any interfaces with the actuation system for engineered safety features (including the use of shared signals and parameter measurement channels).

(d) Any interfaces with non-safety-related instrumentation, control or display systems, together with provisions to ensure independence.

(e) The means employed to ensure the separation of redundant reactor trip system channels and the means by which coincidence signals are generated from redundant independent channels.

(f) Provisions for the manual actuation of the reactor trip system from both the main control room and the supplementary control room.

(g) Where reactor trip logic is implemented by means of digital computers, a discussion of the software design and quality assurance (QA) programmes, and the software verification and validation programme.

7.3 Actuation systems for engineered safety features

This part of the SAR should provide relevant information on the engineered actuation systems for safety features as described in Annex 2. In some plant designs the actuation systems for reactor trip and engineered safety features are designed as one single system. In this case it is appropriate to have a single section describing the actuation system for reactor trip and engineered safety features as one system.

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

(a) The design bases for each individual actuation system parameter for an engineered safety feature with reference to the PIE whose consequences the parameter is credited with mitigating; interfaces with the reactor trip system (including the use of shared signals and parameter measurement channels); interfaces with non-safety-related systems, together with provisions to ensure the proper isolation of electrical signals; and the means employed to ensure the physical separation of redundant actuation system channels for engineered safety features.

(b) Where the actuation logic for engineered safety features is implemented by means of digital computers, a discussion of the software design and QA programmes, and the software verification and validation programme.

(c) The specification of actuation system set points for engineered safety features, time delays in system operation and measurement uncertainties and how these relate to the assumptions made in the safety analyses chapter of the report.

(d) Provisions for equipment protective interlocks (e.g. pump and valve interlocks and motor protection) within the actuation system for engineered safety features, together with a demonstration that such interlocks will not adversely affect the operation of engineered safety features.

(e) Provisions for manually initiating engineered safety features from the main control room and the supplementary control room.

(f) Any relevant remote operator and/or automatic control, local control, on–off control or modulating control envisaged in the design and credited in the safety analysis.

7.4 Systems required for safe shutdown

This section should provide a description of the systems that are needed for safe shutdown of the plant, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, defense-in depth design features, and actuated devices. Any supporting systems should also be identified and described.

The major design considerations indicated in Annex 2 should be emphasized.

For remote shutdown capability the provisions to provide the required equipment outside the control room to achieve and maintain hot and cold shutdown conditions should be described. The design of remote shutdown stations should provide appropriate displays so that the operator can monitor the status of the shutdown. Access to remote shutdown stations should be under strict administrative controls. Logic diagrams, piping and instrumentation diagrams, and location layout drawings of all safe shutdown systems and supporting systems should be provided.

Analyses aimed to demonstrate how regulatory requirements have been complied with or justification should be provided for any deviation from meeting the agency's regulation. These analyses should include considerations of instrumentation installed to permit a safe shutdown in the event of the following:

- loss of plant instrument air systems,
- loss of cooling water to vital equipment,
- plant load rejection,
- turbine trip.

The analyses also should discuss the need for and method of changing to more restrictive trip set-points during abnormal operating conditions, such as operation with fewer than all reactor coolant loops operating.

7.5 Information systems important to safety

This subsection should provide relevant information on the systems for safety related display instrumentation and the computerized plant information system as described in Annex 2. In addition, specific information on the following should also be provided:

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

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

Further on, this section should provide relevant information on any other diagnostic and instrumentation systems required for safety, and should cover: any particular system needed for the management of severe accidents; leak detection systems; monitoring systems for vibrations and loose parts; and protective interlock systems that are credited in the safety analyses with preventing damage to safety related equipment and preventing accidents of certain types (e.g. valve interlocks at interfaces between low pressure and high pressure fluid systems whose operation could result in an intersystem LOCA).

7.6 Interlock systems important to safety

This section should contain information describing all other instrumentation systems required for safety that are not addressed in the sections describing the reactor protection system, ESF systems, safe shutdown systems, information system, or any of their supporting systems. These other systems include interlock systems to prevent over-pressurization of low-pressure systems when these systems are connected to high-pressure systems, interlocks to prevent over-pressurizing the primary coolant system during low-temperature operations, interlocks to isolate safety systems from non-safety systems, and interlocks to preclude inadvertent inter-ties between redundant or diverse safety systems for the purposes of testing or maintenance.

Relevant analyses should also be provided. These analyses should include, but not be limited to, considerations of the following interlocks:

- interlocks to prevent over-pressurization of low-pressure systems,
- interlocks to prevent over-pressurization of the primary coolant system during low-temperature operations of the reactor vessel,
- interlocks for ECCS accumulator valves,
- interlocks required to isolate safety systems from non-safety systems,
- interlocks required to preclude inadvertent inter-ties between redundant or diverse safety systems.

7.7 Control systems not important to safety

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

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

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

(c) A demonstration that such systems do not challenge the operation of safety related systems.

7.8 Diverse instrumentation and control systems

A description of the diverse I&C systems should be provided that includes initiating circuits, logic, bypasses, interlocks, redundancy, diversity, defense-in-depth design features, and actuated devices. This section should identify and describe supporting systems. Mitigation functions for anticipated transient without scram and address diverse manual controls and diverse display provisions should be provided. Logic diagrams, piping and instrumentation diagrams, and location layout drawings of all diverse I&C systems should also be included.

7.9 Data Communication Systems

This section should describe all data communication systems that are part of or support the other systems described in this chapter, addressing both safety and non-safety communication systems. Relevant layout drawings and network routing information should be included. The scope and depth of the system description will vary according to the system's importance to safety. Communication between systems and communication between computers within a system should be addressed.

The applicable criteria according to the importance to safety of the system should be addressed. The following major design considerations should be emphasized:

- the quality of components and modules,
- software quality,
- description how the performance requirements of all supported systems are met,
- the potential hazards to the system, including inadvertent actuations, error recovery, self-testing, and surveillance testing,
- unauthorized access control,
- redundancy and diversity requirements,
- independence requirements,
- fail safe design of the protection systems,
- system testing and surveillances,
- status of the data communication systems in the design of bypass and inoperable status indications,
- prevention of a fault propagation path for environmental effects (e.g., high-energy electrical faults and lightning) from one redundant portion of a system to another, or from another system to a safety system,
- defense-in-depth and diversity analyses for each potential failure mode,
- exposure of the system to seismic hazards.

Analysis should be provided to demonstrate that the data communication systems conform to the relevant recommendations in the regulatory guides and industry codes and in the standards applicable to these systems. The means and criteria for determining if a function has failed as a result of communications failure should be described.

7.10 Main control room

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

7.11 Supplementary control room

This section should provide an appropriate description of the supplementary control room, including the layout, with an emphasis on the human–machine interface. The electrical design standards for equipment signals routed to the supplementary control room have already been described in previous sections and need not be repeated here. The means of physical and electrical isolation between the plant systems and communication signals routed to the MCR and the supplementary control room should be described in detail to demonstrate that the supplementary control room is redundant and independent of the MCR. The mechanisms for the transfer of control and communications from the MCR to the supplementary control room should be described in detail so as to demonstrate how this transfer would occur under accident conditions.

7.12 Digital instrumentation and control systems application guidance

If digital instrumentation and controls systems are used, the overall scope of the application should include information on (1) the design qualification of digital systems, (2) protection against common-cause failure, and (3) functional requirements when implementing a digital protection system. The following topics should be addressed:

(1) The design criteria to be applied to the proposed system.

(2) The I&C design as applicable to the individual sections of this chapter.

(3) Defense in depth and diversity in a reactor trip system or an ESF actuation system as relevant for the combined ability of the I&C systems to cope with common-cause failure.

(4) Functional requirements and commitments.

(5) Life-cycle process planning (the computer system development process, particularly the software life-cycle activities for digital systems).

(6) Life-cycle process requirements, documenting the computer system functional requirements.

(7) Software life-cycle process design outputs, identifying also the documents to be developed for the regulatory review. The system test procedures and test results (validation tests, site acceptance tests, preoperational and start-up tests) should provide assurance that the system functions as intended.

For a system incorporating commercial-grade digital equipment, the preceding topics still apply. There should be evidence in the application of an acceptance process that has determined that there is reasonable assurance that the equipment will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a QA programme. The commercial-grade dedication process should be described in detail and include information on the original design and test of the commercial equipment.

Chapter 8. Electric power

Chapter 8 should provide relevant information on the electrical power systems.

8.1 General principles and design approach

In this section information on the following, specific to electrical systems, should be presented:

(a) The plant specific divisions of electrical power systems, including the differing system voltages and which parts of the system are considered to be essential.

(b) Substantiation of the functional adequacy of the safety related electrical power systems, including breakers, and assurance that these systems have adequate redundancy, physical separation, independence and testability in conformance with the design criteria. Electrical equipment protection, including the provisions to bypass this protection under accident conditions, should be described.

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

8.2 Offsite power systems

This section should provide information relevant to the plant on the off-site electrical power systems following the structure described in Annex 2. It should include a description of the off-site power systems, with emphasis on features for control and protection (breaker arrangements, manual and automatic disconnect switches) at the interconnection to the on-site power system. Special emphasis should be put on all design provisions used to protect the plant from off-site electrical disturbances and to maintain power supply to in-plant auxiliaries. Information on grid reliability should also be provided and any design specific provisions necessary to cope with frequent grid failures should also be described.

8.3 Onsite power systems

AC power systems

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

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

(a) In a design basis accident with a subsequent loss of off-site power the required engineered safety feature loads can be sequenced onto the emergency diesel generators without overloading the diesel generators and in time frames consistent with the assumptions presented in the chapter on safety analyses.

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

c) Non-interruptible AC power is continuously provided to essential safety systems and safety related instrumentation and control systems while normal off-site AC power systems are available and during postulated loss of off-site power events.

(d) The maximum frequency decay rate and the limiting underfrequency value for coastdown of the reactor coolant pumps are justified and the minimum number of engineered safety feature trains to be energized simultaneously (if more than two trains are provided) is ensured.

DC power systems

This subsection should provide relevant information on the DC power system as described in Annex 2. In addition, the following information on specific DC power systems should be provided: an evaluation of the long term discharge capacity of the battery (the projected voltage decay as a function of time without charging when subjected to design loads); the major DC loads present (including the non-interruptible AC power system inverters and any non-safety-related DC loads such as the lubrication oil pumps for the turbine bearings); and a description of the fire protection measures for the DC battery vault area and cable systems.

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

8.4 Cabling and raceways

In this section the cables and their raceways including cable supports, wall and floor penetrations, fire stops etc should be described in order to demonstrate that they are selected, rated and qualified for their service and for environmental conditions with account taken of the cumulative radiation effects and thermal ageing expected over their service life. Buses, cable trays and their supports should be designed to withstand, with an appropriate margin, the mechanical loads, including SL-2 earthquake loads. The buses and cables should also be sufficiently fire retardant to prevent the propagation of fires. At least three classes of cables should be identified: (1) control and instrumentation cables,

(2) low voltage power cables (e.g. 1000 V or less), and (3) medium voltage power cables (e.g. 20 kV or less). Special attention should be given to the qualification of cables that have to withstand conditions inside the containment during and after a LOCA, a main steam line break or other adverse environmental conditions.

8.5 Grounding and lightning protection

This section should provide description of the grounding and lightning protection (both internal and external protection) system, including the components associated with the various grounding subsystems (e.g., station grounding, system grounding, equipment safety grounding, any special grounding for sensitive instrumentation, and computer or low-signal control systems. Grounding and lightning protection plan drawings should be also included. The industry-recognized consensus standards used in designing the subsystems should be identified, as well as the bases for the related acceptance criteria. Analyses and any underlying assumptions used should be provided to demonstrate that the acceptance criteria for the grounding subsystems will be successfully incorporated into the as-built plant.

Chapter 9. Auxiliary systems and civil structures

Chapter 9 should provide in its first part A information about the facility's auxiliary systems unless such information is provided in other chapters in connection with other plant systems. In particular, this chapter should identify systems that are essential for safe shutdown of the plant or for protection of the health and safety of the public. For each system, the description should to the extent possible follow the structure given in Annex 2. For systems that have little or no role in protecting the public against exposure to radiation, the description should provide enough information to understand the design and operation and their effect on reactor safety, with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity. In addition, the information provided (e.g., a failure analysis) should clearly show the system's capability to function without compromising the safe operation of the plant under various plant conditions.

Other major part B of chapter 9 should describe civil structures of the plant. This part should describe how various civil structures in the plant comply with the general design requirements and other rules specified in chapter 3. Again, for each civil structure the description should to the extent possible to follow the structure of information given in Annex 2.

It is clear that both plant auxiliary systems as well as civil structures can vary between the designs. The examples of subsystems provided below are not therefore intended to represent a complete list of systems to be discussed in this chapter of the SAR. The structure of the chapter can be modified accordingly to the specificities of the design.

9A Auxiliary systems

9A.1 Fuel storage and handling systems

This section should provide relevant information on the fuel handling and storage systems. It should include details of the proposed arrangements for the shielding, handling, storage, cooling, transfer and transport of nuclear fuel. The following subsystems should be covered:

- New fuel storage and handling system,
- Spent fuel storage and handling system,
- Spent fuel pool cooling and cleanup system,
- Handling systems for refuelling.

For fresh fuel information should include details of the measures proposed to ensure that fresh fuel is maintained in a safe condition at all times. This should include considerations such as packaging, fuel accounting systems, storage, criticality prevention, fuel integrity control and fuel security.

For irradiated fuel information provided should include details of the measures proposed to ensure that irradiated fuel is maintained in a safe condition at all times. This should include considerations such as appropriate provisions for radiological protection, criticality prevention, fuel integrity control, including special provisions to deal with failed fuel, fuel chemistry, fuel cooling, fuel accounting systems, fuel security and arrangements for fuel consignment and transport.

9A.2 Water systems

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

- Service water system,
- Component cooling water system,
- De-mineralized water make-up system,
- Ultimate heat sink system,
- Condensate storage facilities,
- Potable and sanitary water systems.

9A.3 Process auxiliary systems

This section should provide relevant information on the auxiliary systems associated with the reactor process system structured in a format described in Annex 2. It should include, for example, information on the process and post-accident sampling systems and the equipment drainage and floor drainage systems. The compressed air systems are dealt with separately in another section of this chapter, the chemical control and volume control system was already covered in chapter 5.

9A.4 Air and gas systems

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

9A.5 Heating, ventilation, and air conditioning systems

This section should provide relevant information on the heating, ventilation, air conditioning (HVAC) and cooling systems in a format described in Annex 2. The following subsystems should be covered:

- Control room HVAC,
- Spent fuel pool area HVAC,
- Auxiliary and radwaste area HVAC,
- Turbine building HVAC,
- Engineered safety feature HVAC,
- Chilled water system.

9A.6 Fire protection systems

This section should provide relevant information on the fire protection systems following the structure described in Annex 2. It should justify the provisions made to ensure that the plant design provides adequate fire protection. The design should include adequate provisions for defence in depth in the event of a fire, and should provide fire prevention, fire detection, fire warning, fire suppression and fire containment. Consideration should be given to the selection of materials, the physical separation of redundant systems, the seismic qualification of equipment and the use of barriers to segregate redundant trains.

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

9A.7 Emergency diesel generator and supporting systems

Supporting systems for the emergency diesel generator should be covered by this section. Electrical part of the system has been already covered in chapter 8. The following subsystems should be addressed in this section:

- Diesel generator fuel oil storage and transfer system,
- Diesel generator cooling water system,
- Diesel generator starting air system,
- Diesel generator lubrication system,
- Diesel generator combustion air intake and exhaust system.

9A.8 Miscellaneous auxiliary systems

This section should provide relevant information on any other plant auxiliary system whose operation may influence plant safety and that has not been covered in any other part of the SAR; for example, the communication systems, the lighting and emergency lightning systems.

9A.9 Overhead heavy-load handling system

The overhead heavy-load handling system with respect to critical load handling evolutions should be described in this section. Critical load handling evolutions are those handling evolutions with the potential for inadvertent operations or equipment malfunctions to affect the handling system in the following ways:

- cause a significant release of radioactivity,
- cause a loss of margin to criticality,
- uncover irradiated fuel in the reactor vessel or spent fuel pool,
- damage equipment essential to achieve or maintain safe shutdown.

Necessary information includes parameters defining the load that, if dropped, would cause the greatest damage; the areas of the plant where the load would be handled; the design of the overhead heavy-load handling system; and the operating, maintenance, and inspection procedures applied to the load handling system. The following systems should be described in particular:

- Reactor building crane,
- Fuel building crane.

9B Civil works and structures

This part of the SAR should describe how general design requirements specified in section 3.5 have been complied with in the design of plant specific structures. Three groups of civil structures should be considered: foundations, reactor building, and all other civil structures. In description of the structures, the unified format of the information provided (specified in Annex 2) should be followed to the extent possible. In particular, the following information specific to civil engineering works and structures should also be provided:

(a) Details of the range of anticipated structural loadings, together with the defined performance requirements of the buildings and structures and the consideration given to hazards in the design.

(b) A description of the extent to which load–source interactions have been considered, with a confirmation of the ability of the buildings and structures to withstand the required load combinations while fulfilling their safety functions.

(c) If a safety and/or seismic classification system for buildings and structures has been used, the basis of the classification should be described for the design option outlined. It should be demonstrated that the safety classification of buildings containing equipment important to safety is commensurate with the classification of the systems, components and equipment that it contains.

(d) If a building structure or wall is intended to provide separate functions from its structural function (e.g. functions of radiation shielding, separation and containment), the additional requirements identified for these functions should be specified and reference should be made to other sections of the SAR, as appropriate.

9B.1 Foundations

In this section information on foundations should be provided, including plan and section views of each foundation, to define the primary structural aspects and elements relied on to perform the foundation function. The description should include the relationship between adjacent foundations, including any separation and the reasons for such separation. The type of foundation and its structural characteristics and provide the general arrangement of each foundation should be discussed, with emphasis on the methods of transferring horizontal shears, such as those that are seismically induced, to the foundation media. In particular, foundations of steel or concrete containment should be discussed, as well as all seismic Category I structures.

9B.2 Reactor building

This section should describe design features of the reactor building provided to comply with the applicable safety requirements. Specific design features of the primary containment such as its leaktightness, mechanical strength, pressure resistance and resistance to hazards should be covered. Concrete and steel internal structures of the containment should be described. If the design incorporates a secondary containment, this should also be described here.

9B.3 Other structures

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

Chapter 10. Steam and power conversion systems

Chapter 10 should provide relevant information on the plant steam and power conversion system. Information specific to steam and power conversion systems should also be provided on the following, where appropriate:

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

(b) A description of the main steamline piping and the associated control valves, the main condensers, the main condenser evacuation system, the turbine gland sealing system, the turbine bypass system, the circulating water system, the condensate cleanup system, the condensate and feedwater system, and, where applicable, the steam generator blowdown system. Also to be provided is a description of the water chemistry programme, together with a discussion of the materials of the steam, feedwater and condenser systems.

The chapter should emphasize those aspects of the design and operation that affect or could potentially affect the reactor and its safety features or contribute toward the control of radioactivity. The information provided should show the capability of the system to function without compromising (directly or indirectly) the safety of the plant, under both normal operating and transient situations. In addition, the chapter should include a discussion of how the system design meets the applicable regulatory requirements and is consistent with the applicable regulatory guidance. Where appropriate, this chapter should summarize the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems.

In describing individual parts of the steam and power conversion system, the information provided should follow the structure specified in Annex 2.

10.1 Role and general description

In this section a summary description indicating principal design features of the steam and power conversion system should be provided. This description should include an overall system flow diagram and a summary table of the important design and performance characteristics, including a heat balance at rated power and at stretch power and indicate safety-related system design features.

10.2 Main steam supply system

In this section the main steam supply system and main steamline piping should be described including P&IDs showing system components, including interconnected piping. On the P&IDs the physical division between the safety-related and nonessential portions of the system should be indicated.

The main steam supply system consists of the components, piping, and equipment that function to transport steam from the NSSS to the power conversion system and various safety-related and nonsafety-related auxiliaries. For the BWR direct cycle plant, the main steam system extends from the outermost containment isolation valves up to and including the turbine stop valves and includes connected piping of larger diameters, up to and including the first valve that is ether normally closed or is capable of automatic closures during all modes of reactor operation. For the PWR plants, the main steam system extends from the connections to the secondary sides of the steam generators up to and including the turbine stop valves and includes the containment isolation valves, safety and relief valves, connected piping of larger diameters, up to and including the first valves that are either normally closed or capable of automatic closure during all modes of operation, and the steamline to the auxiliary feedwater pump turbine.

10.3 Feedwater systems

Both main and auxiliary feedwater systems should be described in this section, including the capability to supply adequate feedwater to the NSSS, criteria for isolation from the steam generator or RCS, supply of condensate available for emergency purposes, and environmental design requirements. An analysis should be included of component failure and of the effects of equipment malfunction on the RCS and an analysis of detection and isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break and/or degradation of the integrity of safety-related equipment.

For PWRs, the following information with reference to fluid flow instabilities (e.g., water hammer, for steam generators using top feed) should be provided:

(1) Describe normal operating transients that could cause the water level in the steam generator to drop below the sparger or cause the nozzles to uncover and allow steam to enter the sparger and feedwater piping.

(2) Provide a summary of the criteria for routing or isometric drawings showing the routing of the feedwater piping system from the steam generators to the restraint that is closest, on the upstream side, to the feedwater isolation valve that is outside containment.

(3) Describe the piping system analyses, including any forcing functions, or the result of test programmes performed to verify that uncovering of feedwater lines could not occur or that such uncovering would not result in unacceptable damage to the system.

10.4 Turbine generator

In this section, the turbine generator (TG) system, associated equipment (including moisture separation), use of extraction steam for feedwater heating, and control functions that could influence operation of the RCS should be described in this section In addition, piping and instrumentation diagrams (P&IDs) and layout drawings that show the general arrangement of the TG system and associated equipment with respect to essential safety-related SSC should be provided. Information to demonstrate the structural integrity of turbine rotors and the protection against damage to a safety-related component due to failure of a turbine rotor that produces a high-energy missile should be provided.

This section should describe the TG system equipment design and design bases, including the performance requirements under normal, upset, emergency, and faulted conditions. It should also describe the intended mode of operation (base loaded or load following), functional limitations imposed by the design or operational characteristics of the RCS (e.g., the rate at which the electrical load may be increased or decreased with and without reactor control rod motion or steam bypass), and design codes to be applied. The information provided should include the seismic design criteria, the bases for the chosen criteria, and the seismic and quality group classifications for TG system components, equipment, and piping.

10.5 Turbine and condenser systems

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

- Main condenser,
- Condenser air extraction system,
- Circulating water system,
- Condensate system,
- Condensate cleanup system,
- Turbine auxiliary systems (turbine gland sealing system, turbine bypass system),
- Generator auxiliary systems.

10.6 Steam generator blowdown system

The steam generator blowdown system and its design basis should be described in this section in terms of its ability to maintain optimum secondary-side water chemistry in recirculating steam generators of PWRs during normal operation, including AOO (e.g., main condenser in-leakage, primary-to-secondary leakage). The design bases should include consideration of expected and design flows for all modes of operation (i.e., process and process bypass), process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process I&C for maintaining operations within established parameter ranges.

10.7 Break preclusion implementation for main steam and feedwater lines

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

Chapter 11. Radioactive waste management

This chapter should justify the adequacy of the measures proposed for the safe management of radioactive waste of all types that is generated throughout the lifetime of the plant. More specifically, the chapter should describe:

1. The capabilities of the plant to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials, and

2. The instrumentation used to monitor the release of radioactive wastes.

The information should cover normal operation, including AOOs (refuelling, purging, equipment downtime, maintenance, etc.).

For all kinds of waste (liquid, gaseous, solid) their control, handling, conditioning, storage and disposal should be described as follows:

Control of waste; Measures to control or contain the waste produced at all stages of the lifetime of the plant should be described, including proposals to categorize and separate waste, as necessary.

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

Conditioning of waste; Measures to condition the waste produced at all stages of the lifetime of the plant should be described. Where it is considered prudent, waste may be processed in accordance with established procedures, and the options considered should be described here. However, consideration should also be given to establishing the most suitable option that, to the extent possible, does not foreclose alternative options, in case preferences for waste disposal change over the lifetime of the plant.

Storage of waste; Measures to store the waste produced at all stages of the lifetime of the plant should be described. The quantities, types and volumes of radioactive waste and the need to categorize and separate waste within the provisions for storage should be considered. The possible need for specialized systems to deal with issues of long term storage, such as cooling, containment, volatility, chemical stability, reactivity and criticality, should also be addressed, and any such system in place should be described.

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

The sections 11.2 – 11.5.should provide relevant information on the radioactive waste treatment systems. It should include the design features of the plant that safely control, collect, handle, process, store and dispose of solid, liquid and gaseous forms of radioactive waste arising from all activities on the site throughout the lifetime of the plant. This should include the structures, systems and components provided for these purposes and also the instrumentation incorporated to monitor for possible leaks or escapes of radioactive waste. The potential for radioactive waste to be adsorbed and/or absorbed should be considered in deciding on the measures necessary to deal with this hazard. Scope and structure of the description for all systems should follow the common structure specified in Annex 2.

11.1 Source terms

A description of the main sources of solid, liquid and gaseous waste and estimates of their generation rate in compliance with the design requirements should be provided. The section should also provide information on the characteristics of the accumulation rates and the quantities, conditions and forms of radioactive waste with different states of aggregation and activity level, for normal and abnormal conditions of operation and for accident conditions, and on the methods and technical means for its processing and/or conditioning, storage and transport. The consideration of waste should cover solid, liquid and gaseous waste, as appropriate, in all stages of the development of measures to deal with radioactive waste safely throughout the lifetime of the plant. This section should consider the options for the safe predisposal management of waste.

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

11.2 Liquid waste management systems

This section should describe the capabilities of the plant to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including AOOs.

11.3 Gaseous waste management systems

This section should describe the capabilities of the plant to control, collect, process, handle, store, and dispose of gaseous radioactive waste generated as the result of normal operation and AOOs.

11.4 Solid waste management systems

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

11.5 Process and effluent radiological monitoring and sampling systems

This section should describe the systems that monitor and sample the process and effluent streams in order to control releases of radioactive materials generated as the result of normal operations, including AOOs, and during postulated accidents.

Chapter 12. Radiation protection

This chapter should provide information on the policy, strategy, methods and provisions for radiation protection. The expected occupational radiation exposures during normal operation and AOOs, including measures to avoid and restrict exposure, should also be described.

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

12.1 ALARA considerations

This section should provide a description of the operating organization's policy and the operational application of the ALARA (as low as reasonably achievable) principle. It should be in line with the conceptual description, and should demonstrate that the recommendations for the application of the ALARA principle have been followed.

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

12.2 Radiation sources

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

12.3 Radiation protection design features

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

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

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

(b) The occupational exposures in the course of normal operation shall be ALARA;

(c) Dose constraints shall be used to avoid inequities in the dose distributions;

(d) Measures shall be taken to prevent any workers from receiving doses near the dose limits year by year;

(e) All practicable steps shall be taken to prevent accidents with radiological consequences;

(f) All practicable steps shall be taken to minimize the radiological consequences of any accident.

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

12.4 Dose assessment

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

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

12.5 Operational radiation protection programme

This section should describe the administrative organization, the equipment, instrumentation and facilities, and the procedures for the radiation protection programme. It should be demonstrated that, the plant radiation protection programme is based on a prior risk assessment that takes into account the location and magnitude of all radiation hazards, and covers:

(a) Classification of work areas and access control;

(b) Local rules and supervision of work;

(c) Monitoring of individuals and the workplace;

(d) Work planning and work permits;

(e) Protective clothing and protective equipment;

(f) Facilities, shielding and equipment;

(g) Health surveillance;

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

(i) Source reduction;

(j) Training;

(k) Arrangements for response to emergencies.

Chapter 13. Conduct of operations

This chapter should contain a description of the important operational issues relevant to safety throughout the lifetime of the plant and should also present the operating organization's approaches to address the identified issues adequately. The chapter should provide assurance that the applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the licensee are adequate to protect public health and safety.

13.1 Organizational structure of operating organization

This section should provide a description of the arrangements of the operating organization and specify the functions and responsibilities of the different components within it. The organization and responsibilities of review bodies (e.g. safety committees and advisory panels) should also be described. The description should cover the organizational structure, its functions and responsibilities, the number and the qualifications of personnel, and should be directed to activities that include facility design, design review, design approval, construction management, testing, and operation of the plant. The description of the organizational structure should demonstrate that all the management functions for the safe operation of the power plant, such as policy making functions, operating functions, supporting functions and reviewing functions, are adequately addressed.

This section should also identify qualification requirements for key staff, which should be described in terms of educational background and experience requirements, for each identified position or class of positions.

13.2 Training

3.168. This section should provide justification that the training programme for plant staff is adequate to achieve and maintain the required level of professional competence of staff throughout the lifetime of the plant. Information should be provided to describe the staff training programme, including refresher training and retraining, and also the applicable documentation system for recording the present positions for plant staff. Training programmes and facilities, including simulator facilities, should reflect the status, characteristics and behaviour of the plant units, and should be briefly described.

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

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

13.3 Operational programme implementation

Operational programmes are specific programmes that are required by regulations. This section of the SAR should sufficiently describe such programmes and provide the schedule for implementation of the programmes. Typical operational programmes are described below:

Maintenance, surveillance, inspection and testing

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

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

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

The operating organization should also identify all testing that can affect the safety functions of a NPP. This should include, in addition to a schedule of identified testing, a system for ensuring that testing is initiated, carried out and confirmed within the timescales allowed. This section should also refer to methods for the audit and review of the testing identified.

Core management and fuel handling

The SAR should demonstrate that the operating organization makes the necessary arrangements for all operational activities associated with core management and fuel handling to ensure the safe use of the fuel in the reactor and safety in its transport and storage on the site. It should be shown that, for each batch refuelling, tests are performed to confirm that the core performance meets the design intent. It should also be shown that the core conditions are monitored and compared with predictions to determine whether they are as expected and are within operational limits. In addition, it should be shown that criteria have been established and procedures established for dealing with failures of fuel rods or control rods, so as to minimize the amounts of fission products and activation products in the primary coolant or in gaseous effluents.

Management of ageing

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

Control of modifications

The operating organization should describe the proposed method of identifying, controlling, planning, executing, auditing, reviewing and documenting the necessary modifications to the plant throughout its lifetime. This should take account of the safety significance of the proposed modifications to allow them to be graded and referred to the regulatory body, where necessary. The modification control process should cover the changes made to the plant systems and components, OLCs, plant procedures and process software. It should also be demonstrated that the modification control covers permanent and temporary changes to the plant. When a proposed modification would affect the performance of the operators or the operating organization, it should be demonstrated that provisions are in place to ensure that the principles of human factors engineering are considered and applied throughout the design and implementation of the modifications. Records of all modifications should be retained, and, where necessary, all documentation, procedures, instructions and drawings should be routinely revised to reflect these changes. It should also be demonstrated that the requirements for configuration management are met in the implementation of the plant modifications.

Programme for the feedback of operational experience

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

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

This section of the SAR should demonstrate the suitability of the proposed system for feedback of operational experience for the purpose of analysing the root causes of equipment failures and human errors, improving job descriptions and operational procedures, and assessing the need for backfitting and modernization of the plant, including organizational changes, if necessary.

Documents and records

The operating organization should provide details here of the provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and deleting documents and records that relate to the operational activities over the lifetime of the plant. In particular, this should include the operator’s documentary provisions for the management of plant configuration, as well as the management of waste and decommissioning of the plant.

Outages

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

13.4 Plant procedures

This section should describe administrative and operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the SAR is not expected to include detailed written procedures. Depending on the stage of the project, the SAR should either provide preliminary schedules for their preparation, or should provide a brief description of the nature and content of the procedures and a schedule for their preparation. Four categories of the procedures should be covered as described below.

Administrative procedures

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

Operating procedures

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

Emergency operating procedures

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

Accident management guidelines

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

13.5 Security

Security issues are usually dealt with separately according special regulations and materials are withheld from public disclosure. This section of the SAR should note that the applicant's plans for physical protection of the facility are described in a separate part of the application. Optionally, a short description of the security programme for the site and the implementation schedule for the programme can be provided.

Chapter 14. Plant commissioning

The operating organization should demonstrate that the plant will be suitable for service prior to its entering the operational phase. The process that the operating organization has adopted to demonstrate this suitability should be presented here. The operating organization should describe the tests intended to validate the plant's performance against the design prior to the operation of the plant. For this purpose a well planned, controlled and properly documented commissioning programme should be prepared and made ready for implementation. The proposal of the commissioning programme should be presented in this chapter of the SAR. A clear link from the plant safety justification to the commissioning programme should be demonstrated. The commissioning programme should, among other things, confirm that the separate plant items will perform within their specifications and that in the various safety systems they function together to ensure that the system's safety functions are reliably performed. In addition, the operating procedures should be validated to the extent practicable as part of the commissioning programme, with the participation of the future operating personnel.

This chapter should also present the details of the commissioning organization, including the appropriate interfaces between design, construction and operating organizations during the commissioning period, which should include any provisions for additional personnel and their interactions with the commissioning organization. It should also be shown that sufficient numbers of qualified operating personnel at all levels will be directly involved in the commissioning process. The processes established to develop and approve test procedures, to control test performance and to review and approve test results should be described in detail. This should include the process to be followed when the initial outcomes of the tests do not fully meet the design requirements.

14.1 Specific information to be included in SAR prior to construction

Specific information to be included in the SAR prior to plant construction should include:

- Description of the major phases of the initial test programme and discussion of the overall test objectives and general prerequisites for each major phase,
- A summary description of preoperational and/or start-up testing planned for each unique or first-of-a-kind principal design feature including the test method and test objectives,
- The applicant's plans for using guidance in applicable regulatory guides in the development and conduct of the initial test programme,
- The applicant's plans for the utilization of available information on reactor plant operating experiences to establish where emphasis may be warranted in the test programme,
- A summary description on the overall schedule, relative to the expected fuel loading date, for developing and conducting the major phases of the test programme,

- The applicant's plans pertaining to the trial use of plant operating and emergency procedures during the period of the initial test programme,
- The applicant's general plans for the assignments of additional personnel to supplement his plant operating and technical staff during each major phase of the test programme.

14.2 Specific information to be included in SAR prior to commissioning

Specific information to be included in the SAR prior to plant commissioning should include updated information on:

- Description of the major phases of the test programme and the specific objectives to be achieved for each major phase,
- Description of the applicant's organizational units and any augmenting organizations or other personnel that will manage, supervise, or execute any phase of the test programme,
- Description of the system that will be used to develop, review, and approve individual test procedures, including the organizational units or personnel that are involved and their responsibilities,
- Description of the administrative controls that will govern the conduct of each major phase of the test programmes,
- The measures to be established for the review, evaluation, and approval of test results for each major phase of the programme,
- The applicant's requirements pertaining to the disposition of test procedures and test data following completion of the test programme,
- The list all regulatory guides applicable to initial test programmes that will be used or alternative methods along with justification for their use,
- Information on the programme for utilizing available information on reactor operating experiences in the development of the initial test programme should be described, including identification of the organizations participating in the, their roles in the programme, and a summary description of their qualifications,
- The schedule for development of plant procedures as well as a description of how, and to what extent, the plant operating and emergency procedures will be use-tested during the initial test programme,
- Description of the procedures that will guide initial fuel loading and initial criticality, including the safety and precautionary measures to be established for safe operation,
- The schedule, relative to the fuel loading date, for conducting each major phase of the test programme,

- Test abstracts for all tests that will be conducted during the initial test programme ; emphasis should be placed on system and design features that **(1)** are relied on for the safe shutdown and cooldown of the facility under normal and faulted conditions, (2) are relied on for establishing conformance with limits or limiting conditions for operation that will be established by the technical specifications, and (3) are relied on to prevent or to limit or mitigate the consequences of anticipated transients and postulated accidents.

Chapter 15. Safety analysis

This chapter should provide a description of the results of the safety analyses performed to assess the safety of a plant in response to PIEs on the basis of safety criteria and authorized limits on radioactive releases. These analyses include deterministic safety analyses used in support of normal operation, analyses of AOOs, design basis events, beyond design basis events and selected severe accidents, and PSAs. The description may be supported by reference material, where necessary. The level of details provided in this chapter should develop with increasing stages of the project.

The information provided in the chapter on safety analyses should be sufficient to justify and confirm the design basis for the items important to safety, and to ensure that the overall plant design is capable of meeting the established acceptance criteria, in particular the authorized limits for radiation doses and radioactive releases for each category of plant conditions.

All analyses demonstrating comprehensively plant safety should be preferably addressed in this single chapter. If this is not the case, i.e. if certain analyses are placed in other chapters of the SAR, then proper reference to those chapters should be made here.

15.1 General considerations

This section provides an introduction to the chapter on safety analysis. The scope of safety analysis and the approach adopted should be described here, individually for various plant states both within and beyond the design basis conditions, including internal and external hazards. Any applicable reference documents used in safety analysis should also be introduced here. Due to large complexity of the chapter it is appropriate to describe the structure of the whole chapter in this section.

15.2 Safety objectives and acceptance criteria

This section should briefly describe how safety analysis refer to the principles and objectives of nuclear safety, radiation protection and technical safety applicable to the particular plant design, as previously identified in the chapter on general design aspects.

In particular both high level radiological acceptance criteria as well as derived (detailed) acceptance criteria specific to structures, systems and components for different classes of events and types of analyses should be specified. These criteria should not only take into account different classes of events (including hazards) according their frequencies but also different safety aspects (challenges to the barriers against releases of radioactivity) of the same events.

The specification of the acceptance criteria should be well justified and documented in this part of the SAR. The range and conditions of applicability of each specific criterion should be clearly specified (e.g. dependency on the fuel burn-up, use of specific correlation or methodology for demonstration of the compliance with the criterion).

15.3 Identification and classification of PIEs

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

Events should be classified in accordance with their anticipated frequencies and grouped according their types. The purpose of this classification is: (a) To justify the basis for the range of events under consideration; (b) To reduce the number of initiating events requiring detailed analysis to a set that includes the most bounding cases in each of the various event groups credited in the safety analyses, but that does not contain events with identical system performance (such as in terms of timing, plant systems response and radiological release fractions); (c) To allow for differing acceptance criteria for the safety analyses to be applied to differing event classes.

The basis for event classification should be described and justified.

Typically the list of PIEs to be addressed in the SAR will cover AOOs and design basis accidents. It should also include results from the analysis of beyond design basis accidents performed. Some of the design basis accidents or beyond design basis accidents may further develop, if additional faults are assumed, and lead to severe accidents involving significant core degradation and/or off-site radioactive releases. The results of severe accident analyses should also be included in the SAR to the extent that they are needed to demonstrate compliance with the acceptance criteria (if applicable), as well as needed for plant or system design or for development of the plant accident management programme and to support emergency preparedness.

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

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

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

15.4 Human actions

This section should describe and justify in general the approaches adopted to take into account human actions in the different types of safety analyses and the methods selected to model these actions in each type of analysis. Differences in approaches to consideration of human actions between deterministic and probabilistic analyses should be described.

15.5 Deterministic analyses

In this section of the SAR all the deterministic analyses performed to evaluate and justify plant safety should be considered. Deterministic safety analysis predicts the plant response to PIEs in specific predetermined operational states. It applies specific rules and uses specific acceptance criteria. The analyses typically focus on neutronics and thermal-hydraulic, structural and radiological aspects that are analysed with different computational tools.

15.5.1 General description of the approach

In general, the deterministic analysis for design purposes should be conservative, i.e. to ensure sufficient safety margins. The analysis of beyond design basis accidents is generally less conservative than that of design basis accidents. It is acceptable that best estimate codes are used for deterministic analyses provided that they are either combined with a reasonably conservative selection of input data or associated with the evaluation of the uncertainties of the results. The SAR should describe how conservatism in safety analysis has been ensured.

The models and the computer codes used for the deterministic analyses as well as the general assumptions made concerning plant parameters, the operability of systems, including control systems, and the operators' actions (if any) in the events should be described. Sufficiently detailed plant data used for development of the plant models should be provided in order to provide for independent verification of safety analysis. Important simplifications made should be justified. The set of limiting assumptions for safety analysis used in the deterministic safety analyses performed for the different types of PIEs should be described in this section.

A general summary of the verification and validation processes used for the computer codes should be presented, with reference to more detailed topical reports. Any computer programmes used should be identified with reference to the supporting documentation. Emphasis should be given to the substantiation of the applicability of the computer programme to the particular event, and reference should be made to the validation documentation, which should refer to relevant supporting experimental programmes and/or actual plant operating data. The validation status of the plant model should also be presented.

Any general guidelines for the analysis (such as on the choice of operating states of systems and/or support systems, conservative time delays and operator actions) used in setting up the methods and models used to demonstrate acceptability in the deterministic safety analyses should be described.

15.5.2 Safety in normal operation

This section should demonstrate that the normal operations of the plant can be carried out safely and hence confirm that

- radiation doses to workers and members of the public and planned discharges and/or releases of radioactive material from the plant are within the authorized limits.
- Plant parameters are maintained within the boundaries specified by the plant limits and conditions.

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

(a) Normal reactor startup from shutdown, to criticality, to full power;

(b) Power operation, including full power and low power operation;

(c) Changes in reactor power, including load follow modes and return to full power after an extended period at low power, if applicable;

(d) Reactor shutdown from power operation;

(e) Hot shutdown;

(f) The cooling down process;

(g) Refuelling during normal operation, where applicable;

(h) Shutdown in a refuelling mode or another maintenance condition that opens the reactor coolant or containment boundary;

(i) Handling of fresh and irradiated fuel.

15.5.3 Analysis of individual groups of PIEs

This section should provide a description of the results of the analyses of AOOs and design basis accidents performed to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety systems. The analyses should cover events initiated during all normal operational states, including low power and shutdown modes. All potential sources of radiological hazards should be considered, including spent fuel pools or radwaste treatment systems. Not only internal initiating events, but also internal and external (both natural as well as man-made) hazards should be covered.

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. Those plant parameters important to the outcome of the safety analysis should be identified. These would typically include: reactor power and its distribution; core temperature; cladding oxidation and/or deformation; pressures in the primary and secondary system; containment parameters; temperatures and flows; reactivity coefficients; reactor kinetics parameters; and the worth of reactivity devices.

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

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

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

(a) *PIE:* A description of the PIE, the class to which the PIE belongs and the acceptance criteria to be met.

(b) *Accident boundary conditions:* A detailed description of the plant operating configuration prior to the occurrence of the PIE, the model specific and event specific assumptions, and the computer codes used. A description should also be included of systems and operator actions that are credited in the analysis, such as:

(i) Normally operating plant systems and support systems;

(ii) Normally operating plant instrumentation and controls;

(iii) Plant and reactor protection systems;

(iv) Engineered safety systems and their actuation set points;

(v) Operator action, if any.

(c) *Initial plant state:* Specific values of important plant parameters and initial conditions used in the analysis; these may be presented in a table. An explanation should be provided of how these values have been chosen and the degree to which they are conservative for the specific PIE being analysed.

(d) *Identification of additional postulated failures:* A discussion of any additional single failure postulated to occur in the accident scenario and a justification of the basis for selecting it as the limiting single failure.

(e) *Plant response assessment:* A discussion of the modelled plant behaviour, highlighting the timing of the main events (initial event, any subsequent failures, times at which various safety groups are actuated and time at which a safe long term stable state is achieved). Individual system actuation times, including the reactor trip time and the time of operator intervention, should be provided. Key parameters should be graphically presented as functions of time during the event. The parameters should be selected so that a complete picture of the event’s progression can be obtained within the context of the acceptance criterion being considered. For example, in evaluating fuel cladding temperatures, parameters such as power, heat flux, pressure of the RCS, fluid inventories of the RCS, fuel temperatures and flow rates in the emergency core cooling system should be given, where appropriate to the type and design of the reactor. The results should present the relevant plant parameter and a comparison with the acceptance criteria, with a final statement on the acceptability of the result. The status of the physical barriers and the fulfilment of the safety functions should also be discussed.

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

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

15.5.4 Consideration of design capability for beyond design basis accidents

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

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

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

(a) The objective of the analysis of beyond design basis events and/or the specific acceptance criteria should be stated.

(b) A discussion of the additional postulated failures in the accident scenario should be provided, together with a discussion of the basis for their selection.

(c) Whenever operator action is taken into account, it should be demonstrated that the operators will have reliable information, sufficient time to perform the required actions and procedures to follow, and will have been trained. The key results should be compared with the specific acceptance criteria, and the conclusions on meeting the acceptance criteria should be clearly stated.

15.5.5 Severe accidents

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

The severe accident analysis should generally be carried out using best estimate assumptions, data, methods and decision criteria. Nevertheless reasonably conservative assumptions should be made which take account of the uncertainties in the understanding of the physical processes being modelled and in interpretation of the results in terms of predicted timing and severity of phenomena.

Another issue is connected with assumptions regarding operability of plant systems in case of severe accidents. Consideration of operability of all plant systems even beyond their normal operating range is usually recommended and acceptable for development of severe accident management guidelines, but is very complicated to rely on survivability of systems in demonstrating acceptability of the plant design. In addition, majority of systems would not be available due to complete lack of normal and emergency power supply. It is therefore advisable to demonstrate acceptability of the design using only systems dedicated to severe accident mitigation.

In addition to demonstration of the acceptability of the design, the results of the most relevant severe accident analyses used in the development of the accident management programmes and emergency preparedness planning for the plant should be specified and presented in this section. The accident management measures that could be carried out to mitigate the accidents' effects, and also to provide input for emergency planning and preparedness, should have been identified and optimized in the severe accident analysis. Reference should be made to those relevant chapters of the SAR in which these results are used.

15.6 Probabilistic analyses

An integrated review of the plant design and operational safety should be used to complement the results of the deterministic analyses and to give an indication of the success of the deterministic design in achieving the design objectives. One possible means of undertaking an integrated review is through the use of a PSA. This section should provide a description of the scope of the PSA study, the methods used and the results obtained, covering both Level 1 and Level 2 studies, as applicable. If any quantitative probabilistic safety criteria or goals have been used in the development of the plant design (as mentioned in the section of the SAR on probabilistic design criteria), these should also be referred to here.

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

(a) Justification of the selected scope of the PSA study;

(b) Accident sequence modelling, including event sequence and system modelling, human performance analysis, dependence analysis and classification of accident sequences into plant damage states;

(c) Data assessment and parameter estimation, including the assessment of the frequency of initiating events, component reliability, common cause failure probabilities and human error probabilities;

(d) Quantification of accident sequences, including uncertainty, importance and sensitivity analyses;

(e) Source term analysis and assessment of off-site consequences.

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

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

15.7 Summary of results of the safety analyses

This section should provide a summary of the overall results of the safety analyses, individually for each category of the events and covering both deterministic and probabilistic analysis. It should be confirmed that the requirements of the analyses have been met in every respect, providing justification if requirements have been changed, and clearly justifying where requirements have not been met entirely or have been changed as a result of further considerations. In the latter case any compensatory measures taken to meet the safety requirements should be specified.

Chapter 16. Operational Limits and Conditions (Technical Specifications)

The OLCs form an important part of the basis on which the operating organization is authorized to operate the plant. The OLCs can either be presented as part of the SAR; or they are prepared as a separate document that is referenced in the SAR.

16.1 Use and application

This section should describe the way how the OLCs have been developed, their scope and range of applicability. The licensing process should generally include a consideration of the OLCs in the form of controls, limits, conditions, rules and required actions that are formally derived from the safe operating envelope. The safe operating envelope is encompassed by the possible operating states included in the establishment of the design basis. This is to ensure that the operation of the plant will not present an intolerable risk to the health and safety of workers or the public, operation being at all times within the safe operating regime established for the plant. The OLCs should provide clear and unambiguous instructions to operators that are clearly linked to the safety justification for the plant.

16.2 Safety limits

If included in the SAR, detailed OLCs for operation should contain numerical values of limiting parameters and operability conditions of systems and components.

16.3 Limiting conditions for operation, protection thresholds, actions, and surveillance requirements

The corresponding requirements for surveillance, maintenance and repair to ensure that these parameters remain within acceptable limits and that systems and components are operable should also be specified and described in this section. Where appropriate such requirements should be justified by means of a PSA. The actions to be taken in the event that operational limits and conditions are not fulfilled should also be clearly established.

16.4 Administrative requirements

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

16.5 Bases

In this section it should be demonstrated that the OLCs have been developed in a systematic way. In particular, the OLCs should be based on the safety analyses of the plant and its environment in accordance with the provisions made in the design. The OLCs should be determined with due account taken of the uncertainties in the process of safety analysis. The justification for each of the OLCs should be substantiated by means of a written indication of the reason for its adoption and any relevant background information. Amendments should be incorporated as necessary as a result of testing carried out during commissioning.

Chapter 17. Management systems

17.1 General considerations

This section should describe the operating organization's overall management structure and introduces management of safety as an integral component of the management of the entire organization. This should include the roles of on-site safety assessment organizations and any off-site safety advisory committees that will advise the operating organization's management. The aim is to demonstrate that the operating organization will be able to fulfil its responsibility to operate the plant safely throughout its operating lifetime.

The framework and components of the system for management of safety should be described.

17.2 Specific aspects of management of safety processes

The site management structure and technical support organization of the operator should be described in this section. The way in which effective management control of the design and operating organizations will be achieved so as to promote safety, as well as the measures employed to ensure the implementation and observance of the management safety procedures, should be presented and justified. In particular, following aspects of the management of safety should be described:

- Implementation of the management system, including safety culture, grading in the system and its documentation.
- Responsibilities of senior management for the development and implementation of an effective management system.
- Resource management, including management of human resources, infrastructure and the working environment.
- Specification and development of the processes of the installation, including some generic processes of the management system.
- Measurement, assessment and improvement of the management system of a nuclear installation.

17.3 Consideration of safety culture

This section should present the operating organization's strategy to encourage the development, maintenance and enhancement of a strong safety culture throughout the lifetime of the plant. The information provided should demonstrate that the necessary arrangements are adequate and are in place at the plant. These arrangements should be aimed at promoting good awareness of all aspects of safety and at regularly reviewing with staff the level of safety awareness achieved on the site. The operating organization should, where possible, determine indicators of safety culture and should develop a programme to monitor such indicators. The staff should be consulted on the determination of these indicators and should be kept informed of the outcome of the reviews. Action should be taken in response to any indications of declining safety levels.

17.4 Monitoring and review of safety performance

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

This section should include a description of the way in which the operating organization intends to identify any development of the organization that could lead to the degradation of safety performance and should justify the appropriateness of the measures planned to prevent such degradation.

17.5 Quality Management

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

Chapter.18 Human Factors Engineering

Chapter 18 of the SAR should describe how state-of-the-art human factor engineering (HFE) principles are incorporated into (1) the planning and management of HFE activities; (2) the plant design process; (3) the characteristics, features, and functions of the human-system interfaces (HSIs), procedures, and training; (4) the implementation of the design; and (5) monitoring of performance at the site. This chapter should illustrate how human characteristics and capabilities are successfully integrated into the NPP design in such a way that they result in a state-of-the-art design and support successful performance of the required job tasks by plant personnel. Although this chapter should cover the issues associated with the human factors comprehensively, it is obvious that such factors should be adequately considered in many other chapters of the SAR, e.g. in chapters relevant for siting, operation, safety analysis, radiation protection, etc.

Description of this chapter was largely taken from the RG 1.206 and it is recommended to consult further this reference document for additional information.

18.1 HFE programme management

In this section the applicant should describe the HFE programme plan, including the following topics:

- General HFE programme goals and scope,
- HFE team and organization,
- HFE process and procedures,
- HFE issues tracking,
- HFE technical programme.

18.2 Review of NPP operating experience

The objective of this section is to document that the applicant has identified and analyzed HFE-related problems and issues in previous designs. In this way, negative features associated with predecessor designs may be avoided in the current one, while retaining positive features. The SAR should describe the applicant’s operating experience review (OER) and how it was used to identify HFE-related safety issues.

In addition to specification of objectives and scope of the assessment, the section should also describe in sufficient details the methodology used for the development and assessment, and to summarize the results of the assessment (this comment applies equally to all subsequent sections 18.3 through 18.12.

18.3 Functional requirements analysis and function allocation

This section should describe the objectives of the functional requirements analysis and the scope of the analyses performed. The objective of this section is to document that the applicant allocated those functions to human and system resources in a manner that takes advantage of human strengths and avoids human limitations. The scope should include identification and analysis of those functions that must be performed to satisfy the plant’s safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The SAR should describe the objectives of the applicant's function allocation analysis and the scope of the analyses performed. The scope should include analysis of requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and/or passive, self-controlling features), and (3) combinations of personnel and system elements (e.g., shared control or automatic systems with manual backup).

18.4 Task analysis

This section should describe the objectives and scope of the applicant's task analysis including assumptions and bounding conditions. The objective of this section is to document that the applicant's task analyses identify the specific tasks that are needed for function accomplishment and their information, control, and task support requirements. The scope description should address how representative and important operations, maintenance, test, inspection, and surveillance tasks were selected, as well as the range of operating modes included in the analyses. The SAR description should also discuss the use of PSA/human reliability analysis (HRA) analyses for the identification of the risk-important HAs, including the monitoring and backup of automatic actions.

18.5 Staffing and qualifications

This section should describe the objectives of the applicant's staffing and qualifications analyses, and the scope of the analyses performed. The objective of this section, in coordination with Section 13.1, is to document that the applicant has analyzed the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements. The scope should include the number and qualifications of personnel for the full range of plant conditions and tasks, including operational tasks (normal, abnormal, and emergency), and plant maintenance and testing (including surveillance testing). The personnel that should be considered include licensed control room operators and other categories of personnel, including non-licensed operators, shift supervisor, shift technical advisor, instrument and control technician, electrical maintenance personnel, mechanical maintenance personnel, radiological protection technician, chemistry technician, and engineering support personnel. In addition, any other plant personnel who perform tasks that directly relate to plant safety should be addressed.

18.6 Human reliability analysis

This section should describe the objectives of the applicant's use of the HRA in the HFE programme. The objective and scope of this section are to document that the applicant has incorporated the HRA/PSA results into other activities of the HFE programme such that risk-important human actions have been thoroughly addressed in the design of the plant.

18.7 Human-system interface design

The objective of this section is to document the applicant's HSI design process and scrope, including the translation of function and task requirements into the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. The SAR should also describe the process by which HSI design requirements are developed and HSI designs are identified and refined.

18.8 Procedure development

The objective of this section is to document, in coordination with Section 13.5, that the applicant's development programme incorporates HFE principles and criteria, along with other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. The SAR should describe the objectives and scope of the applicant's procedure development programme. This section should address the following procedures:

- generic technical guidelines for EOPs,
- plant and system operations (including startup, power, and shutdown operations),
- test and maintenance,
- abnormal and emergency operations,
- accident management guidelines,
- alarm response.

18.9 Training programme development

The objective of this section is to document, in coordination with Section 13.2, a systematic approach for the development of personnel training. The SAR should describe the objectives and scope of the applicant's training programme. The overall scope of training should be defined, and should include the following:

- Categories of personnel to be trained, including the full range of positions of operational personnel. The objective of this section is to document the applicant's HSI design process and scope, including the translation of function and task requirements into the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. The SAR should also describe the process by which HSI design requirements are developed and HSI designs are identified and defined,including licensed and non-licensed personnel whose actions may affect plant safety.
- The full range of plant conditions (normal, upset, and emergency).
- Specific operational activities (e.g., operations, maintenance, testing and surveillance).

- The full range of plant functions and systems, including those that may be different from those in predecessor plants (e.g., passive systems and functions).
- The full range of relevant HSIs (e.g., MCR, remote shutdown panel, local control stations, technical support centre) including characteristics that may be different from those in predecessor plants (e.g., display space navigation, operation of “soft” controls).

18.10 Verification and validation of HFE results

The objective of this section is to document that the applicant’s V&V activities to confirm that the HSI design conforms to HFE design principles and that it enables plant personnel to successfully perform their tasks to achieve plant safety and other operational goals. The scope should include the MCR, the remote shutdown panel, and local stations associated with the risk important human actions. The scope should identify which aspects of the plant HFE were included in the HSI task support verification, HFE design verification, and integrated system validation.

18.11 Design implementation

The objective of this section is to document (at the stage of the FSAR) how it will be verified that the as-built design conforms to the verified and validated design that resulted from the HFE design process. The objectives and scope of the design implementation should be described. The scope should include the following considerations:

- V&V of design aspects that cannot be completed as part of the HSI V&V program,
- confirmation that the as-built HSI, procedures, and training conform to the approved design
- confirmation that all HFE issues in the tracking system are appropriately addressed.

The FSAR should describe how aspects of the design that were not addressed in V&V will be evaluated. These aspects may include design characteristics, such as new or modified displays for plant-specific design features, and features that cannot be evaluated in a simulator, such as control room lighting and noise. The FSAR should describe how the final (as-built) HSIs, procedures, and training will be compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process and V&V activities. Also, the FSAR should describe the process for correcting any identified discrepancies. Justification should be provided for not changing design features that cause discrepancies. In addition, the FSAR should describe the process for ensuring that all HFE-related issues documented in the issue tracking system will be verified as adequately addressed.

18.12 Human performance monitoring

The objective of this section is to document the applicant has prepared a human performance monitoring strategy for determining that no significant safety degradation occurs because of any changes that are made in the plant and to confirm that the conclusions that have been drawn from the Integrated System Validation remain valid over time. The SAR should describe the objectives and scope of the applicant's human performance monitoring programme. The programme description should address how the programme provides reasonable assurance that the following criteria are met:

- The design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centres.
- Changes made to the HSIs, procedures, and training do not have adverse effects on personnel performance (e.g., changes do not interfere with previously trained skills).
- Human actions can be accomplished within established time and performance criteria.
- The acceptable level of performance established during the integrated system validation is maintained.

Chapter 19. Emergency preparedness

This chapter should provide information on emergency preparedness, demonstrating in a reasonable manner that, in the event of an accident, all actions necessary for the protection of the public, workers and the plant could be taken, and that the decision making process for implementation of these actions would be timely, disciplined, co-ordinated and effective. The emergency preparedness arrangements should cover the full range of accidents (in particular beyond design basis accidents and severe accidents) that would have effects on the environment and the off-site areas where preparations for the implementation of protective measures are warranted. The description should include information on the objectives and strategies, organization and management, and should provide sufficient information to show how the practical goals of the emergency plan will be met.

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

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

19.1 Emergency management

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

A general description should be provided here of the emergency arrangements for the protection of workers and the public in the event of an accident, including measures for: establishing emergency management; identifying, classifying and declaring emergency conditions; notifying off-site officials; activating the response; performing mitigatory actions; taking urgent protective actions on and off the site; protecting emergency workers; assessing the initial phase; managing the medical response; and keeping the public informed.

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

19.2 Emergency response facilities

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

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

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

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

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

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

19.3 Capability for the assessment of accident progression, radioactive releases and the consequences of accidents

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

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

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

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

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

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

19.4 Emergency plan considerations for multi-unit sites

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

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

(2) Include a review of the proposed extension of the existing site's emergency plan to ensure that the addition of a new reactor(s) would not decrease the effectiveness of the existing plans and the plans.

(3) Describe any required updates to existing emergency facilities and equipment, including the alert notification system.

(4) Incorporate any required changes to the existing onsite and offsite emergency response arrangements and capabilities with state and local authorities or private organizations.

(5) If applicable, address the exercise requirements for collocated licensees.

(6) Describe how emergency plans, to include security, is integrated and coordinated with emergency plans of adjacent sites.

Chapter 20. Environmental aspects

This chapter should provide a brief description of the approach taken to assess the impact on the environment of the construction of the plant, its operation under normal operational conditions as well as under accident conditions including severe accidents, and its decommissioning. The chapter should cover all aspects of site activities that have the potential to affect the radiological impacts of the site throughout the lifetime of the plant, including construction, operation and decommissioning. It is assumed that an overall environmental impact is covered by a dedicated environment assessment report. This chapter of the SAR is supposed to make a link between the environmental assessment report and the SAR itself. Depending on the stage of the project either relevant data from the environmental assessment report are referred to, or appropriate update of the information originally covered by the environmental report is provided. Consistency of chapter 20 with other chapters of the SAR should be ensured, with appropriate references to other chapters made. Only radiologically relevant information is included in chapter 20 of the SAR. Non-radiological aspects of the impact of the NPP on the environment are not covered.

20.1 Introduction to the environmental impact assessment

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

20.2 Environmental site description

This section should briefly summarizes all site characteristics which are important from environmental impact point of view. Land, water, ecology, socioeconomics, geology, and meteorology characteristics should be addressed. References to the Chapter 2 of the SAR should be made as appropriate.

20.3 Plant description

All plant characteristics which can be a source of the radiological environmental impact should be summarized here, with references made to other chapters of the SAR as appropriate.

20.4 Environmental impacts of construction

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

20.5 Environmental impacts of plant operation

This section should specify any authorized limits and operational targets for solid, liquid and gaseous discharges and measures to comply with such limits. All radiation impacts on surroundings under nuclear facility operation should be considered including:

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

Further on, the section should provide a description of the measures that will be taken to control discharges to the environment of solid, liquid and gaseous radioactive effluents during normal and abnormal operation. These discharges should be in accordance with the ALARA principle. External exposure from the plume of radioactive gases and aerosols released from the ventilation stacks, external exposure from radioactive fall-out (deposition) and internal exposure from inhalation of radionuclides should be addressed. The subsequent impact on contaminated land used for agricultural purposes and radioactive nuclides consisted in liquid discharges should also be taken into account. These radionuclides in the hydrosphere and biosphere get into the food chain and their subsequent intake in agricultural products and drinking water results in further exposure of the concerned population.

20.6 Environmental impacts of postulated accidents involving radioactive materials

The environmental risks of accidents involving radioactive material that can be postulated for the plant under review shall be addressed in this section. The list of accidents covered should be provided. The scope of the section should cover of the offsite dose consequences and resulting health effects for design basis accidents as well as for selected severe accidents for sufficient distance from the plant. The type of data and information needed will be affected by site- and station-specific factors, and the degree of detail should be modified according to the anticipated magnitude of the potential impacts. Overview of the measures and controls to limit adverse impacts during accidents should be described.

20.7 Environmental impacts of plant decommissioning

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

20.8 Environmental measurements and monitoring programmes

This section should describe the off-site monitoring regime for contamination levels and radiation levels. The dedicated environmental monitoring programmes and alarm systems should be described that are required to respond to unplanned radioactive releases and the automatic devices to interrupt such releases, if applicable. All routes, which could be the source of uncontrolled ionization radiation and radioactive substance leakage beyond the power plant systems should be addressed. Warning signals or automatic blockades preventing the unauthorized regime together with the activation levels settings should be specified.

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

This section should identify methods to make, store and retain records of radioactive releases that will routinely be made from the site. Further on this section should identify the measures that will be taken to make appropriate data available to the authorities and the public. All information relating to nuclear safety, radiation protection, physical protection and emergency preparedness should be provided in the form and deadlines determined by specific execution regulations and terms given by the state regulatory body in the operation license to operate a NPP.

Chapter 21. Decommissioning and end of life aspects

Decommissioning of the plant will become necessary either at the end of the lifetime of the plant or earlier if the operator so decides. The capability for decommissioning the plant should be demonstrated before initial criticality or before plant operation commences. This chapter of the SAR should contain the proposals anticipated at this point for the eventual decommissioning of the plant up to the regulatory delicensing of the site. First of all the principles associated with the decommissioning should be presented followed by the explanation how these principles are implemented in the design. It should be periodically updated to allow for an increasing level of detail and to reflect developments in the strategy for decommissioning.

21.1 General principles and regulations

In addition to general principles adopted for the decommissioning, this section should provide the information on the documentation required and regulations to be followed which ensures that both the radioactive dose received by the workers and the amount of radwaste and hazardous material produced are adequately reduced.

21.2 Differing approaches to decommissioning

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

21.3 Decommissioning concept

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

(a) Design solutions that minimize the amount of waste material produced and that facilitate decommissioning;

(b) Consideration of the type, volume and activity of radioactive waste produced during the operational and decommissioning phases;

(c) Identified options for decommissioning;

(d) Planning, phasing or staging of the decommissioning process, including appropriate surveillance requirements throughout the process;

(e) Adequate documentary control and maintenance of suitable and sufficient records;

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

21.4 Decommissioning plan

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

(a) The development of an engineering study for decommissioning, identifying the policy and objectives;

(b) The development of a rational strategy for decommissioning, including the identification of a staged approach to decommissioning, if appropriate;

(c) The development of a SAR for decommissioning;

(d) The development of a programme for bringing the reactor to a safe condition for total or partial dismantling;

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

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

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

(h) The observation of the licensing process throughout decommissioning.

21.5 Provisions for safety during decommissioning

This section should provide a short description of the measures necessary to ensure safety during decommissioning on the basis of the specified safety principles and safety objectives. The measures should be described that are adopted at the design and required in future operation with the objective to minimize the volume of radioactive structures, to reduce toxicity of the waste, lower the activity level of irradiated components, restrict the spread of contamination and pernmit easier decontamination, to facilitate the access of personnel and machines and removal of waste, and to ensure the collection of important data. An estimate of the volume of low and intermediate level waste should be provided. Special attention should be paid to the following aspects:

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

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

21.6 Decommissioned site end point

This section should specify a generic site end point to be reached following decommissioning and site clearance works.

Annex 1. Safety Report development in the course of the Project phases

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

	Safety Report Chapter	Project phases		
		Site Permit Initial SAR	Construction Permit Preliminary SAR	Commissioning Pre-operational SAR (Final SAR)
			systems	
9	**Auxiliary Systems and Civil Structures**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
10	**Steam and Power Conversion System**	General requirements on the design of SSC	Description of SSC and requirements on operation of systems	Verified/updated information
11	**Radioactive Waste Management**	General requirements on the design of SSC	Description of source terms, SSC and requirements on operation of systems	Verified/updated information
12	**Radiation Protection**	General requirements on radiation protection	Demonstration of compliance with the requirements	Verified/updated information
13	**Conduct of Operations**	General requirements on conduct of operations	Demonstration of compliance with the requirements	Verified/updated information
14	**Plant Commissioning**	General requirements on commissioning	Demonstration of compliance with the requirements	Demonstration of compliance with the requirements
15	**Safety Analysis (deterministic and probabilistic)**	General requirements on scope, methods and criteria for safety analysis	Demonstration of compliance with the requirements	Verified/updated demonstration of compliance with the requirements
16	**Operational Limits and Conditions (Technical Specifications)**	General requirements on operational limits and conditions	Description and specification of operational limits and conditions	Verified/updated description and specification of operational limits and conditions
17	**Management Systems**	General requirements on management	Description of management system	Updated description of management

	Safety Report Chapter	**Project phases**		
		Site Permit Initial SAR	Construction Permit Preliminary SAR	Commissioning Pre-operational SAR (Final SAR)
		system		system
18	**Human Factors Engineering**	General requirements on human factor engineering	Description of scope, methodology and results of human factor engineering	Updated description of human factor engineering
19	**Emergency Preparedness**	General requirements on emergency preparedness	Description of emergency facilities and emergency plans	Updated description of emergency facilities and emergency plans
20.	**Environmental Aspects**	Preliminary or expected information, consistent with EIA document	Updated information, referring to other parts of the SAR	Updated information, referring to other parts of the SAR
21	**Decommissioning and End of Life Aspects**	General requirements on decommissioning and end of life aspects	Preliminary information on decommissioning and end of life aspects	Updated information on decommissioning and end of life aspects

Annex 2. Unified description of the design of plant systems

The information to be presented in the SAR on various plant systems will inevitably depend on the particular type and design of reactor selected for construction. For some types of reactor many of the sections discussed below will be entirely relevant, while for other types they may not apply directly. However, as a general rule, all systems that have the potential to affect safety should be described in the SAR, and for such systems the following general approach should be considered.

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

A common structure is proposed for sections dealing mainly with systems or equipment. The structure and intended content is given below. When a topic is not relevant to a system or item of equipment, the intent is to keep the section and to note in the content guidance that “Typically no presentation on this topic is necessary.”

1. SYSTEM / EQUIPMENT FUNCTIONS

The safety and non-safety functions of the equipment/system are described here.

2. SAFETY DESIGN BASES

This section includes the safety design criteria, rules and regulations applying to the equipment/system, such as:

- postulated initiating events,
- safety requirements related to operating conditions, including stresses and environmental conditions (temperature, humidity, etc.)
- safety and seismic classification,
- protection against external hazards,
- protection against internal hazards,
- single failure criterion and protection against common cause failures,
- isolation,
- equipment qualification,
- design standards and fabrication codes, etc…

and other more specific design aspects such as:

- overpressure protection,
- thermal shock,
- leakage detection, etc…

3. DESCRIPTION

In this section, the equipment/system is described as well as its components.

The description includes the layout.

4. MATERIALS

In this section, adequate and sufficient information should be provided regarding the materials used in components, as well as the material interactions with fluids that could potentially impair operation of ESF systems. The intent of the information included in this section of the SAR is to ensure compatibility of the materials with the specific fluids to which the materials are subjected. Their specific properties, quality and chemistry requirements are described in this section. Ageing and degradation processes are also taken into account. The corresponding specific testing and surveillance programmes are presented as a complement to section 8 below concerning the whole equipment/system monitoring.

5. INTERFACES WITH OTHER EQUIPMENT OR SYSTEMS

The support systems (e.g., electric power, cooling water…), supported systems, and other connected systems are presented as well as the corresponding design requirements.

6. SYSTEM / EQUIPMENT OPERATION

This section summarizes the operation of the system or equipment.

7. INSTRUMENTATION AND CONTROL

This section describes the method of control, the alarms, indications and interlocks associated with operation of the equipment/system and its components.

8. MONITORING, INSPECTION, TESTING AND MAINTENANCE

This section presents the monitoring, inspection, testing and maintenance which will help demonstrate that:

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

9. RADIOLOGICAL ASPECTS

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

10. PERFORMANCE AND SAFETY EVALUATION

This section presents the measures taken to address each of the safety design aspects or requirements listed in the above section 2.

In addition, the design issues that may be raised in the other sections above are also addressed here.

Annex 3. Comparison of structure of Safety Analysis Report according IAEA Safety Guide GS-G-4.1 and US NRC Regulatory Guide 1.206

US NRC Regulatory Guide 1.206	**IAEA Safety Guide GS-G-4.1**
1. Introduction and general description of the plant	1. Introduction
2. Site Characteristics	2. General Plant Description
3. Design of Structures, Components, Equipment and Systems	3. Management of Safety
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	9B.3.1 Other structure 1
	9B.3.1.1 Structural role
	9B.3.1.2 Safety design bases
	9B.3.1.3 Structural description
	9B.3.1.4 Materials
	9B.3.1.5 Interfaces with other equipment or systems
	9B.3.1.6 System / Equipment operation
	9B.3.1.7 Instrumentation and control
	9B.3.1.8 Monitoring, testing, inspection, and maintenance
	9B.3.1.9 Radiological aspects
	9B.3.1.10 Performance and safety evaluation
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	9B.3.n Other structure n

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10 Steam and Power Conversion System	10 Steam and Power Conversion System
10.1 Summary Description	10.1 Role and General Description
10.3 Main Steam Supply System	10.2 Main Steam Supply System
	10.2.1 System / Equipment Functions
	10.2.2 Safety design bases
	10.2.3 Description
	10.2.4 Materials
	10.2.5 Interfaces with other equipment or systems
	10.2.6 System / Equipment Operation
	10.2.7 Instrumentation and control
	10.2.8 Monitoring, inspection, testing, and maintenance
	10.2.9 Radiological aspects
	10.2.10 Performance and safety evaluation
10.4.7 Condensate and Feedwater Systems	10.3 Feedwater systems
	10.3.1 Main feedwater system
	10.3.1.1 System / Equipment Functions
	10.3.1.2 Safety design bases
	10.3.1.3 Description
	10.3.1.4 Materials
	10.3.1.5 Interfaces with other equipment or systems
	10.3.1.6 System / Equipment Operation
	10.3.1.7 Instrumentation and control
	10.3.1.8 Monitoring, inspection, testing, and maintenance
	10.3.1.9 Radiological aspects
	10.3.1.10 Performance and safety evaluation
10.4.9 Auxiliary Feedwater System (PWR)	10.3.2 Auxiliary feedwater system (non-safety)
	10.3.2.1 System / Equipment Functions
	10.3.2.2 Safety design bases
	10.3.2.3 Description
	10.3.2.4 Materials
	10.3.2.5 Interfaces with other equipment or systems
	10.3.2.6 System / Equipment Operation
	10.3.2.7 Instrumentation and control
	10.3.2.8 Monitoring, inspection, testing, and maintenance
	10.3.2.9 Radiological aspects
	10.3.2.10 Performance and safety evaluation
10.2 Turbine Generator	10.4 Turbine Generator
10.2.1 Design Bases	10.4.1 Design Bases
10.2.2 Description	10.4.2 Description
10.2.3 Turbine Rotor Integrity	10.4.3 Turbine Rotor Integrity
	10.5 Turbine and Condenser systems

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10.4.1 Main Condensers	10.5.1 Main Condenser 10.5.1.1 System / Equipment Functions 10.5.1.2 Safety design bases 10.5.1.3 Description 10.5.1.4 Materials 10.5.1.5 Interfaces with other equipment or systems 10.5.1.6 System / Equipment Operation 10.5.1.7 Instrumentation and control 10.5.1.8 Monitoring, inspection, testing, and maintenance 10.5.1.9 Radiological aspects 10.5.1.10 Performance and safety evaluation
10.4.2 Main Condenser Evacuation System	10.5.2 Condenser air extraction system 10.5.2.1 System / Equipment Functions 10.5.2.2 Safety design bases 10.5.2.3 Description 10.5.2.4 Materials 10.5.2.5 Interfaces with other equipment or systems 10.5.2.6 System / Equipment Operation 10.5.2.7 Instrumentation and control 10.5.2.8 Monitoring, inspection, testing, and maintenance 10.5.2.9 Radiological aspects 10.5.2.10 Performance and safety evaluation
10.4.5 Circulating Water System	10.5.3 Circulating Water System 10.5.3.1 System / Equipment Functions 10.5.3.2 Safety design bases 10.5.3.3 Description 10.5.3.4 Materials 10.5.3.5 Interfaces with other equipment or systems 10.5.3.6 System / Equipment Operation 10.5.3.7 Instrumentation and control 10.5.3.8 Monitoring, inspection, testing, and maintenance 10.5.3.9 Radiological aspects 10.5.3.10 Performance and safety evaluation

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10.4.7 Condensate and Feedwater Systems	10.5.4 Condensate system 10.5.4.1 System / Equipment Functions 10.5.4.2 Safety design bases 10.5.4.3 Description 10.5.4.4 Materials 10.5.4.5 Interfaces with other equipment or systems 10.5.4.6 System / Equipment Operation 10.5.4.7 Instrumentation and control 10.5.4.8 Monitoring, inspection, testing, and maintenance 10.5.4.9 Radiological aspects 10.5.4.10 Performance and safety evaluation
10.4.6 Condensate Cleanup System	10.5.5 Condensate Cleanup System 10.5.5.1 System / Equipment Functions 10.5.5.2 Safety design bases 10.5.5.3 Description 10.5.5.4 Materials 10.5.5.5 Interfaces with other equipment or systems 10.5.5.6 System / Equipment Operation 10.5.5.7 Instrumentation and control 10.5.5.8 Monitoring, inspection, testing, and maintenance 10.5.5.9 Radiological aspects 10.5.5.10 Performance and safety evaluation

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	10.5.6 Turbine auxiliary systems
10.4.3 Turbine Gland Sealing System	10.5.6.1 Turbine Gland Sealing System\ 10.5.6.1.1 System / Equipment Functions 10.5.6.1.2 Safety design bases 10.5.6.1.3 Description 10.5.6.1.4 Materials 10.5.6.1.5 Interfaces with other equipment or systems 10.5.6.1.6 System / Equipment Operation 10.5.6.1.7 Instrumentation and control 10.5.6.1.8 Monitoring, inspection, testing, and maintenance 10.5.6.1.9 Radiological aspects 10.5.6.1.10 Performance and safety evaluation
10.4.4 Turbine Bypass System	10.5.6.2 Turbine Bypass System 10.5.6.2.1 System / Equipment Functions 10.5.6.2.2 Safety design bases 10.5.6.2.3 Description 10.5.6.2.4 Materials 10.5.6.2.5 Interfaces with other equipment or systems 10.5.6.2.6 System / Equipment Operation 10.5.6.2.7 Instrumentation and control 10.5.6.2.8 Monitoring, inspection, testing, and maintenance 10.5.6.2.9 Radiological aspects 10.5.6.2.10 Performance and safety evaluation
	10.5.7 Generator auxiliary systems 10.5.7.1 System / Equipment Functions 10.5.7.2 Safety design bases 10.5.7.3 Description 10.5.7.4 Materials 10.5.7.5 Interfaces with other equipment or systems 10.5.7.6 System / Equipment Operation 10.5.7.7 Instrumentation and control 10.5.7.8 Monitoring, inspection, testing, and maintenance 10.5.7.9 Radiological aspects 10.5.7.10 Performance and safety evaluation

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10.4.8 Steam Generator Blowdown System (PWR)	10.6 Steam generator blowdown system 10.6.1 System / Equipment Functions 10.6.2 Safety design bases 10.6.3 Description 10.6.4 Materials 10.6.5 Interfaces with other equipment or systems 10.6.6 System / Equipment Operation 10.6.7 Instrumentation and control 10.6.8 Monitoring, inspection, testing, and maintenance 10.6.9 Radiological aspects 10.6.10 Performance and safety evaluation 10.7 Break preclusion implementation for main steam and feedwater lines

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11 Radioactive Waste Management	11 Radioactive Waste Management
11.1 Source Terms	11.1 Source Terms
11.2 Liquid Waste Management Systems	11.2 Liquid Waste Management Systems 11.2.1 System / Equipment Functions 11.2.2 Safety design bases 11.2.3 Description 11.2.4 Materials 11.2.5 Interfaces with other equipment or systems 11.2.6 System / Equipment Operation 11.2.7 Instrumentation and control 11.2.8 Monitoring, inspection, testing, and maintenance 11.2.9 Radiological aspects 11.2.10 Performance and safety evaluation
11.3 Gaseous Waste Management Systems	11.3 Gaseous Waste Management Systems 11.3.1 System / Equipment Functions 11.3.2 Safety design bases 11.3.3 Description 11.3.4 Materials 11.3.5 Interfaces with other equipment or systems 11.3.6 System / Equipment Operation 11.3.7 Instrumentation and control 11.3.8 Monitoring, inspection, testing, and maintenance 11.3.9 Radiological aspects 11.3.10 Performance and safety evaluation
11.4 Solid Waste Management System	11.4 Solid Waste Management System 11.4.1 System / Equipment Functions 11.4.2 Safety design bases 11.4.3 Description 11.4.4 Materials 11.4.5 Interfaces with other equipment or systems 11.4.6 System / Equipment Operation 11.4.7 Instrumentation and control 11.4.8 Monitoring, inspection, testing, and maintenance 11.4.9 Radiological aspects 11.4.10 Performance and safety evaluation

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
11.5 Process and Effluent Radiological Monitoring and Sampling Systems	11.5 Process and Effluent Radiological Monitoring and Sampling Systems 11.5.1 System / Equipment Functions 11.5.2 Safety design bases 11.5.3 Description 11.5.4 Materials 11.5.5 Interfaces with other equipment or systems 11.5.6 System / Equipment Operation 11.5.7 Instrumentation and control 11.5.8 Monitoring, inspection, testing, and maintenance 11.5.9 Radiological aspects 11.5.10 Performance and safety evaluation

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
12 Radiation Protection	12 Radiation Protection
12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	12.1 ALARA Considerations
12.2 Radiation Sources	12.2 Radiation Sources
12.3 Radiation Protection Design Features	12.3 Radiation Protection Design Features
12.3.1 Facility Design Features	12.3.1 Facility Design Features
12.3.2 Shielding	12.3.2 Shielding
12.3.3 Ventilation	12.3.3 Ventilation
12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation	12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation
12.4 Dose Assessment	12.4 Dose Assessment
12.5 Operational Radiation Protection Program	12.5 Operational Radiation Protection Programme

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
13 Conduct of Operations	13 Conduct of Operations
13.1 Organizational Structure of Applicant	13.1 Organizational Structure of Operating Organization
13.1 Organizational Structure of Applicant	13.1 Organizational Structure of Applicant
13.1.2 Operating Organization	13.1.2 Operating Organization
13.1.3 Qualifications of Nuclear Plant Personnel	13.1.3 Qualifications of Nuclear Plant Personnel
13.2 Training	13.2 Training
13.2.1 Plant Staff Training Program	
13.2.2 Applicable NRC Documents	
Section 13.3 transfer into a separate chapter	
13.3 Emergency Planning	Covered by chapter 19.
13.3.1 Combined License Application and Emergency Plan Content	
13.3.2 Emergency Plan Considerations for Multi-unit Sites	
13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria	
13.4 Operational Program Implementation	13.3 Operational Programme Implementation
	13.3.1 Maintenance, surveillance, inspection and testing
	13.3.2 Core management and fuel handling
	13.3.3 Management of ageing
	13.3.4 Control of modifications
	13.3.5 Programme for the feedback of operational experience
	13.3.6 Documents and records
	13.3.7 Outages
	13.3.x etc.
13.5 Plant Procedures	13.4 Plant Procedures
13.5.1 Administrative Procedures	13.4.1 Administrative Procedures
13.5.2 Operating and Maintenance Procedures	13.4.2 Operating Procedures
	13.4.3 Emergency Operating Procedures
	13.4.4 Accident Management Guidelines
13.6 Security	13.5 Security
13.6.1 Physical Security - Combined License	
13.7 Fitness for Duty	

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
14 Initial test program	14 Plant Commissioning
14.1 Specific Information To Be Included in the PSAR	14.1 Specific Information To Be Included in SAR prior to construction
14.2 Specific Information To Be Included in the FSAR	14.2 Specific Information To Be Included in SAR prior to commissioning

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
15 Transient and Accident Analyses 15.1 Transient and Accident Classification 15.2 Frequency of Occurrence 15.3 Plant Characteristics Considered in the Safety Evaluation 15.4 Assumed Protection System Actions	15 Safety Analysis (deterministic and probabilistic) 15.1 General considerations and applicable reference documents 15.1.1 Introduction 15.1.2 Scope of safety analysis and approach adopted 15.1.3 Analysis of design basis conditions 15.1.4 Analysis of design extension conditions 15.1.4.1 Analysis of complex sequences
15.5 Evaluation of Individual Initiating Events 15.6 Event Evaluation	15.1.4.2 Analysis of severe accidents 15.1.5 Analysis of the hazards 15.1.6 Applicable reference documents 15.1.7 Structure of the chapter 15.2 Safety objectives and acceptance criteria 15.2.1 Safety objectives and safety analysis 15.2.2 Deterministic acceptance criteria 15.2.2.1 Acceptance criteria for analysis of core cooling and system pressure 15.2.2.2 Acceptance criteria for analysis of radiological effects of design basis conditions and design extension conditions 15.2.2.3 Acceptance criteria for analysis of pressure-temperature transients in the containment 15.2.2.4 Acceptance criteria for thermal shocks 15.2.2.5 Acceptance criteria for analysis of primary to secondary system leakages 15.2.2.6 Acceptance criteria for hazards 15.3 Identification and classification of PIEs 15.3.1 Basis for classification of PIEs 15.3.2 Categorization of events according their frequencies 15.3.3 Grouping of events according their safety aspects 15.3.4 List of events/scenarios analysed in the current SAR 15.4 Human actions 15.4.1 General considerations 15.4.2 Human actions in deterministic safety analysis 15.4.3 Human actions in probabilistic safety analysis 15.5 Deterministic analyses 15.5.1 General description of the approach 15.5.1.1 Conservatism in safety analysis 15.5.1.2 Description of the computer codes used

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
	15.5.1.3 Description of the mathematical models 15.5.1.4 Database for the deterministic safety analysis 15.5.2 Safety in normal operation 15.5.2.1 Introduction 15.5.2.2 Acceptance criteria for normal operation 15.5.2.3 Description of normal operating regimes 15.5.2.4 Method and scope of analysis 15.5.2.5 Results of analysis 15.5.2.6 Conclusions 15.5.3 Analysis of core cooling and system pressure for reactivity induced accidents 15.5.4 Analysis of core cooling and system pressure for a decrease of reactor coolant flow 15.5.5 Analysis of system pressure for increase of reactor coolant inventory 15.5.6 Analysis of core cooling and system pressure for increase of heat removal by the secondary circuit 15.5.7 Analysis of core cooling and system pressure for decrease of heat removal by the secondary circuit 15.5.8 Analysis of loss of electrical power supply 15.5.9 Analysis of core cooling for LOCAs 15.5.10 Analysis of primary to secondary circuit leakage (with and without iodine spiking) 15.5.11 Analysis of pressurized thermal shocks 15.5.12 Analysis of pressure-temperature transients in the containment 15.5.13 Analysis of radioactivity transport during bounding design basis accidents 15.5.14 Analysis of fuel handling events 15.5.15 Analysis of internal hazards 15.5.16 Analysis of natural external hazards 15.5.17 Analysis of man-made external hazards 15.5.18 Radioactive release from subsystem or component 15.5.19 Anticipated transients without scram (complex sequences) 15.5.20 Beyond design basis accidents other than ATWS (complex sequences) 15.5.21 Severe accidents

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
	15.6 Probabilistic analyses 15.6.1 General approach to probabilistic safety analysis 15.6.1.1 Probabilistic safety goals 15.6.1.2 PSA scope and level of detail 15.6.2 Results of PSA Level 1 15.6.2.1 PSA Level 1 results and conclusions 15.6.2.1.1 Core Damage Frequency 15.6.2.1.2 Contribution from accident sequences at power operation 15.6.2.1.3 Assessment of shutdown states 15.6.2.1.3.1 Results by plant operating states 15.6.2.1.3.2 Results by initiating events 15.6.3 Results of PSA Level 2 15.6.3.1 Interface Level 1 - Level 2 PSA 15.6.3.2 Containment performance analysis and source term 15.6.3.3 Main PSA Level-2 results 15.6.5 PSA insights and applications
	15.7 Summary of results of the safety analyses 15.7.1 Introduction 15.7.2 Scope of analysis 15.7.3 Results of deterministic safety analysis 15.7.3.1 Deterministic analysis of internal DBC events 15.7.3.2 Deterministic analysis of complex sequences 15.7.3.3 Deterministic analysis of severe accidents 15.7.3.4 Analysis of internal hazards 15.7.3.5 Analysis of natural external hazards 15.7.3.6 Analysis of man made external hazards 15.7.4 Results of probabilistic analysis 15.7.5 Conclusions

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
16 Technical specifications 16.1 Preliminary Technical Specifications 16.2 Proposed Final Technical Specifications	16 Operational Limits and Conditions (Technical Specifications) 16.1 Use and Application 16.2 Safety Limits 16.3 Limiting Conditions for Operation, Protection Thresholds, Actions, and Surveillance Requirements 16.4 Administrative Requirements 16.5 Bases

Number and title of the chapter - RG 1.206 - NRC Combined Operating License applications (Safety Analysis Report)	Number and title of the chapter - Proposed
	17 Management Systems 17.1 General considerations 17.2 Specific aspects of management of safety processes 17.3 Consideration of safety culture 17.4 Monitoring and review of safety performance
17 Quality Assurance and Reliability Assurance	17.5 Quality Management
17.5 Quality Assurance Program Guidance	17.5.1 Quality Management Programme requirements 17.5.2 Quality Management Programme Implementation
17.1 Quality Assurance During the Design and Construction Phase	17.5.2.1 Quality Management Programme during design 17.5.2.2 Quality Management Programme during construction
17.2 Quality Assurance During the Operations Phase	17.5.2.3 Quality Management Programme during operations
17.3 Reliability Assurance Program Description	See Chapter 13
17.6 Description of the Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule	See Chapter 13

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18 Human Factors Engineering	18 Human Factors Engineering
18.1 HFE Program Management	18.1 HFE Programme Management
18.1.1 General HFE Program Scope	18.1.1 General HFE Programme Scope
18.1.2 HFE Team And Organization	18.1.2 HFE Team And Organization
18.1.3 HFE Process And Procedures	18.1.3 HFE Process And Procedures
18.1.4 HFE Issues Tracking	18.1.4 HFE Issues Tracking
18.1.5 HFE Technical Program	18.1.5 HFE Technical Programme
18.2 Operating Experience Review	18.2 Review of NPP Operating Experience
18.2.1 Objectives And Scope	18.2.1 Objectives And Scope
18.2.2 Methodology	18.2.2 Methodology
18.2.3 Results	18.2.3 Results
18.3 Functional Requirements Analysis And Function Allocation	18.3 Functional Requirements Analysis And Function Allocation
18.3.1 Objectives And Scope	18.3.1 Objectives And Scope
18.3.2 Methodology	18.3.2 Methodology
18.3.3 Results	18.3.3 Results
18.4 Task Analysis	18.4 Task Analysis
18.4.1 Objectives And Scope	18.4.1 Objectives And Scope
18.4.2 Methodology	18.4.2 Methodology
18.4.3 Results	18.4.3 Results
18.5 Staffing And Qualifications	18.5 Staffing And Qualifications
18.5.1 Objectives And Scope	18.5.1 Objectives And Scope
18.5.2 Methodology	18.5.2 Methodology
18.5.3 Results	18.5.3 Results
18.6 Human Reliability Analysis	18.6 Human Reliability Analysis
18.6.1 Objectives And Scope	18.6.1 Objectives And Scope
18.6.2 Methodology	18.6.2 Methodology
18.6.3 Results	18.6.3 Results
18.7 Human-System Interface Design	18.7 Human-System Interface Design
18.7.1 Objectives And Scope	18.7.1 Objectives And Scope
18.7.2 Methodology	18.7.2 Methodology
18.7.3 Results	18.7.3 Results
18.8 Procedure Development	18.8 Procedure Development
18.8.1 Objectives And Scope	18.8.1 Objectives And Scope
18.8.2 Methodology	18.8.2 Methodology
18.8.3 Results	18.8.3 Results
18.9 Training Program Development	18.9 Training Programme Development
18.9.1 Objectives And Scope	18.9.1 Objectives And Scope
18.9.2 Methodology	18.9.2 Methodology
18.9.3 Results	18.9.3 Results
18.10 Verification And Validation	18.10 Verification And Validation of HFE results
18.10.1 Objectives And Scope	18.10.1 Objectives And Scope
18.10.2 Methodology	18.10.2 Methodology
18.10.3 Results	18.10.3 Results

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18.11 Design Implementation	18.11 Design Implementation
18.11.1 Objectives And Scope	18.11.1 Objectives And Scope
18.11.2 Methodology	18.11.2 Methodology
18.11.3 Results	18.11.3 Results
18.12 Human Performance Monitoring	18.12 Human Performance Monitoring
18.12.1 Objectives and scope	18.12.1 Objectives and scope
18.12.2 Methodology	18.12.2 Methodology
18.12.3 Results	18.12.3 Results

Originally included as a section 13.3 in of RG 1.206 chapter on operation

13.3 Emergency Planning
13.3.1 Combined License Application and Emergency Plan Content
13.3.2 Emergency Plan Considerations for Multi-unit Sites
13.3.3 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria
In RG 1.70 only two sections
13.3.1 Preliminary Planning (PSAR)
13.3.2 Emergency Plan (FSAR)

19 Emergency Preparedness
19.1 Emergency management

19.2 Emergency response facilities

19.3 Capability for the assessment of accident progression, radioactive releases and the consequences of accidents

19.4 Emergency Plan Considerations for Multi-unit Sites

Environmental aspects were not covered neither by RG 1.70 nor by RG 1.206. The list below is taken from NUREG 1555.

19 Probabilistic Risk Assessment and Severe Accident Evaluation	See subchapter 15 6
19.1 Regulatory Basis	
19.2 Uses of PRA and Severe Accident Evaluations	
19.3 Scope	
19.4 Level of Detail	
19.5 Technical Adequacy	
19.6 Development of Risk Insights	
19.7 PRA Maintenance and Update	
19.8 Severe Accidents	
19.9 Documentation	

16. Decommissioning
16.1 Introduction
16.2 General
16.3 Differing Approaches to Decommissioning
16.4 Decommissioning Concept
16.5 Outline Plan – Main Activities
16.6 Provisions for Safety during Decommissioning
16 6.1 Inherently Simple Design
16.6.2 Design Features for Radiation Protection
16.6.3 Design Features for Protection against the Limitation of Contamination
16.6.4 Design Features Supporting Decommissioning
16.7 Decommissioned Site End Point
(this structure is taken from AP 1000 Pre-construction SAR for UK, not covered by original RG 1.206)

21 Decommissioning and End of Life Aspects
21.1 General principles and regulations
21.2 Differing Approaches to Decommissioning
21.3 Decommissioning Concept
21.4 Decommissioning Plan
21.5 Provisions for Safety during Decommissioning
21.6 Decommissioned Site End Point