No: AERB/NPP/SG/G-9

GOVERNMENT OF INDIA

AERB SAFETY GUIDE

STANDARD FORMAT AND CONTENTS OF SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE: AERB/NPP/SG/G-9

STANDARD FORMAT AND CONTENTS OF SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

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FOREWORD

Activities concerning establishment and utilisation of nuclear facilities and use of radioactive sources are to be carried out in India in accordance with the provisions of the Atomic Energy Act, 1962. In pursuance of ensuring safety of members of the public and occupational workers as well as protection of the environment, the Atomic Energy Regulatory Board (AERB) has been entrusted with the responsibility of laying down safety standards and enforcing rules and regulations for such activities. The Board has, therefore, undertaken a programme of developing safety standards, safety codes, and related guides and manuals. While some of these documents cover aspects such as siting, design, construction, operation, quality assurance and decommissioning of nuclear and radiation facilities, other documents cover regulatory aspects of these facilities.

Safety codes and safety standards are formulated on the basis of nationally and internationally accepted safety criteria for design, construction and operation of specific equipment, structures, systems and components of nuclear and radiation facilities. Safety codes establish the objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety guides and guidelines elaborate various requirements and furnish approaches for their implementation. Safety manuals deal with specific topics and contain detailed scientific and technical information on the subject. These documents are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of the Board before they are published. The documents are revised when necessary, in the light of experience and feedback from users as well as new developments in the field.

In order to standardize the format and required contents of Safety Analysis Report (SAR) for various types of nuclear power projects/plants and thereby helping in quick and efficient safety review of the project/plant, this safety guide is prepared as a technologically neutral document giving guidelines for preparing and submission of SAR. In drafting this document, the information available in a number of national and international publications has been made use of, including those of the International Atomic Energy Agency (IAEA), USNRC and Atomic Energy Regulatory Board (AERB).

This guide has been prepared by a working group comprising AERB, NPCIL, BARC, IGCAR and BHAVINI officers. It has been reviewed by experts and the Advisory Committee on Code and Guides on Governmental Organisation for Nuclear and Radiation Facilities (ACCGORN).

AERB wishes to thank all individuals and organisations who have prepared and reviewed the document and helped in its finalisation. The list of persons, who have participated in this task, along with their affiliations, is included for information.

SBhardwaj

(S. A. Bhardwaj)
Chairman, AERB

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DEFINITIONS

Acceptable Limits

Limits acceptable to the regulatory body for accident condition or potential exposure.

Acceptance Criteria

The standard or acceptable value against which the value of a functional or condition indicator is used to assess the ability of a system, structure or component to perform its design function or compliance with stipulated requirements.

Accident

An unplanned event resulting in (or having the potential to result in) injury or damage to equipment which may or may not cause release of unacceptable quantities of radioactive material or toxic/hazardous chemicals.

Activation

The production of radionuclides by irradiation.

Active Component

A component whose functioning depends on an external input, such as actuation, mechanical movement, or supply of power, and which, therefore, influences the system process in an active manner, e.g. pumps, valves, fans, relays and transistors. It is emphasized that this definition is necessarily general in nature as is the corresponding definition of passive component. Certain components, such as rupture discs, check valves, injectors and some solid state electronic devices, have characteristics which require special consideration before designation as an active or passive component.

Ageing

General process in which characteristics of structures, systems or component gradually change with time or use (although the term 'ageing' is defined in a neutral sense - the changes involved in ageing may have no effect on protection or safety, or could even have a beneficial effect - it is commonly used with a connotation of changes that are (or could be) detrimental to protection or safety, i.e. as a synonym of 'ageing degradation').

ALARA

An acronym for 'As Low As Reasonably Achievable'. A concept meaning that the design and use of sources, and the practices associated therewith, should be such as to ensure that exposures are kept as low as reasonably practicable, with economic and social factors taken into account.

Analysis

A process of mathematical or other logical reasoning or deduction that leads from stated premises to the conclusion/response/outcome/adequacy of a system or any other item of interest.

Anticipated Operational Occurrences (AOO)

An operational process deviating from normal operation, which is expected to occur during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety, nor lead to accident conditions.

Assessment

Systematic evaluation of the arrangements, processes, activities and related results for their adequacy and effectiveness in comparison with set criteria.

Atomic Energy Regulatory Board (AERB)

A national authority designated by the Government of India having the legal authority for issuing regulatory consent for various activities related to the nuclear and radiation facility and to perform safety and regulatory functions, including their enforcement for the protection of site personnel, the public and the environment against undue radiation hazards.

Authorisation

A type of regulatory consent issued by the regulatory body for all sources, practices and uses involving radioactive materials and radiation-generating equipment (see also 'Consent').

Channel (Coolant)

The primary heat-transport coolant tube and accessories through which the reactor coolant flows in a reactor.

Channel (Instrumentation)

An arrangement of interconnected components within a system that initiates output(s).

Cladding

An external sheath of material over nuclear fuel or other material that provides protection from a chemically reactive environment and containment of radioactive products produced during the irradiation of the composite. It may provide a structural support.

Classification (Radioactive Waste)

Determination of the physical, chemical and radiological properties of the waste to establish the need for further adjustment, treatment, conditioning, or its suitability for further handling, processing, storage or disposal.

Clearance Levels

A set of values established by the regulatory body and expressed in terms of activity concentrations and/or total activity, at or below which sources of radiation may be released from regulatory control.

Collective Dose

An expression for the total radiation dose incurred by a population and defined as the product of the number of individuals exposed to a source and their average radiation dose.

Commencement of Operation of Nuclear Power Plant

The specific activity/activities in the commissioning phase of a nuclear power plant towards first approach to criticality, starting from fuel loading.

Commissioning

The process during which structures, systems and components of a nuclear or radiation facility, on being constructed, are made functional and verified in accordance with design specifications and found to have met the performance criteria.

Component

The smallest part of a system necessary and sufficient to consider for system analysis.

Conditioning of Waste

The processes that transform waste into a form suitable for transport and/or storage and/or disposal. These may include converting the waste to another form, enclosing the waste in containers and providing additional packaging.

Confinement

Barrier, which surrounds the main parts of a nuclear facility, carrying radioactive materials and designed to prevent or to mitigate uncontrolled release of radioactivity into the environment during commissioning, operational states, design basis accidents or in decommissioning phase. (see ‘'Containment'’ also)

Consent

A written permission issued to the “Consentee” by the regulatory body to perform specified activities related to nuclear and radiation facilities. The types of consents are 'Licence', 'Authorisation', 'Registration' and 'Approval', and will apply according to the category of the facility, the particular activity and radiation source involved.

Construction

The process of manufacturing, testing and assembling the components of a nuclear or radiation facility, the erection of civil works and structures, the installation of components and equipment and the performance of associated tests.

Containment

(See ‘’Primary Containment’’/’’Secondary Containment’’/’’Confinement’’).

Containment Isolation

The process of isolating containment or boxing up the containment so that there is no direct path from the system available for the radioactivity to reach the environment.

Containment Penetrations

Openings in the containment envelope for passage of personnel, materials, process piping and cables.

Contamination

The presence of radioactive substances in or on a material/the human body or other places in excess of quantities specified by the competent authority.

Controlled Area

A delineated area to which access is controlled and in which specific protection measures and safety provisions are, or could be, required for

- controlling normal exposures or preventing the spread of contamination during normal working conditions; and
- preventing potential exposures or limiting their extent should they occur.

Control System

A system performing actions needed for maintaining plant variables within prescribed limits.

Core Components

All items other than fuel, which reside in the core of a nuclear power plant and have a bearing on fuel integrity and/or utilisation (e.g. calandria, coolant channels, in-core detectors and reactivity devices).

Core Damage

Reactor state brought about by the accident conditions with loss of core geometry or resulting in crossing of design basis limits or acceptance criteria limits for one or more parameters. (The parameters to be considered include fuel clad strain, fuel clad temperature, pressure for primary and secondary systems, fuel enthalpy, clad oxidation, % of fuel failure, H_2 generation from metal-water reaction, radiation dose, time required for operator to take emergency mitigatory action).

Core Management

All activities associated with the use of fuel and core components in a nuclear power plant with the ultimate aim of ensuring integrity and efficient use of the same.

Countermeasures

An action aimed at alleviating or mitigating the consequences of accidental release of radioactive material into the environment.

Criteria

Principles or standards on which a decision or judgement can be based. They may be quantitative or qualitative.

Critical Component

Component, whose failure, in a given operating state of the system, results in the system failure.

Criticality

The 'stage' or 'state' of a fissile material system where a self-sustained nuclear chain reaction is just maintained.

Cyclone

A low-pressure belt generated in the upper atmosphere, which has circular isobaric pattern and associated wind speed greater than 60 km/h.

Decontamination

The removal or reduction of contamination by physical or chemical means.

Defence-in-Depth

Provision of multiple levels of protection for ensuring safety of workers, the public or the environment.

Derived Limits

Values of quantities related to the primary or secondary limits by a defined model such that if the derived limits are not exceeded, it is most unlikely that the primary limits will be exceeded.

Design

The process and results of developing the concept, detailed plans, supporting calculations and specifications for a nuclear or radiation facility.

Design Basis Accidents (DBAs)

A set of postulated accidents which are analysed to arrive at conservative limits on pressure, temperature and other parameters which are then used to set specifications to be met by plant structures, systems and components, and fission product barriers.

Design Basis Events (DBEs)

The set of events, that serve as part of the basis for the establishment of design requirements for systems, structures and components within a facility. Design basis events (DBEs) include operational transients and certain accident conditions under postulated initiating events (PIEs) considered in the design of the facility (see also ‘’Design Basis Accidents’’).

Design Basis Fire

A hypothetical fire, which is assumed for the purpose of fire protection design or analysis. Fire is assumed to be one that would lead to the most severe damage in the area under consideration in the absence of fire protection systems.

Design Basis External Events (DBEEs)

The parameter values associated with, and characterising, an external event (e.g. missile impact, chemical explosion in the vicinity, etc.) or combinations of external events selected for design of all or any part of a nuclear facility.

Design Basis Flood (DBF)

The flood selected for deriving a design basis for a nuclear facility.

Design Life

The period of time for which the item will perform satisfactorily meeting the criteria set forth in the design specification.

Design Limits

Limits on the design parameters within which the design of the structures, systems and components of a nuclear facility have been shown to be safe.

Deterministic Effects

A radiation effect for which generally a threshold level of dose exists, above which the severity of the effect is greater for a higher dose.

Discharge (Radioactive)

Planned and controlled release of (gaseous or liquid) radioactive material into the environment.

Discharge Limits

The limits prescribed by the regulatory body for effluent discharges into atmosphere/aquatic environment from nuclear/radiation facilities.

Dispersion Coefficient (Atmospheric)

The standard deviation in a specified direction of the Gaussian distribution model used for dispersion of material in atmosphere.

Disposal (Radioactive Waste)

The emplacement of waste in a repository without the intention of retrieval or the approved direct discharge of waste into the environment with subsequent dispersion.

Disposition

An act to determine how a departure from a specified requirement is to be handled or settled.

District Authority

A person notified by the appropriate government(s), with jurisdiction over the area outside the exclusion zone of the nuclear/radiation facility and who is having responsibility for coordinating the activities of various government agencies for protecting the public and the environment in case of an off-site emergency.

Diversity

The presence of two or more different components or systems to perform an identified function, where the different components or systems have different attributes, so as to reduce the possibility of common cause failure.

Document

Recorded or pictorial information describing, defining, specifying, reporting or certifying activities, requirements, procedures or results.

Domain (Radiological Emergency)

Region bound by time-space considerations is categorised on the basis of radiological characteristics.

Dose

A measure of the radiation received or absorbed by a target. The quantities termed absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose, or committed effective dose are used, depending on the context. The modifying terms are used when they are not necessary for defining the quantity of interest.

Dose Limit

The value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded.

Earthquake

Vibration of earth caused by the passage of seismic waves radiating from the source of elastic energy.

Effluent

Any waste discharged into the environment from a facility, either in the form of liquid or gas.

Electrical Protection System

A part of electrical system which protects an equipment or system. This encompasses all the electrical, electronic, mechanical, thermal, pneumatic devices and circuitry, including the sensors which generate the input signal for protection logic.

Electrical Separation

Means of preventing one electric circuit from influencing another through electrical phenomena.

Emergency

A situation which endangers or is likely to endanger safety of the site personnel, the nuclear/radiation facility or the public and the environment.

Emergency Electric Power System

The portion of the electrical power system, which is provided for the purpose of supplying electric power to safety related and safety systems during operational states, as well as during and following accident conditions.

Emergency Exercise

A test of an emergency plan with particular emphasis on coordination of the many inter-phasing components of the emergency response, procedures and emergency personnel/agencies. An exercise starts with a simulated/postulated event or series of events in the plant in which an unplanned release of radioactive material is postulated.

Emergency Plan

A set of procedures to be implemented in the event of an accident.

Emergency Planning Zone (EPZ)

The zone defined around the plant upto 16 km radius providing a basic geographic framework for decision making on implementing measures as part of a graded response in the event of an off-site emergency.

Engineered Safety Features (ESF)

The system or features specifically engineered, installed and commissioned in a nuclear power plant to mitigate the consequences of accident condition and help to restore normalcy, e.g. containment atmosphere clean-up system, containment depressurisation system, etc.

Environment

Everything outside the premises of a facility, including the air, terrain, surface and underground water, flora and fauna.

Environmental Conditions

Parameters such as pressure, temperature, humidity, chemical spray, flooding, and radiological conditions associated with operational states and accident conditions.

Environmental Monitoring

The measurement of external dose rates due to sources in the environment or of radionuclide concentrations in environmental media.

Equilibrium Core

The condition of the core of an operating reactor in which the rate of charging and discharging of the fuel in the core, averaged over a sufficiently long period of time, reaches and remains close to design value.

Evacuation

The temporary removal of persons from locations where dose rates or projected doses arising in an emergency situation are unacceptably high, or where the avertable dose exceeds the relevant intervention level.

Event

Occurrence of an unplanned activity or deviations from normalcy. It may be an occurrence or a sequence of related occurrences. Depending on the severity in deviations and consequences, the event may be classified as an anomaly, incident or accident in ascending order.

Exclusion Zone

An area extending upto a specified distance around the plant, where no public habitation is permitted. This zone is physically isolated from outside areas by plant fencing and is under the control of the plant management.

Exemption Level

A value, established by regulatory body and expressed in terms of activity concentration and/or total activity, at or below which a source of radiation may be granted exemption from regulatory control without further consideration.

Explosion

An abrupt oxidation or decomposition reaction producing an increase in temperature, or in pressure, or in both simultaneously.

Exposure

The act or condition of being subject to irradiation. Exposure can be either external (irradiation by sources outside the body) or internal (irradiation by sources inside the body). Exposure can be classified as either normal exposure or potential exposure; either occupational, medical or public exposure; and in intervention situations, either emergency exposure or chronic exposure. The term 'exposure' is also used in radiation dosimetry to express the amount of ions produced in air by ionising radiation.

External Events

Events unconnected with the operation of a facility which could have an effect on the safety of the facility.

Fail Safe Design

A concept in which, if a system or a component fails, then the plant/component/ system will pass into a safe state without the requirement to initiate any operator action.

Fresh Core

The condition of the core after initial loading, which contains all fresh bundles with zero burnup.

Fuel Failure (Failed Fuel)

A fuel bundle having failure of clad or end-plug in one or more fuel elements, leading to release of radioactive material.

Fuel Handling

All activities relating to receipt, inspection, storage and loading of unirradiated fuel into the core and unloading of irradiated fuel from the core, its transfer, inspection, storage and dispatch

Full Power

The rated thermal power of the reactor, i.e. the gross fission power as established by the station heat balance, using approved methodology from the nuclear power plant.

Grading (QA)

Category or rank given to entities having the same functional use but different requirements for quality.

High Level Waste (HLW)

A type of waste, which contains any of the following:

- The radioactive liquid containing most of the fission products and actinides present in spent fuel, which forms the residue from the first solvent extraction cycle in reprocessing, and some of the associated waste streams;

- Solidified high level waste from above and spent reactor fuel (if it is declared as waste);
- Any other waste with similar radiological characteristics.

Incident

Events that are distinguished from accidents in terms of being less severe. The incident, although not directly or immediately affecting plant safety, has the potential of leading to accident conditions with further failure of safety system(s).

Independence

The ability of equipment, channel or system to perform its function irrespective of the normal or abnormal functioning of any other equipment, channel or system. Independence is achieved by functional isolation and physical separation.

In-service Inspection (ISI)

Inspection of structures, systems and components carried out at stipulated intervals during the service life of the plant.

Integrated Leakage Rate Test (Containment)

The leakage test performed on the containment by pressurising the same to particular leakage rate test pressure, and determining the overall integrated leakage rate.

Leakage

The quantity of fluid escaping from a leak.

Leak Tightness

The ability of a component to maintain leakage rate within a prescribed value.

Licence

A type of regulatory consent, granted by the regulatory body for all sources, practices and uses for nuclear facilities involving the nuclear fuel cycle and also certain categories of radiation facilities. It also means authority given by the regulatory body to a person to operate the above said facilities (see "Licenced Person" and "Licenced Position").

Licenced Person

A person who has been licenced to hold certain licensed position of a nuclear power plant after due compliance with authorised procedure of certification by the regulatory body.

Licenced Position

A position, which can be held only by person certified by the regulatory body or a body, designated by it.

Limiting Conditions for Operation (LCO)

Conditions that are imposed on operation which are intended to ensure safety during startup, normal operation and shutdown. They also help to avoid reaching the limiting safety system settings and ensure readiness for performing necessary functions in the event of an accident. LCO include limits of operating parameters, requirements of minimum operable equipment of various systems, minimum specified staffing as well as prescribed actions to be taken by operating staff.

Limiting Safety System Settings (LSSS)

Settings on instrumentation, which initiate the automatic protection action at a level such that the safety limits are not exceeded.

Liquefaction (of Soil)

Sudden loss of shear strength and rigidity of saturated and cohesionless soils due to vibratory ground motion.

Loss of Coolant Accident (LOCA)

An accident resulting from the loss of coolant to the fuel in a reactor due to a break in pressure retaining boundary of the primary coolant system.

Main Control Room (MCR)

Room all the time supervised by the licensed personnel for operation of the facility and performing intended functions for all the operational states of the facility.

Maintenance

Organised activities covering all preventive and remedial measures, both administrative and technical, to ensure that all structures, systems and components are capable of performing as intended for safe operation of the plant.

Main Steam Line Break

A break in steam pipeline which leads to discharge of high enthalpy steam.

Mean Sea Level (MSL)

The average height of the surface of the sea for all stages of the tide determined from hourly height readings over a long period.

Member of the Public

Any individual in the population except for one who is subject to occupational or medical exposure. For the purpose of verifying compliance with the annual dose limit for public exposure, the member of the public is the representative individual in the relevant critical group.

Missile

A mass that has kinetic energy and has left its design location.

Mitigation

Process of minimising the severity of a consequence following an incident/accident.

Monitoring

The continuous or periodic measurement of parameters for reasons related to the determination, assessment in respect of structure, system or component in a facility or control of radiation.

Near Surface Disposal

Disposal of waste with/without engineered barriers, or below the ground surface with adequate final protection covering to bring the surface dose rate within prescribed limits.

Normal Exposure

An exposure which is expected to be received under normal operating conditions of an installation or a source, including possible minor mishaps that can be kept under control.

Normal Operation (NO)

Operation of a plant or equipment within specified operational limits and conditions. In case of a nuclear power plant, this includes, start-up, power operation, shutting down, shutdown state, maintenance, testing and refuelling.

Normal Power Supply

Power supply derived from the grid via transmission lines or the plant generator or a combination of these for supply of electrical power to equipment in nuclear power plant.

Nuclear Facility (NF)

All nuclear fuel cycle and associated installations encompassing the activities from the front end to the back end of nuclear fuel cycle processes and also the associated industrial facilities such as heavy water plants, beryllium extraction plants, zirconium plants, etc.

Nuclear Power Plant (NPP)

A nuclear reactor or a group of reactors together with all the associated structures, systems, equipment and components necessary for safe generation of electricity.

Nuclear Safety

The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards.

Nuclear Security

All preventive measures taken to minimize the residual risk of unauthorised transfer of nuclear material and/or sabotage, which could lead to release of radioactivity and/or adverse impact on the safety of the plant, plant personnel, public and environment.

Objective Evidence (Quality Assurance)

Qualitative or quantitative information, record or statement of fact pertaining to quality of an item or service which is based on observation, measurement or test and which can be verified.

Off-site

Area in public domain beyond the site boundary.

Off-site Emergency

Accident condition/emergency situation involving excessive release of radioactive materials/hazardous chemicals from the plant to the public domain calling for intervention.

Operating Basis Earthquake (OBE)

An earthquake which, considering the regional and local geology and seismology and specific characteristics of local sub-surface material, could reasonably be expected to affect the plant site during the operating life of the plant. The features of a nuclear power plant necessary for continued safe operation are designed to remain functional, during and after the vibratory ground motion caused by the earthquake.

Operating Organisation

The organisation so designated by responsible organisation and authorised by the regulatory body to operate the facility.

Operation

All activities following and prior to commissioning performed to achieve, in a safe manner, the purpose for which a nuclear/radiation facility is constructed, including maintenance.

Operational Limits and Conditions (OLCs)

Limits on plant parameters and a set of rules on the functional capability and the performance level of equipment and personnel, approved by the regulatory body, for safe operation of the nuclear/radiation facility (see also ‘’Technical Specifications for Operation’’).

Operational Records

Documents such as instrument charts, certificates, log books, computer printouts and magnetic tapes, made to keep objective history of the operation of nuclear/radiation facility.

Operational States

The states defined under “normal operation” and “anticipated operational occurrences”.

Personnel Emergency

Emergency resulting in serious injury and/or excessive contamination of personnel involving radioactive/toxic chemicals.

Physical Barrier

A fence or wall or a similar impediment, which provides penetration delay and complements access control.

Physical Protection

Measures for the protection of nuclear/radiation facility designed to prevent unauthorised access or removal of radioactive material, or sabotage.

Physical Separation

A means of ensuring independence of equipment through separation by geometry (distance, orientation, etc.), appropriate barriers or a combination of both.

Planned Exposure

Exposure of a radiation worker beyond the normally permitted exposures with prior authorisation.

Plant Emergency

Declared emergency conditions in which the radiological/other consequences, confined to the plant or a section of the plant, requiring immediate operator action.

Plant Management (PM)

Members of the site personnel who have been delegated responsibility and authority by the operating organisation for directing the operation of the plant.

Poison (Neutron Poison)

A substance used to reduce reactivity in a reactor core, by virtue of its high neutron absorption cross-section.

Postulated Initiating Events (PIE)

Identified events during design that lead to anticipated operational occurrences or accident conditions, and their consequential failure effects.

Power Operation

Operation at a power level exceeding the conditional trip values as stipulated by the regulatory body for plant operation.

Preliminary Safety Analysis Report (PSAR)

Safety analysis report submitted to regulatory body for obtaining consent for construction.

Prescribed Limits

Limits established or accepted by the regulatory body.

Pre-Service Inspection (PSI)

The inspection performed prior to or during commissioning of the plant to provide data on initial conditions supplementing manufacturing and construction data as a basis for comparison with subsequent examinations during service.

Preventive Action

Action to eliminate the cause of a potential nonconformity (non-fulfillment of requirements) or other undesirable situation.

Primary Containment

The principal structure of a reactor unit that acts as a pressure retaining barrier, after the fuel cladding and reactor coolant pressure boundary, for controlling the release of radioactive material into the environment. It includes containment structure, its access openings, penetrations and other associated components used to effect isolation of the containment atmosphere.

Probable Maximum Flood (PMF)

The postulated flood (characterised by peak flow, volume and hydrograph shape) that is considered to be most severe but reasonably possible, corresponding to the probable maximum precipitation.

Probable Maximum Precipitation (PMP)

The estimated depth of precipitation for a given duration, drainage area and time of year of which there is virtually no risk of exceeding. The probable maximum precipitation for a given duration and drainage area approaches and approximates to that maximum which is thought to be physically possible within the limits of contemporary hydro-meteorological knowledge and techniques.

Probable Maximum Water Level

A hypothetical water level (exclusive of wave run-up from normal wind-generated waves) that might result from a most severe combination of hydrological, meteorological, geo-seismic and other geophysical factors that is considered reasonably possible in the region involved, with each of these factors considered as affecting the locality in a maximum manner.

Process Systems

Nuclear and conventional systems required for operation as per the design intent.

Protection System

A part of the safety critical system which encompasses all those electrical, mechanical devices and circuitry, from and (including the sensors) upto the input terminals of the safety actuation system and the safety support features, involved in generating the signals associated with the safety tasks.

Protective Barrier or Shielding (Radiation)

A barrier of appropriate thickness used to reduce radiation levels to specified values.

Public Exposure

Exposure incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation, but, including exposure from authorised sources and practices and from intervention situations.

Qualified Person

An individual who, by virtue of certification by appropriate authorities and through experience, is duly recognised as having expertise in a relevant field of specialisation like quality assurance, radiation protection, plant operation, fire safety or any relevant engineering or safety speciality.

Quality Assurance (QA)

Planned and systematic actions necessary to provide the confidence that an item or service will satisfy given requirements for quality.

Quality Control (QC)

Quality assurance actions, which provide means to control and measure the characteristics of an item, process or facility in accordance with the established requirements.

Radioactive Waste

Material, whatever its physical form, left over from practices or interventions for which no further use is foreseen: (a) that contains or is contaminated with radioactive substances and has an activity or activity concentration higher than the level for clearance from regulatory requirements, and (b) exposure to which is not excluded from regulatory control.

Radioactive Waste Management Facility

Facility specifically designed to handle, treat, condition, temporarily store or permanently dispose of radioactive waste.

Radiological Protection

The protection of people from the effects of exposure to ionising radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur.

Reactivity

A measure of the deviation from the criticality (defined as 'ρ') of a nuclear chain reacting medium. Reactivity 'ρ', is related with effective multiplication factor 'k_{eff} 'by the relation

$\rho = (k_{eff}-1)/k_{eff}$

Reactivity is expressed in terms of mk (10^{-3} k). Other units used are dollar, cent, in hour and pcm.

Reactor Building

The concrete containment structure that contains and supports the reactor and other related systems, such as the heat transport system, the moderator system.

Reactor Startup State

A subcritical state of reactor in which reactor protection system is poised, reactor regulating system is in specified state and positive reactivity addition is permitted to achieve reactor criticality.

Reactor Trip

Actuation of a shutdown system to bring the reactor to shutdown state.

Redundancy

Provision of alternative structures, systems, components of identical attributes, so that anyone can perform the required function, regardless of the state of operation or failure of the other.

Responsible Organisation (RO)

An organisation having overall responsibility for siting, design, construction, commissioning, operation and decommissioning of a facility.

Review

Documented, comprehensive and systematic evaluation of the fulfillment of requirements, identification of issues, if any.

Safe Shutdown Earthquake (SSE)

The earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology and specific characteristics of the local sub-surface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems and components are designed to remain functional. These structures, systems and components are those which are necessary to assure

- the integrity of the reactor coolant pressure boundary; or
- the capability to shutdown the reactor and maintain it in a safe shutdown condition; or
- the capability to prevent the accident or to mitigate the consequences of accidents which could result in potential off-site exposures higher than the limits specified by the regulatory body; or
- the capacity to remove residual heat.

Safety Analysis

Evaluation of the potential hazards (risks) associated with the implementation of a proposed activity.

Safety Analysis Report (SAR)

A document, provided by the applicant/consentee to the regulatory body, containing information concerning the nuclear or radiation facility, its design, accident analysis and provisions to minimise the risk to the public, the site personnel and the environment.

Safety Classification

Classification of structures, systems and components based on their nuclear safety functions.

Safety Code

A document stating the basic requirements, which must be fulfilled for particular practices or applications. This is issued under the authority of the regulatory body and mandatory to be followed by the respective utilities.

Safety Guide

A document containing detailed guidelines and various procedures/ methodologies to implement the specific parts of a safety code, that are acceptable to the regulatory body, for regulatory review. This is issued under the authority of regulatory body and is of non-mandatory nature.

Secondary Containment

The structure surrounding the primary containment that acts as a further barrier to limit the release of radioactive materials and also protects the primary containment from external effects. It includes secondary containment structure and its access openings, penetrations and those systems or portions thereof, which are connected to the containment structure.

Segregation (Radioactive Waste)

An activity where waste or materials (radioactive and exempt) are separated or are kept separate according to radiological, chemical and/or physical properties to facilitate waste handling and/or processing. It may be possible to segregate radioactive material from exempt material and thus reduce the waste volume.

Seiche

An oscillation of an enclosed water body in response to a disturbing force (seismic or atmospheric) having the same frequency as the natural frequency of the water body.

Seismic Hazard

Any physical phenomenon (e.g. ground vibration, ground failure) associated with an earthquake that may produce adverse effects.

Setback

Controlled gradual reduction in reactor power effected by the reactor regulating system in response to an identified abnormality in one or more plant process variables, until the conditions causing the setback are cleared or the preset limit for power rundown is reached.

Shutdown Margin

The minimum specified sub-criticality of a reactor under shutdown condition at any time during the operation from the most reactive state of the core or under postulated failure of a specified number of shutdown devices of the highest reactivity worth(s) for the given shutdown system.

Shutdown State

State of a reactor when it is maintained subcritical with specified negative sub-criticality margin.

Significant Events (Nuclear Facility)

Unusual occurrences exceeding the limits and conditions stipulated by the regulatory body.

Single Failure

A random failure, which results in the loss of capability of a component to perform its intended safety function. Consequential failures resulting from a single random occurrence are considered to be part of the single failure.

Site

The area containing the facility defined by a boundary and under effective control of the facility management.

Site Emergency

Accidental condition/emergency situation in the plant involving radioactivity transgressing the plant boundary but confined to the site, or involving release of hazardous chemicals or explosion, whose effects are confined to the site, with off-site consequences being negligible.

Siting

The process of selecting a suitable site for a facility including appropriate assessment and definition of the related design bases.

Source

Anything that causes radiation exposure, either by emitting ionising radiation or releasing radioactive substances or materials.

Spent Fuel

Irradiated fuel not intended for further use in reactors in its present form.

Station Blackout (SBO)

The complete loss of both off-site and on-site AC power supplies.

Storage (Radioactive Waste)

The placement of radioactive waste in an appropriate facility with the intention of retrieving it at some future time. Hence, waste storage is by definition an interim measure and the term interim storage should not be used.

Storm

Violent disturbance of the atmosphere marked by wind and usually by rain, snow, hail, sleet or thunder and lightning.

Storm Surge

A rise above normal water level on the open coast due to the action of wind stress on the water surface together with the atmospheric pressure reduction caused by a cyclone.

Structural integrity

The ability of a structure to withstand prescribed loads.

Supervised Area

Any area not designated as a controlled area but for which occupational exposure conditions are kept under review even though specific protective measures and safety provisions are not normally needed.

Supplementary Control Room (SCR)

Separate room provided in the nuclear power plant so that in case of non-availability of main control room in emergency situations viz. fire, the main safety functions of reactor i.e. shutting down the reactor, maintain it in safe shutdown state, decay heat removal, containment isolation and monitoring of essential reactor parameters, are performed from another room having independent sensors, power supplies, ventilation and access.

Surveillance

All planned activities, viz. monitoring, verifying, checking including in-service inspection, functional testing, calibration and performance testing carried out to ensure compliance with specifications established in a facility.

Technical Specifications for Operation

A document approved by the regulatory body, covering the operational limits and conditions, surveillance and administrative control requirements for safe operation of the nuclear or radiation facility. It is also called as “operational limits and conditions”.

Test

An experiment carried out in order to measure, quantify or classify a characteristic or a property of an entity.

Topography

The configuration of a terrain giving general description of physical features like hills, valleys, slopes, water bodies and other man-made structures.

Tropical Storm

An intense tropical cyclone in which winds tend to spiral inward towards a core of low pressure, with maximum surface wind velocities that are less than 120 km/h for several minutes or longer at some points.

Tsunami

A wave train produced by impulsive disturbances in a body of water caused by displacements associated with submarine earthquakes, volcanic eruptions, submarine slumps or shoreline slides.

Ultimate Heat Sink

The atmosphere or a body of water or the ground water to which a part or all of the residual heat is transferred during normal operation, anticipated operational occurrences or accident conditions.

Unavailability

The inability of an entity to be in a state to perform a required function under given conditions at a given point of time. It is measured as the probability (relative frequency) that the entity is in an unavailable state at a point of time.

Validation (Computer Code)

The evaluation of software at the end of the software development process to ensure compliance with the user requirements. Validation is therefore 'end-to-end verification'.

Verification (Computer Code)

The process of determining that the controlling physical and logical equations have been correctly translated into computer code.

Waste Management

All administrative and operational activities involved in the handling, pre-treatment, treatment, conditioning, transportation, storage and disposal of radioactive waste.

Waste Treatment

Operations intended to benefit safety and/or economy by changing the characteristics of the wastes by employing methods such as

(a) volume reduction;

(b) removal of radionuclides;

(c) change of composition.

After treatment, the waste may or may not be immobilised to achieve an appropriate waste form.

SPECIAL DEFINITION

Additional Safety Systems/Features

Item designed to perform a safety function or which has a safety function, in design extension conditions without core melt.

Beyond Design Basis Accident (BDBA)

This term is superseded by design extension conditions.

Complementary Safety Features

A design feature outside of the design basis envelope that is introduced to cope with design extension conditions with core melt/severe accidents.

Design Authority

The defined function of a licensee's organisation with requisite knowledge and with responsibility for maintaining the design integrity and the overall basis for safety of its nuclear facilities throughout the full lifecycle of those facilities. Design authority relates to the attributes of an organisation rather than the capabilities of individual post holders.

Design Extension Conditions (DEC)

Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.

First-Of-A-Kind (FOAK) system

A system being implemented for the first time for the reactor type which may have a new, unique design feature, and is yet to demonstrate its in-plant functionality.

Plant States

<table>
<tr><th colspan="2">Operational States</th><th colspan="3">Accident Conditions</th><th>Practically Eliminated</th></tr>
<tr><td rowspan="2">Normal Operations</td><td rowspan="2">Anticipated Operational Occurrences</td><td rowspan="2">Design Basis Accidents</td><td colspan="2">Design Extension Conditions</td><td>Large release of radioactivity from containment</td></tr>
<tr><td>Accidents without Core melt</td><td>Accidents with Core melt</td><td></td></tr>
</table>

Safety Support System

A system designed to support the operation of one or more safety systems.

Safety System

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

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CONTENTS

1. INTRODUCTION

1.1 General

The safety analysis report (SAR) of a nuclear power plant (NPP) is the main safety document for design, construction, commissioning, operation and decommissioning that contains all relevant information to demonstrate the overall safety of the NPP. SAR is prepared by the applicant for submission to the regulatory body to enable it to assess the suitability of the plant for consenting.

The SAR should provide information, such as design bases, site and plant characteristics, safety analyses and conduct of operations, in such a way that facilitates AERB to assess and review the safety of the plant. It should provide adequate justification that a nuclear power plant (NPP) meets all applicable safety requirements, and that the plant has been designed, built and commissioned as intended, and that all design, construction and commissioning changes have been properly addressed.

In general, the SAR contains information on general description of the plant, site characteristics, description of individual systems, structures and components (SSCs) including criteria for mechanical components, civil structures, electrical systems, instrumentation and control systems. 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 references for the safe operation. The justification of safe operation is given by safety analysis (deterministic and probabilistic), operational and administrative measures brought out in the SAR.

1.2 Objective

The objective of this Guide is to provide guidance for development of the SAR by the applicant by bringing out the possible content and structure of a SAR. Use of this format will help ensure the completeness of the information provided. The Guide is intended to indicate a standard format of SAR and assist in checking completeness and adequacy of the SAR by regulatory body.

1.3 Scope

This guide is applicable for preparation of SAR (Preliminary and Final) of all types of NPPs design. This Guide aims to be a technology neutral document so as to cater for different types of reactor design. However, in case, some sections and sub-sections are not directly applicable for a reactor type, technically applicable equivalent systems should be described or the non-applicability of the same should be brought out in SAR.

Although this guide is intended mainly for preparation of SAR for new plants, the guidance presented here may also be considered for existing NPPs when the responsible organizations periodically review their existing SARs.

2. GENERAL CONSIDERATIONS

2.1 Aspects of the Safety Analysis Report (SAR) and its Format

a) The SAR (Preliminary and Final) should document the safety of an NPP in sufficient scope and detail to support the conclusions reached and to provide an adequate input to enable regulatory review including independent verification.

b) Preliminary Safety Analysis Report (PSAR) chapters are submitted for review and assessment of adequacy of safety during various stages of consenting process as elaborated in AERB Safety Guide 'Consenting Process for Nuclear Power Plants and Research Reactors' [AERB/NPP&RR/SG/G-1; 2007] [1].

c) Final Safety Analysis Report (FSAR) is submitted for consent of operation. This should be the updated version of the PSAR incorporating the changes/feedback during construction and commissioning stages reflecting as-built plant with appropriate reference.

d) The information presented in the SAR should be clear, concise and complete in itself. Each subject should be treated in sufficient depth and should be documented to permit a reviewer to evaluate the safety level independently. Tables, drawings, sketches, plots and figures should be used wherever they contribute to the clarity and conciseness of the report. All information presented in drawings, maps, diagrams, sketches and charts should be legible. The symbols should be defined.

e) Contents of some section of SAR may be of confidential/proprietary nature. Their uses will be governed by the confidentiality agreement between the applicant and AERB.

f) The cover page should indicate the name of the reactor type, organisation submitting the report, approved by and submitted for etc. The subsequent pages should contain appropriate information for traceability of the page.

g) A table of contents should be provided. When a report consists of several volumes, at least an abridged table of contents should be included in each volume.

h) Abbreviations used should be consistent with the general usage and those not in general usage should be defined in each volume where they are used.

i) Removal and reinsertion of a page(s) and insertion of a modified page(s) should be easy with due reference to the amended version.

j) Reports or other documents that are referenced in the text of the SAR should be listed at the end of the Chapter in which they are referenced. In cases where proprietary documents are referenced, a non-proprietary summary of the document should also be referenced.

k) In the electronic form of SAR, internal reference links (detailed design documents, references to standards, detailed analysis reports, code validation reports, source material for PSA, etc.) should be provided.

2.2 Chapters of Safety Analysis Report (SAR)

SAR should contain the following 20 chapters,

Chapter 1	Introduction and General Description of Plant
Chapter 2	Site Characteristics
Chapter 3	Design of Structures, Systems and Components
Chapter 4	Reactor
Chapter 5	Reactor Coolant and Reactor Auxiliary Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Control
Chapter 8	Electrical Power Supply Systems
Chapter 9	Plant Auxiliary Systems
Chapter 10	Steam and Power Conversion System
Chapter 11	Radioactive Waste Management
Chapter 12	Radiation Protection
Chapter 13	Conduct of Operation
Chapter 14	Commissioning
Chapter 15	Accident Analysis
Chapter 16	Technical Specifications for Operation
Chapter 17	Management of Safety and Quality Assurance
Chapter 18	Human Factor Engineering
Chapter 19	Probabilistic Safety Assessment
Chapter 20	Decommissioning

The content of each chapter of SAR mentioned above are described subsequently in Section 3 of this guide and a table of content for each chapter of SAR is given in Appendix-A. The format and content of SAR should adhere to the same. While normally the suggested format and coverage should be adhered to, at times there can be deviations to ensure systematic and logical presentation of information associated with the evaluation of individual safety aspects peculiar to that issue. When a particular section/sub-section of the given format is not applicable to the type of reactor design, it can be mentioned as "Not Applicable". A general structure that is to be followed for relevant section/sub-sections describing with systems or equipment is given in Section 2.3 of this guide.

Any other national or international standard format for presenting the information to be contained in SAR is acceptable to AERB, provided the information contained in the submission is adequately covered as indicated in this guide. However, if need arises content may be modified to suit the specific design. In such case, the applicant should provide a mapping (chapter/section) of the information in various chapters of the SAR vis-a–vis the content of the guide.

2.3 Unified Description for Systems and Equipment

A common structure for description of systems and equipment, and the intended content is given below. When a topic is not applicable to a system or item of equipment, the intent is to keep the section and report as "Not Applicable."

1. System / Equipment Functions

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

2. Safety Design Bases and Design Parameters

Principal design criteria including the fundamental architectural and engineering design objectives, established for the project are described here. It represents the broad frame of reference within which the more detailed plant design effort is to proceed and against which the project will be reviewed.

Design Bases comprise information which identifies the specific functions to be performed by a major component, system or structure in terms of performance objectives, together with specific values or range of values, chosen for controlling parameters as reference bounds or limits for design.

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

a) design codes used (mechanical, electrical, instrumentation)
b) postulated initiating events,
c) safety requirements related to various operating conditions, including stresses and environmental conditions (temperature, humidity, etc.),
d) design limits,
e) safety, seismic and quality classification and its compliance,
f) protection against external hazards,
g) protection against internal hazards,
h) single failure criterion and protection against common cause failures,
i) isolation of system and equipment,
j) equipment qualification,
k) design standards and fabrication codes, etc. and other more specific design aspects such as:
   a. overpressure protection,
   b. thermal shock and pressure wave,
   c. leakage detection(including leak before break, as applicable),
   d. internally generated missile.

3. System Description

In this section, the equipment/system/structure and its components are described. The description should include principal dimensions, sketches, layout with locations of system/equipment, simplified process and instrumentation etc. as applicable.

4. Materials

In this section, adequate information should be provided regarding the materials used in components, their specific properties, and quality and chemistry requirements of environment. Ageing and degradation processes should also be taken into account. The corresponding specific testing and surveillance programmes for the materials are to be presented as a complement to item no. 8 below concerning the whole equipment/system monitoring.

5. **Interfaces and interaction with other Systems and Equipment**

The support systems (e.g. electric power, civil structures, fire protection, cooling water, compressed air etc.) and other connected systems are to be presented along with corresponding design requirements.

6. **System / Equipment Operation**

The operation of the system or equipment including support systems is summarized in this section. This should specify the various operating states namely normal operation and off normal operations.

7. **Instrumentation and Control**

This section describes the method of control, logics, the trip and alarms, indications and interlocks associated with safe operation as well as consequences of mal-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:

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

9. **Radiological Aspects**

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

10. **Performance and Safety Evaluation**

This section describes a study of the functional and physical features of the major plant structure/systems and components to determine:

a) whether the design can or has met performance objectives with an adequate margin of safety, and
b) Susceptibility to failures, either in equipment/structure or control over process variables, which could be a possible initiating event for accidents.

This section presents the measures taken to address each of the safety design aspects or requirements. In addition, the design issues that may be raised in the other sections above are also addressed here.

11. **Built in Safety Features**

The built in safety features, if any, along with the intent and principle of operation should be described in this section.

12. **System Commissioning**

The procedure and requirements to be followed and the process by which the system/equipment/components that are assembled/erected are made functional and verified to be in accordance with the design specifications are to be described.

13. Compliance with Applicable Clauses of AERB Design Code

This section should elaborate on the compliance with applicable design codes and justification for non-compliance, if any.

14. Feedback and Comparison with Similar Design

Comparison and experience feedback aspects with similar design should be described.

15. References

The list of reference documents should be provided and should be referred appropriately in the text.

3. STANDARD FORMAT AND CONTENTS FOR SAR CHAPTERS

The contents of Chapters 1 to 20 of SAR are described in the subsequent subsections (3.1-3.20), accordingly, sub heading under each sub-section are numbered so as to be consistent with the numbering format of the SAR Chapters. A table of content of all the Chapters of SAR is given in Appendix-A titled “Table of Contents of SAR Chapters”

3.1 Chapter 1. Introduction and General Plant Description

1.1. Introduction

This chapter of 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) A description of the structure of the SAR, the brief description, objectives and scope of each of its Chapters and the intended correlation between them, as applicable.

1.2. Identification of Stakeholders

The primary agencies responsible for the design, construction, commissioning and operation of the NPP should be specified in this section. The division of responsibility between the responsible organization (RO), design authority/group, reactor/facility designer(s), vendor, contractor and project/plant management should also be delineated.

1.3. General Plant Description

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

The section should present briefly (in a table, where appropriate) the principal elements of the installation, that includes

a) Number of units at the plant and the type of plant
b) Principal design criteria and operating characteristics (design and rated power)
c) Plant layout
d) Protection and control system
e) Classification of structures, equipment, components and systems, its bases, categories, tabulations giving detailed classification list
f) Seismic considerations
g) Type of containment structure and its design pressures
h) Reactor and auxiliary systems
i) Type of nuclear steam supply system and safety considerations
j) Engineered safety features
k) Electrical systems
l) Power conversion systems
m) Fuel handling and storage systems

n) Cooling water and other auxiliary systems
o) Radioactive waste management systems
p) Radiation protection aspects
q) Emergency plans and procedures

In addition, the thermal power levels in the core, the corresponding net electrical power output for rated thermal power level, and any other characteristics necessary for understanding the main technological processes from safety considerations, included in the design should be described.

The general arrangement of major structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. In addition, layout of equipment of safety systems and safety related systems should include, amongst other aspects, requirements of ISI, nuclear security, operational surveillance, fire safety, radiation zoning, internal events, maintainability and life extensions.

Those features of the plant likely to be of special interest because of their relationship to safety should be identified. Such items as unusual site characteristics, solutions to particularly difficult engineering problems, experiments and tests in progress for development of FOAK systems and significant extrapolations in technology represented by the design should be highlighted.

The applicant should provide estimated schedules for the completion of construction and the start of continuous operation.

1.4. Comparison with NPP of Similar Design

It is useful to compare the plant design with similar earlier designs already approved/constructed and operated, so as to identify the main differences and assist in the justification of any modifications and improvements made. This comparison should not be restricted to a comparison of reactor design parameters but should also include engineered safety features, containment concept, the instrumentation and electrical systems, the radioactive waste management system and other principal systems.

A comparison list of selected plant characteristics should be included in tabular form.

A statement of any similar (or identical) plants that has/have been already reviewed and approved by AERB or other regulatory authority elsewhere, and a statement of the specific differences and improvements that have been made since such an approval was granted should be included.

1.5. Additional Information Concerning New Safety Features

This section should describe information or provide references to information in SAR that demonstrates the performance of new safety features for NPPs that differ significantly from previous reactor designs such as use of simplified, inherent, passive, or other innovative first-of-a-kind (FOAK) features 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.

1.7. 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. Information should also be provided on alternative accesses mainly for exits. The responsible organization should provide general layout drawings for the entire plant including basic geological and seismological information related to layout of the 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 groups 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.

1.8. Principles of Safety Management

This section should briefly introduce management of safety as an integral component of the management of the responsible organization.

It should be confirmed that the plant management will be able to fulfill its responsibility towards plant safely throughout its lifetime.

1.9. Additional Documents Considered as a Part of the Safety Analysis Report

This section should provide a list of the topical reports those are referred in the SAR. The applicant is responsible for providing additional technical information that may be necessary for review of SAR. Results of tests and analyses, additional information as required may be submitted as separate reports.

1.10. Conformance with Applicable Codes, Guides and Standards

This section should provide an overview of relevant codes, standards and guides that provide the general and specific design criteria that have been used in the design. A justification of their appropriateness should be provided if codes and standards other than those of AERB are used. 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.

3.2 Chapter 2. Site Characteristics

Chapter 2 should provide information on the geological, geotechnical, seismological, hydrological and meteorological characteristics of the site and surrounding region. It should also provide information on site demography, the present and projected population distribution, nearby industries and land and water use that is relevant to the safe design and operation of the plant. Sufficient data should be included to permit an independent evaluation. Aspects with respect to environmental impact assessment (EIA) should also be provided as per format given in Section 3.24 (Chapter 24 of SAR) of this guide and reference here.

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 as a support for the design phase and periodic safety review, and may include:

a) Site specific hazard evaluation for external events (human and natural origin).
b) Design targets in terms of recurrence probability of external events.
c) Design basis for external events.
d) Basis for arriving at the finished grade levels
e) Collection of site specific data for the plant design (e.g. geotechnical, seismological, hydrological and meteorological).
f) Evaluation of the impact of the site related issues to be considered in the parts of the SAR on emergency preparedness and accident management.
g) Arrangements for the monitoring of site related parameters throughout the lifetime of the plant.
h) Collection of base line data on incidence of diseases and fatalities.

Site related information represents a very important input to the design process and final safety evaluation. Thus, utmost care should be taken to finalize these inputs.

2.1. Geography and Demography

2.1.1 Site Location and Description

This section should describe the location of each reactor at the site specified by latitude, longitude and mean sea level (MSL). The State and district (s) in which the site is located should be identified, as well as the location of the site with respect to prominent natural and man-made features such as rivers, lakes and sea boundary. A clear and legible site area map providing the following details should be included.

a) The area of plant property should be stated.
b) Location of the site boundary. If the site boundary lines are the same as the plant property lines, this should be stated.
c) The location and orientation of principal plant structures within the site area. Principal structures should be identified according to the function of their contents (e.g. reactor building, auxiliary building, turbine building, etc.).
d) The location of any industrial, commercial, institutional, recreational or residential structures within the site area.

e) The boundary lines of the plant exclusion area. The minimum distance from each reactor to the exclusion area boundary should be shown and specified.
f) A scale that will permit the measurement of distances with reasonable accuracy.
g) True north as well as plant north should be indicated.
h) National/State highways, railways, air-traffic and waterways that traverse or are adjacent to the site.

The site description should define the boundary lines of the restricted area and should describe how access to this area is controlled for radiation protection purposes. The site map discussed above may be used to identify this area, or a separate map of the site may be used. Indicate the location of the boundary line with respect to the water part of nearby rivers and lakes. Distances from plant effluent release points (Main Out-Fall) to the boundary line (Exclusion Zone boundary) along with the discharge route should be clearly specified.

2.1.2 Exclusion Area Authority and Control

The applicant should include specific description of legal rights with respect to all areas that lie within the designated exclusion area. A detailed description of the following topics should be included.

a) Arrangements for traffic control.
b) Abandonment or relocation of roads, railways and waterways.

2.1.3 Population Distribution

Population distribution (sector wise) in natural growth zone (upto 5 km) and emergency planning zone (currently 16 km) around the plant should be provided covering transient population and population density. AERB safety documents [2, 3] may be referred for further guidance. Information on food habits of the population should also be included.

2.1.4 Land and Water Use

In order that emergency measures can be implemented under accident conditions the details of use of land and water should be described. These should include:

a) Extent of agriculture land, principal food products and their yields.
b) Extent of dairy farming and yield.
c) Extent of drinking water demand and its sources in the near vicinity of the plant.
d) Extent of alternate food supply and easy access from outside.
e) Studies on all water bodies in the vicinity of the site and their outflow characteristics.
f) Use of water for drinking, irrigation, fishing, agriculture, and industry.
g) Information on alternative accesses mainly for exits.

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 (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 comparison with the acceptable limits (as specified) 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.

A list of natural, man-made, external/ internal hazards for which analysis is to be carried out should be provided.

2.3. Nearby Industrial, Transportation, and Military Facilities

This section should present the results of a detailed evaluation of the effects of potential accidents from 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 in determining the requirement of any additional design features considered necessary to mitigate the effects of the potential incidents identified. A description of projected developments having potential impact on NPP 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. 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. The extreme values of meteorological parameters, including temperature, humidity levels, rainfall, wind speeds for straight and rotational winds (such as cyclone, tornado etc.) should be evaluated in relation to the design. AERB Safety Guide on ‘Extreme Values of Meteorological Parameters’ [AERB/NF/SG/S-3] [4] can be referred. The potential for lightning 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 radionuclides to and from the site and the dispersion of radionuclides to the environment. To this end, data deriving from on-site meteorological monitoring programmes should be documented.

The applicant should use the metrology data to calculate (1) the short-term atmospheric dispersion estimates for accident releases discussed in Section 2.5.1 and (2) the long-term atmospheric dispersion estimates for routine releases discussed in Section 2.5.2.

2.5.1. Short-term Atmospheric Dispersion Estimates for Accident Releases

This section should provide, for appropriate time periods after an accident, conservative estimates of atmospheric dispersion factors (χ/Q values) at the site boundary (exclusion area) and upto EPZ, and at the control rooms for postulated accidental radioactive airborne releases.

2.5.2. Long-term Atmospheric Dispersion Estimates for Routine Releases

This section should provide realistic estimates of annual average atmospheric dispersion and deposition to a distance (as per regulatory requirements) from the plant for annual average release limit calculations and person-Sv estimates.

2.6. 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 include a consideration of effects of flood waves resulting from dam failures, river diversion and seismic events on and off the site. This section should also include the most severe drought considered reasonably possible in the region and adequacy of ultimate heat sink. 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.

2.7. 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 specific 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, soil improvements and liquefaction potential 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 neighboring plant units and other external sources, if any, should be described in sufficient detail to serve as an initial reference point and to permit AERB to develop a view of the radiological conditions at the site.

A brief 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

Emergency planning relies strongly on the availability of adequate access and exit roads, sheltering, medical facilities, animal husbandries, potential locations for sheltering during temporary evacuation and supply networks in the vicinity of the site. Many hazard scenarios for the site are expected to affects the vicinity of the site and thus jeopardizing the evacuation of personnel and 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 need for any necessary administrative measures should be identified, together with the relevant responsibilities of agencies other than the responsible organization. For emergency planning and accident management, AERB Safety Guide on ‘Site Considerations of Nuclear Power Plants for Off-Site Emergency Preparedness’ [AERB/NPP/SG/S-8] [5] can be referred.

2.10. Monitoring of Site Related Parameters

The provisions to monitor site related parameters affected by seismic, atmospheric, water, groundwater and demographic 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 modeling 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 data collected should be used in predictive methods for 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 detail in the SAR.

3.3 Chapter 3. Design of Structures, Systems and Components

This chapter of the SAR should identify, describe and discuss the principal architectural and engineering design of those structures, components, equipment, and systems important to safety. It should outline the general design criteria/requirements, codes and standards, and the approach adopted to meet the fundamental safety objective. The compliance with 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 Design Basis

The overall design philosophy, design 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 AERB safety codes'Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants'[AERB/NPP-PHWR/SC/D (Rev-1); 2009] [6] and 'Design of Light Water Reactor based Nuclear Power Plants'[AERB/NPP-LWR/SC/D; 2015] [7]. The discussion under this section should include the following.

3.1.1 Safety functions

This section should identify plant specific safety functions to fulfil the fundamental safety functions by the plant design features. The corresponding relevant SSCs necessary to fulfil these safety functions should be introduced.

3.1.2 Defence in Depth

This section should describe the approach adopted to incorporate the defence in depth concept into the design of the plant. It should be demonstrated that the defence in depth concept has been considered in all stages of the lifetime of the nuclear power plant, for all plant states and for all safety related activities.

3.1.3 Plant States and PIEs

The basis for the categorization of plant states should be explained. Postulated initiating events (whether of internal origin or caused by internal and external hazards, if relevant) should be listed. This categorization should be commensurate with the content of Chapter 15 of SAR.

This subsection should summarize the design and operational provisions implemented to demonstrate the ‘practical elimination’ of the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release. In this subsection, reference should be also made, as appropriate, to other sections of the safety analysis report (Chapter 15) where relevant confirmatory analysis is presented.

This section should summarize the approach taken to ensure adequate margins to prevent cliff-edge effects related to damage of barriers against releases of radioactive substances to the environment

3.1.4 Radiation Protection and Radiological Acceptance Criteria

This section should describe in general terms the design approach adopted to meet the fundamental safety objective and to ensure that, in all plant states, radiation doses within the installation or at Exclusion Zone boundary due to any release of radioactive material are kept below prescribed limits/acceptable levels and as low as reasonably achievable (ALARA), economic and social factors being taken into account.

3.1.5 Design Provisions for Ageing Management

This section should define the design life time of items important to safety and should describe how relevant mechanisms of ageing and wear out were taken into account in nuclear power plant design in order to ensure design life time of most important nuclear power plant components.

3.2. Classification of SSC, 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 (SSCs). It should include information on the methods used to ensure the following safety functions 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).

a) To permit shutdown of the reactor and maintenance in the safe shutdown condition.
b) To prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary.
c) To contain radioactive material.

This section should identify those structures, systems and components important to safety that are designed and/or qualified to withstand the effects of a Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). All structures, systems and components or portions thereof, which are intended to be designed for OBE, should be listed or otherwise clearly identified.

If there is a potential for structures or systems to interact, then details should be provided here, to reflect 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. AERB Safety Guide on 'Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy Water Reactors' [AERB/NPP-PHWR/SG/D-1] [8] can be referred. A list of safety related systems and main structures and components, with their classifications and categorization, should be included or reference here.

Combinations of various kinds of loads and corresponding allowable stresses should also be described here.

3.3. Protection against External Hazards

This section should provide a list of external hazards(flood, wind and cyclone, tsunami, explosion, aircraft crash etc., as applicable),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 external hazards considered in the plant design.

3.4. Protection against Internal Hazards

This section should provide a list of internal hazards(internal flood, fire, explosion, internal missiles, drop loads etc., as applicable), 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 internal hazards considered in the plant design. For further guidance, AERB Safety Guide on 'Protection against Internally Generated Missiles in Nuclear Power Plants' [AERB/NPP/SG/D-3] [9] can be referred.

3.5. Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

This section should describe design bases and design measures to ensure that the containment and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary and safety related instrumentation and control panels have been adequately protected against the effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located either inside or outside of containment. The information should also include determination of break locations and dynamic effects associated with the postulated rupture of piping.

3.6. Seismic Design

This section should provide the relevant information on seismic design and analysis of SSCs. AERB Safety Guide on 'Seismic Qualification of Structures, Systems and Components of Pressurised Heavy Water Reactors' [AERB/NPP-PHWR/SG/D-23; 2009] [10] can be referred.

3.6.1 Seismic Input

The description of the seismic design should consider the following inputs:

a) Design response spectra
b) Design time-history
c) Damping values
d) Supporting media for seismic category-1 structures
e) Soil-structure interaction
f) Uplift analysis.

3.6.2 Seismic System Analysis

This section should discuss the seismic system analyses applicable to structures, systems and components important to safety. The specific information should include the following:

a) Seismic analysis methods
b) Procedure used for modeling
c) Determination of overturning moments of structures important to safety
d) Damping value used
e) Method used to account for torsional effects
f) Load path under gravity and seismic loads
g) Development of floor response spectra
h) Three components of earthquake motion
i) Natural frequencies and response loads
j) Combination of modal responses
k) Interaction of lower seismic category structures with higher seismic category structures
l) Effects of parameter variations on floor response spectra
m) Use of constant vertical static factors
n) Comparison of responses using response spectra and time history methods (wherever used) at selected points.

3.6.3 Seismic Subsystem Analysis

This section should discuss the seismic subsystem analyses applicable to structures, systems and components important to safety. The specific information on the following should be included:

a) Seismic analysis methods
b) Procedure used for modeling
c) Basis for selection of frequencies
d) Damping value used
e) Three components of earthquake motion
f) Combination of modal responses

g) Analytical procedures for piping qualification
h) Determination of number of earthquake cycles
i) Multiply supported equipment components with distinct inputs
j) Use of equivalent static load method of analysis
k) Buried piping systems and tunnels important to safety
l) Interaction of other piping with seismic category I piping
m) Torsional effects of eccentric masses
n) Use of constant vertical static factors
o) Seismic analyses for reactor internals.

The seismic subsystem analyses that will be used in establishing seismic design adequacy of the reactor internals, including fuel elements, control and protection elements and their drive mechanisms, should be described. The following information should be included:

a) Typical diagrams of dynamic mathematical modeling of the reactor internal structures to be used in the analysis.
b) Damping values and their justification.
c) A description of the methods and procedures that will be used to compute seismic responses.
d) A summary of the results of the dynamic seismic analysis.

3.6.4 Seismic Instrumentation

The proposed seismic instrumentation should be discussed in this section of SAR. This should include in particular relevant information on the following:

a) Location and Description of Instrumentation.
b) Control Room Operator Notification.
c) Comparison of Measured and Predicted Responses.

Provide the criteria and procedures that will be used to compare measured responses of structures and selected components important to safety in the event of an earthquake with the results of the seismic system and subsystem analyses.

3.6.5 Seismic Margin Assessment

Seismic margin assessment should be carried out to quantify a nuclear power plant's ability to withstand an earthquake greater than design and still meet the basic safety objectives. Margin assessment should demonstrate adequate margin over the safe shutdown earthquake (SSE), ensure plant safety and determine any “weak links” that limit the capability of NPP to safely withstand a seismic event larger than the SSE. This assessment should also include consideration of possible cliff edge effects.

Seismic margin assessment should include as minimum the following:

a) Approach and methodology followed for margin assessment
b) List of components considered in assessment and basis for arriving at the same
c) Necessary analysis and design calculations for capacity assessment of these components
d) Margin assessment of the plant
e) Weak links or limiting components.

3.7. Design of Seismic Category-1 Civil Engineering Structures

3.7.1 General Design Principles

This section should present relevant information on the design of seismic category-1 civil engineering structures. It should also briefly address 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 categorization of buildings and structures. Information should be presented covering the following, as a minimum:

a) General description: Functional and physical description of the structure, safety requirements, safety and seismic classification, layout.
b) Applicable design codes, standards and specifications.
c) Materials of construction, material properties.
d) Loads and load combinations.
e) Design and Analysis Procedures: Analysis methodology, mathematical model, design for strength and serviceability, important assumptions in analysis and design, seismic analysis methodology, soil structure interaction, structure-structure interaction, hydrodynamic aspects, design requirements pertaining to geotechnical safety and foundation design, structural acceptance criteria.
f) Construction and maintenance aspects, provisions for in-service inspection.
g) Special requirements, if any, such as tests, structural instrumentation, fire protection, decommissioning, special construction techniques.
h) Input parameters for analysis and design, calculation of design forces from analysis results, sample design calculations and specific design features, if any.
i) Bulk shielding aspects of concrete structures used as shielding for radioactive systems, components and structures also covering layout and shielding of penetrations through these shielding structures. Layout of areas/rooms containing active systems/ equipment requiring shielding should also be covered in detail with respect to structures (walls and slabs) used for shielding.

Information as specified above should be addressed separately in different sections as listed below.

3.7.2 Containment Structures

Specific design features of the primary containment such as its leak tightness, mechanical strength, pressure resistance and resistance to hazards should be covered. Concrete and steel internal structures of the containment should be described. This may include structural integrity test, and proof test for the inner containment structure. If the design incorporates a secondary containment, this should also be described here.

3.7.3 Internal Structures of Reactor Building and/or other Connected Buildings (nuclear building)

This section should describe design features of the reactor building and internal structural walls provided to comply with the applicable safety requirements.

3.7.4 Foundations of Seismic Category – I structures

Information on foundations should be provided in this section, 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 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 structures of seismic Category-I.

3.7.5 Other Seismic Category – I 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.

3.8. Mechanical Systems and Components

Relevant information on design principles and criteria, and the codes and standards used in the design of mechanical components (ASME codes, RCC-MR codes, PNAE, etc.) 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, (analyses/testing), including assumptions used in analyses to determine the structural and functional integrity of the mechanical components should be presented.

The criteria, testing procedures, and dynamic analyses employed to ensure structural and functional integrity of piping systems, mechanical equipment, and reactor internals under vibratory loadings, including those due to fluid flow and postulated seismic events, should be provided. The description should include specific information on design requirements and assessment methods for control rod drive systems, reactor vessel internals, core support systems, end shield etc., as applicable.

In addition, information on seismic qualification and testing of safety-related mechanical equipment required to ensure its functional integrity and operability during and after a postulated seismic occurrence should be provided. This should include the following information.

a) The criteria for seismic qualification, such as the deciding factors for choosing test and/or analysis, considerations in defining the input motion at the equipment monitoring locations, and the process to demonstrate adequacy of the seismic qualification program.
b) The methods and procedures used to test Seismic Category I mechanical equipment operation during and after the Safe Shutdown Earthquake (SSE) and to ensure structural and functional integrity of the equipment after several occurrences of the Operating Basis Earthquake (OBE) in combination with normal operating loads. Included are mechanical equipment such as fans, pump drives, heat exchanger tube bundles, important filters and their supports, valve actuators, battery and

instrument racks, control consoles, cabinets, panels, and cable trays. Broadband seismic excitation, dynamic coupling, and multidirectional loading effects should be considered in the development of the seismic qualification program.

c) The methods and procedures of analysis and for testing of the supports for the above Seismic Category I mechanical equipment, and the verification procedures used to account for the possible amplification of design loads (amplitude and frequency content) under seismic conditions.

The results of tests and analyses to ensure the proper implementation of the criteria accepted in the construction authorization review and to demonstrate adequate seismic qualification should be provided in the updated SAR (FSAR).

A test program should be provided that includes baseline pre-service testing and a periodic in-service test program to ensure that all Safety Class 1, 2, and 3 pumps provided with an emergency power source and all Safety Class 1, 2, and 3 valves will be in a state of operational readiness to perform their safety function throughout the life of the plant.

3.9. Seismic Qualification of Instrumentation and Electrical Equipment

All Seismic Category I instrumentation, electrical equipment, and their supports should be identified. The seismic qualification criteria applicable to the reactor protection system, engineered safety feature Class IA/EA equipment, the emergency power system, and all auxiliary safety-related systems and supports should be provided. Methods and procedures used to qualify electrical equipment, instrumentation, and their supports should also be provided.

The methods and procedures for analysis or testing of Seismic Category IA/EA instrumentation and electrical equipment supports and the verification procedures used to account for the possible amplification of design loads (amplitude and frequency content) under seismic conditions should be provided. Supports include items such as battery racks, instrument racks, control consoles, cabinets, panels, and cable trays.

3.10. Environmental Design of Mechanical, Instrumentation and Electrical Equipment

The purpose of this section is to provide information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features and reactor protection system are designed to ensure acceptable performance in all situations (e.g., normal, tests and accident). This section should also 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 the purpose when subjected to the range of individual or combined environmental challenges identified, throughout the lifetime of the plant. The acceptance criteria used for the qualification of plant items by testing or analysis 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.

The following specific information should be included concerning the design bases related to the capability of the mechanical, instrumentation and electrical portions of the engineered safety features and reactor protection system to perform their intended functions in the combined post-accident environment such as temperature, pressure, humidity, chemistry, and radiation:

a) Equipment identification and environmental conditions.
b) Necessary specification (range, mission time, diversity, seismic and environmental resistance, power source, indicator type, etc.) under which the equipment is qualified.
c) Methods/tests for equipment qualification including assessment of reliability/survivability of equipment under harsh environmental conditions.
d) Qualification test results.
e) Estimated chemical and radiation environment.
f) Electromagnetic environmental factors.

3.11. Compliance with Applicable Codes and Standards

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. The relevant codes and standard used should be cited.

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.

3.4 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 plant states. Reactor coolant and associated systems should be discussed separately in Chapter 5- Reactor Coolant and Reactor Auxiliary Systems of the SAR.

4.1 Summary Description

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

The description for each of the reactor components should follow the format prescribed in Section 2.3 of this guide.

4.2 Fuel System Design

A description should be provided of the main elements of the fuel system with justification for the design bases of the fuel system that 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.

The fuel system components would vary depending on the reactor type. Accordingly, fuel system should be described in detail. Also the basic phenomena, operating conditions, mediums in contact and their implications on fuel design performance vary depending on the reactor type. Hence, the applicable information should be provided in the relevant sub-sections of this section. A comprehensive description should be given about the fuel element/rod, internals, fuel assembly details, supporting systems, structure supporting fuel assemblies such as channels/wrappers, assembly spacing components and assembly support system such as nozzles, tie plates, spacers etc. The fuel material details including fissile, fertile, poison and absorbing material should be specified. Any material installed in the core for any other specified purpose should also be described.

The description of the fuel system mechanical design should include the following aspects:

a) Mechanical design limits such as those for allowable stresses, deflection, cycling, and fatigue,
b) Capacity for retaining fuel fission gas inventory and pressure,
c) A listing of material properties,
d) Considerations for fission product induced swelling, radiation damage, cladding collapse/rupture, materials selection and operational vibration.

Details of seismic loadings as listed in Chapter 3 on Design of Structures, Systems and Components should be referred and effects of seismic loadings and combined seismic and other applicable design basis event should be presented in this section.

The chemical design should consider all possible fuel-cladding-coolant interactions during normal and accident conditions. The description of the thermal design should include such items as maximum fuel and cladding temperatures, clad-to-fuel gap conductance as a function of burnup and operating conditions, fuel cladding integrity criteria and fuel assembly integrity criteria. The operability of the reactor with fuel rod failure should also be discussed.

4.2.1 Design Bases

The design bases for the mechanical, chemical and thermal design of the fuel system that can affect or limit the safe, reliable operation of the plant should be presented. A description and justification should be provided for the selection of design bases from the viewpoint of safety considerations. Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; where the limits extend beyond present practice, an evaluation and an explanation based on developmental work or analysis should be provided. These bases may be expressed as explicit numbers.

The description of design bases should include the functional characteristics in terms of desired performance under stated conditions. This should relate systems, components and material performance under normal operating, AOO and accident conditions.

The description should consider the following with respect to performance:

a) Cladding
b) Fuel material
c) Fuel assembly
d) Structural design
e) Thermal-hydraulic design
f) Mechanical, chemical, thermal, nuclear and irradiation properties of the materials
g) Fatigue (mechanical and corrosion) and Vibration
h) Chemical compatibility with other core components including coolant
i) Reactivity control assembly including burnable poison rods
j) Surveillance program.

4.2.2 Description and Design Drawings

A description and design drawing of the fuel rod components, burnable poison rods, fuel assemblies, and reactivity control assemblies showing arrangement, dimensions, critical tolerances, sealing and handling features, methods of support, internal pressurization, fission gas spaces, burnable poison content and internal components should be provided. Design features that prevent improper orientation or placement of fuel rods or assemblies within the core should also be included.

4.2.3 Design Evaluation

An evaluation of the fuel system design should be presented for the physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interaction. Evaluation of these interactions should include the effects of normal reactor operations, anticipated transients without scram and postulated accidents.

4.2.4 Testing and Inspection Plan

The testing and inspections to be performed to verify the design characteristics of the fuel system components, including clad integrity, dimensions, fuel enrichment, burnable poison concentration, absorber composition and characteristics of the fuel, absorber, and poison pellets should be described. Tests including non-destructive tests, fuel assembly dimensional checks and the program for inspection of new fuel assemblies and new control rods to ensure mechanical integrity after shipment should be described. Where testing and inspection programs are essentially the same as for previously accepted plants, a referenced statement to that effect with an identification of the fabricator and a summary table of the important design and performance characteristics should be provided.

4.3 Nuclear Design

4.3.1 Design Bases

The design bases for the nuclear design of the fuel and reactivity control systems should be provided and discussed, including nuclear and reactivity control limits such as excess reactivity, fuel burnup, reactivity feedback, core design lifetime, fuel replacement program, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power distribution, shutdown margins, stuck rod criteria, rod speeds, chemical and mechanical shim control, burnable poison requirements, backup and emergency shutdown provisions. Any additional applicable neutronics parameters for the reactor type under consideration may be included.

4.3.2 Description

A description of the nuclear characteristics of the design should be provided and should include the information indicated in the following sections, as applicable for the reactor type under consideration.

The nuclear characteristics of the lattice, including core physics parameters such as, effective delayed neutron fraction, neutron generation time, initial fuel loading, fuel enrichment distributions, burnable poison distributions, burnup distributions, control rod locations and refueling schemes. 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, number and location of in-core and ex-core neutron detectors used for control and protection functions should be described. The dynamic stability of the core throughout the fuel cycle, with consideration given to the possible normal and design basis operating conditions of the plant should also be described. The description should include the following elements:

a) Core design
b) Reactivity devices for control and protection
c) Description of ex-core and in-core monitoring
d) Reactor core characteristics during various phases of operation
e) Reactor startup and low power physics measurements
f) Equilibrium core and fuel management strategies
g) Power distribution
h) Bulk and spatial power control
i) Thermal power measurements
j) Flux mapping system (FMS)
k) Reactivity coefficients
l) Control and safety requirements
m) Loss of reactor regulation
n) Regional/Reactor overpower protection system
o) Control rod patterns and reactivity worth
p) Shutdown system worth insertion profile
q) Shutdown margin under normal and accidental conditions
r) Criticality of reactor during refueling

s) Irradiation-fast fluence in core components to the vessel wall
t) Neutronics stability, Xenon oscillations and control.

4.3.3 Analytical Methods

A detailed description of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients and burnup effects should be provided. Computer codes used should be described in detail as to the name and the type of code, how it is used, and its validity based on critical experiments or confirmed predictions of operating plants. Code descriptions should include methods of obtaining parameters like cross sections etc. Estimates of the accuracy of the analytical methods should be included.

4.3.4 Changes from Prior Reactor Practices

Any changes in reactor core design features, calculation methods, data or information relevant to determining important nuclear design parameters that depart from prior practice of the reactor designs should be listed along with affected parameters. Details of the nature and effects of the changes should be addressed in appropriate subsections.

4.4 Thermal and Hydraulic Design

4.4.1 Design Bases

The design bases for the thermal and hydraulic design of the reactor should be provided, including items such as maximum fuel and clad temperatures ,critical heat flux ratio (at rated power, at design overpower and during transients), flow velocities and flow distribution control, coolant and moderator voids, hydraulic stability, transient limits,. The list of design parameters and their design bases applicable for the reactor type under consideration should be presented in the beginning of this section.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

A description of the thermal and hydraulic characteristics of the reactor design should be provided and should include information as indicated below.

A summary comparison of the thermal and hydraulic design parameters of the reactor with previously approved reactors of similar design, if any, should be provided. The critical heat flux ratios for the core hot spot at normal full power and at design overpower conditions should be provided. The core-average linear heat generation rate (LHGR) and the maximum LHGR anywhere in the core should be provided. Figures/data showing the predicted radial and axial distribution of steam quality and steam void fraction in the core should be provided. Flow, pressure and temperature distributions, with the specification of limiting values and their comparison with design limits should also be provided.

The capability of the core to withstand the thermal effects resulting from the anticipated operational transients should be evaluated along with the requirements of the thermal and hydraulic stability of the core.

State the critical heat flux correlation used along with justification wherever required, analysis techniques and methods used, method of employing peaking factors and comparison with other correlations.

The analytical tools, methods and computer codes (together with verification and validation information and uncertainties) used to calculate thermal and hydraulic parameters should be presented. The uncertainties associated with estimating the peak or limiting conditions for thermal and hydraulic analysis (e.g., fuel temperature, clad temperature, pressure drops and orificing effects) should be discussed. In general the description should cover but not limited to the following aspects:

a) Critical heat Flux Ratios (CHFR)
b) Linear Heat Generation Rate (LHGR)
c) Void fraction distribution
d) Core coolant flow distribution
e) Core pressure drops and hydraulic loads
f) List of correlation and physical data
g) Thermal effects of operational transients
h) Analytical tools and uncertainties in estimates
i) Flux tilt considerations.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

The thermal and hydraulic design of the reactor coolant system should be described in this section. Information on the following should be included:

a) Plant configuration data
b) Operating restrictions on pump
c) Power-flow operating map (BWR)
d) Temperature-power operating map (PWR)
e) Power flow mapping of core subassemblies (sodium cooled fast reactors)
f) Load-following characteristics
g) Thermal and hydraulic characteristics summary table.

4.4.4 Evaluation

An evaluation of the thermal and hydraulic design of the reactor and the reactor coolant system should be provided.

4.4.5 Testing and Verification

This section should bring out the testing and verification techniques that are used to ensure that the intended thermal and hydraulic design characteristics of the core and the reactor coolant system remain within required limits throughout core lifetime.

4.4.6 Instrumentation Requirements

The functional requirements for the instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters important to safety should be discussed. The requirements for in-core instrumentation to confirm predicted power density distribution and moderator/coolant temperature distribution should be included. Internal links to the descriptions on the instrumentation design and logic in Chapter 7 of the SAR should be provided.

The equipment and the procedures used for vibration and loose-parts monitoring should be described.

4.5 Reactivity Control and Shutdown Systems

The details of reactor control and shutdown systems should be provided in this section. Description on the reactor shutdown systems including long term safe shutdown system should be provided following the unified description presented in Section 2.3 of this guide. Internal links to the description on reactor protection (I&C aspects) covered in Chapter 7 of the SAR should be provided. Reactivity control assemblies which are extending from the coupling interface of the control rod drive mechanisms should be described in all details similar to fuel system. It should be demonstrated that the Control Rod Drive Systems (CRDS), including any essential auxiliary equipment and systems that are designed and installed provide the required functional performance. Proper isolation from other equipment should also be demonstrated. Additionally, information should be presented to establish the bases for assessing the combined functional performance of all the reactivity control systems to mitigate the consequences of anticipated transients and postulated accidents. It should include the information on the following:

a) Information submitted should include relevant schematic sketches/drawings/diagrams of the rod drive mechanism, layout drawings of the collective rod drive system, process flow diagrams, piping and instrumentation diagrams, component descriptions and characteristics, and a description of the functions of all related ancillary equipment and hydraulic systems.
b) Failure mode and effects analyses of the CRDS should be presented in tabular form with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS, including all essential auxiliary equipment and hydraulic systems, can perform the intended safety functions with the loss of any single active component.
c) A functional testing program should be presented. This should include rod insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the CRDS with imposed single failures, as appropriate, generating test data on reactivity worth varying with time.
d) Preoperational and initial startup test programs should be presented. If these details are covered in Chapters on Commissioning, the appropriate sections should be referred in this section. The objectives, test methods, and acceptance criteria should be included.
e) This section should include a list of all the postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems for preventing or mitigating each accident. Internal link to chapter 15 should be provided.
f) This section should describe the relevant event 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. Reliability analysis of reactivity control systems should be detailed or reference is to be given, if presented in any other Chapter.

4.6 Reactor Internal Structures

Descriptions of the following should be provided, as applicable for the reactor type under consideration. In case, some of the systems are covered in Chapter 5, internal links should be provided:

a) 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.
b) The physical and chemical properties of the materials or relevant standards used for the reactor internal components, as well as nuclear, thermal-hydraulic, 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.
c) Any significant subsystem components, including any separate provisions for moderation and fuel location, with corresponding design drawings, surveillance and/or inspection programmes for reactor, surveillance coupons, reactor (covered in Section 5.3 of the SAR)internals to monitor the effects of irradiation and ageing on the internal components.
d) 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.

4.7 Reactor Materials

This section should provide information on the structural materials used for major components of the reactor internals including Control Rod Drive Mechanisms (CRDM).

For the purpose of this section, the control rod drive system includes the CRDM and extends to the coupling interface with the reactivity control (poison) elements in the reactor vessel. It does not include the electrical and hydraulic or other systems necessary for actuating the CRDMs.

If the type of materials, class of materials, fabrication processes, heat treatment methods and conditions are different for the different reactor components, they should be described separately. The information may include such as material specifications, non-destructive examinations, general fabrication processes and other materials, if any.

The description on material specification should include the following:

a) Provide a list of the materials and their specifications for all the components listed under reactor internals. Include materials treated to enhance corrosion resistance, strength, and hardness. Furnish information regarding the mechanical properties of any material that are not included in the acceptable codes adopted in the design, such as ASME, RCC-MR etc., of the reactor type under consideration and provide justification for the use of such material.
b) State whether any of the high strength steels or hardenable steels or cold worked steels are used in the system. If so, their usage should be identified and it should be shown that parts made of such materials do not fail during their service life due to any of the material failure mechanisms such as stress corrosion cracking, fatigue etc.

The processing and treatment of other special purpose materials such as cobalt-base alloys (Stellites), Inconels, Colmonoys, and Graphites should also be described.

3.5 Chapter 5. Reactor Coolant and Reactor Auxiliary Systems

Chapter 5 should provide relevant information on the reactor coolant system (RCS) and reactor auxiliary systems, in the prescribed format described in Section 2.3 of this guide. In addition, the following information as applicable should be provided so as to demonstrate that the RCS will retain its required level of structural integrity throughout the plant lifetime in both operational states and accident conditions.

(a) Integrity of the reactor coolant pressure boundary:

A description and justification should be provided on the results of the detailed analytical and numerical stress evaluations. Studies of engineering mechanics including fracture mechanics of all components comprising the reactor coolant pressure boundary subjected to normal conditions, shutdown conditions, and postulated accident loads should also be provided. A list of all components should be provided, together with the corresponding applicable codes. The 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:

Adequate information should be provided to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards to ensure the integrity of the reactor vessel. This includes the reactor vessel materials, the pressure–temperature limits and the integrity of the reactor vessel, including embrittlement considerations. Information should also be provided of provisions related to in-service inspection of the reactor vessel.

(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 etc.

5.1 Summary Description

This section should provide a summary description of the RCS and its various components. The 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, pressures, temperatures, flow rates, and coolant volume under normal steady-state full-power operating conditions should be provided. A process and instrumentation diagram of the RCS and connected systems and 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 (RCS) and Reactor Coolant Pressure Boundary (RCPB)

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 codes and standard used, overpressure protection of the reactor coolant pressure boundary, primary side of auxiliary or emergency system connected to the reactor coolant system, secondary side of steam generators and any blowdown or heat dissipation system connected to the discharge of these pressure relieving devices including all pressure-relieving devices like safety and relief valves.

Information should also be provided on reactor coolant pressure boundary materials, its compatibility with reactor coolant, coolant chemistry and fabrication method (including fracture toughness, control of welding, avoidance of stress corrosion cracking and non-destructive testing examination utilized) of pressure boundary components. In addition, the section should describe in-service inspection & testing program and provisions of reactor coolant pressure boundary components and provisions for coolant leakage detection. This should include aspects related to accessibility to perform inspections, methods, techniques and procedure used, inspection interval, provisions for evaluation results, repair procedures and system pressure test.

In case, replacement of any component is envisaged during the life time of the plant, provisions for replacement of such components should also be described.

Overpressure protection

This section should provide information, to assist the evaluation of the systems that protect the RCPB and the secondary side of steam generators from overpressure. These systems include all pressure-relieving devices (safety and relief valves) for the following systems:

a) RCS
b) Primary side of auxiliary or emergency systems connected to the RCS
c) Moderator system (over pressure rupture disc), if applicable
d) Any blowdown or heat dissipation systems connected to the discharge of these pressure-relieving devices
e) Secondary side of steam generators.

Reactor Coolant Pressure Boundary Leakage Detection

It should be demonstrated that the system is capable of separately monitoring and collecting leakage from both identifiable and unidentifiable sources. It should describe the floor drain system to demonstrate that leakage will flow to the sump or tank where it is collected and identify all potential intersystem leakage paths and the instrumentation used in each path. It should also demonstrate adequate monitoring capability to ensure that the limits of intersystem leakage assumed in the safety analyses are not exceeded.

Describe how signals from the various leakage detection systems are correlated to provide information to the plant operators on conditions of quantitative leakage flow rate. Discuss the provisions for testing and calibration of the leak detection systems.

5.3 Reactor Vessel

This section should contain relevant data in sufficient detail to provide assurance of the reactor vessel integrity under all plant states. The details should include material of reactor vessel, method used for fabrication and manufacturing, special controls during manufacturing, material acceptance criterion, material surveillance program and its adequacy and expected effect of radiation on reactor vessel.

In addition, the description on the materials and design of fasteners for the reactor vessel closure should be provided. Include enough detail regarding materials property requirements, nondestructive evaluation procedures, lubricants or surface treatments, and protection provisions to show that the regulatory requirements are fully met. It should also include the results of mechanical property and toughness tests to demonstrate that the material conforms to these recommendations or their equivalent.

Details should be provided on pressure – temperature limit and bases of setting operational limit on pressure – temperature for normal, off-normal & test conditions. The detail should include limit curve under different operating conditions. Provide limits on pressure and temperature for the following conditions:

a) Pre-service system hydrostatic tests,
b) In-service leak and hydrostatic tests,
c) Normal operation, including heat-up and cool-down, and

d) Reactor core operation.

Describe the procedure that will be used to update these limits during service, taking into account radiation effects. Describe the bases used to determine these limits, and provide typical curves with temperatures relative to the RT NDT (as defined in paragraph NB-2331 of Section III of the ASME Code or any other internationally acceptable codes like RCC-M, PNAE etc.) of the limiting material.

In addition, this discussion should include pressurized thermal shock (PTS) and provide detailed fracture toughness requirements for protection against pressurized thermal shock events, (PWRs only) throughout the life of the plant.

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 including fly wheel meet the safety requirements for design. Details should be provided on design bases, description, evaluations and tests & inspections. The information should also discuss the provisions taken to preclude rotor over speeding and reverse rotation of the RCP during normal operation and all accident conditions. Aspects related to shaft seizure should also be provided. Details of auxiliary systems of RCP such as motor cooling, pump seal and cooling, thermal barrier etc. should also be included.

5.5 Primary Heat Exchangers (Steam Generators)

A description and justification should be provided on the performance and design features that have been implemented to ensure that the steam generators meet the safety requirements for design. For fast reactors this also includes intermediate heat exchanger (IHX). 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 due to flow-induced vibration and cavitations should be specified, including:

a) Design conditions and transients that will be specified in the design of the steam generator tubes and the operating condition category selected (i.e., normal, AOO and accidental) that defines the allowable stress intensity limits to be used and the justification for this selection.

b) 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.

Details should also be provided of material, fabrication, compatibility with primary and secondary coolant, cleaning, provisions, codes & standards used and confirm that different components meets the specified requirements and in-service inspection as applicable. Details of monitoring provisions for primary to secondary leaks should also be described.

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 operational 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, tank over pressure relief provisions and the piping from the tank to the cover gas analyzer/ controlled venting systems, and the reactor coolant drain tank. The details should include design bases, system description including piping & instrumentation diagram, safety evaluation of the system to demonstrate design adequacy, instrumentation and control requirements and inspection and testing requirements.

A complete listing of all the safety valves used in primary coolant system should be provided with material of construction, maximum discharge capacity, type of valve (mechanical/ Instrumented/ power operated safety relief valves etc.) with the valve designed and set pressures. Surveillance requirements for these valves should also be specified.

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, coolant channel, piping, pumps, valves, and bolting covering the following areas:

a) accessibility,
b) examination categories and methods,
c) inspection intervals,

d) provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects; and
e) system pressure tests.

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

5.11 Reactor Auxiliary Systems

The list of reactor auxiliary system corresponding to respective reactor design should be provided. Description and justification should be provided of the performance and design features that have been implemented to ensure that the various subsystems associated with the RCS meet the safety requirements for design. The reactor auxiliary systems, where applicable, should include but not be limited to following systems:

a) Chemical and volume control system,
b) Reactor coolant make-up, bleed and purification system,
c) Residual heat removal system,
d) Shutdown cooling system,
e) Reactor core isolation cooling system,
f) Main steam line isolation system,
g) RCS high point vents,
h) Reactor water cleanup system
i) Moderator system and its auxiliary systems
j) End- shield cooling system
k) Calandria vault cooling system
l) Biological shield cooling system
m) Roof slab cooling system
n) Reactor Coolant Service System (Filling / Draining)
o) On power refueling system including fueling machine
p) Inventory Addition and Recovery System (IARS)
q) Reactor Coolant Sampling System
r) Annulus gas monitoring system (AGMS)
s) Cover gas system
t) Leak Monitoring Systems (LMS or LBB system),
u) Vibration Monitoring Systems (VMS); and
v) Hydraulic/Mechanical snubbers and monitoring systems.

The details description should follow the unified format presented in Section 2.3 of this guide and should include, summary description including manual operation requirements, system design, schematic of piping and instrumentation, equipment and component description, design criterion of components including piping, applicable codes & standards, system reliability considerations and testing requirements including preoperational testing.

3.6 Chapter 6. Engineered Safety Features

This chapter should present relevant information on the engineered safety features (ESFs) covering safety system, additional safety feature/system and complementary safety features. ESFs are provided to mitigate the consequences of postulated accidents (including severe accidents). The ESF systems provided in plant designs may vary as per the type of nuclear power plant. The ESF systems explicitly discussed in this chapter are those that are commonly used to limit the consequences of postulated accidents in nuclear power reactors, and should be treated as illustrative of the ESF systems and of the kind of informative material that is needed. Thus the information to be presented in this chapter on various ESF systems will inevitably depend on the particular type and design of reactor selected for construction.

The information presented under this chapter should include the design description, functional requirements of ESF systems and compliance with regulatory requirements

For each of the ESF systems, the more detailed description should include the following:

a) The system function and safety design bases,
b) The description of the system itself including relevant figures, sketches and schematic diagrams,
c) Material specifications, compatibility, including nuclear, radiological, chemical, physical and mechanical properties,
d) Interfaces with the other relevant systems,
e) Monitoring, inspection, testing and maintenance provisions,
f) Demonstration that all relevant functional requirements, requirements of industrial codes and standards, and regulatory requirements have been considered adequately,
g) Performance and safety evaluation including demonstration that the system has sufficient margin to fulfill its safety function.
h) Protections feature provided to ensure operability of ESFs including actuation and instrument systems, access for operator intervention if required in the event of DBAs/DECs.

6.1 Emergency Core Cooling System

This section should present relevant information on the emergency core cooling system (ECCS) and associated fluid systems following the structure specified in Section 2.3 of this guide. A summary description of the ECCS should be provided. Details of systems/sub-systems providing long term cooling should be described. All major subsystems of the ECCS such as active high-pressure and low-pressure safety injection systems and passive safety injection tanks should be identified. Nuclear plants that employ the same ECCS design and that are operating or have been licensed should be referenced. The purpose of the ECCS should be described and each accident or transient for which the required protection includes actuation of the ECCS should be listed. The actuation logic should be described subsequently in the section on protection systems in chapter 7 and need not be described here.

The design bases for selecting the functional requirements for each subsystem of the ECCS should be specified. Bases for selecting such system parameters as operating pressure, ECC flow delivery rate, ECC storage capacity, boron concentration, and hydraulic flow resistance of ECCS piping and valves should be discussed. Design bases concerned with reliability requirements should be specified. Protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation should be described. Protection against valve motor flooding and spurious single failures should be described.

Requirements established for the purpose of protecting the ECCS from physical damage should be specified. This discussion should include design bases for ECCS support structure design, for pipe whip protection, for missile protection, and for protection against such accident loads as loss-of-coolant accident or seismic loads.

Environmental design bases concerned with the high-temperature steam atmosphere and containment sump water level that might exist in the containment during ECCS operation should also be specified.

The relief valve capacity and settings or venting provisions included in the system should be stated. Specify design requirements for ECC delivery lag times. Describe provisions with respect to the control circuits for motor-operated isolation valves in the ECCS, including consideration of inadvertent actuation prior to or during an accident.

All manual actions required for the ECCS to operate properly should be identified. Discuss the information available to the operator, the time delay during which his failure to act properly will have no unsafe consequences, and the consequences if the action is not performed at all.

6.2 Containment Systems

6.2.1 Containment Functional Requirements

This section should discuss the following aspects:

a) Energy management: The sources and amounts of mass and energy that might be released into the containment and the post-accident time dependence of the mass and energy release should be discussed with reference to the design evaluations. The functions of the engineered safety features and their effectiveness as energy removal system in the containment should be discussed. The capability for energy removal from the containment under various postulated single-failure conditions of the engineered safety features should also be discussed.

b) Management of radionuclides: The bases for establishing the containment box up and subsequent depressurization rate should be discussed and justified with reference to the assumptions used in the analysis of the offsite radiological consequences of the accident. The subsequent containment re-pressurisation rate should also be discussed.

c) Management of combustible gases: Systems should be provided, as necessary, to control the concentration of hydrogen and oxygen that may be released into the containment following postulated accidents to ensure that containment integrity is maintained. The systems provided for combustible gas control include systems to mix the containment atmosphere, monitor combustible gas concentrations within containment regions, and reduce combustible gas concentrations within the containment. The functional capability of these systems should be discussed in this section.

d) Management of Severe Accidents/Design Extension Condition (DEC): Systems and features provided for management of severe accident and its operation under such conditions should be discussed.

AERB Safety Guide on 'Containment System Design for PHWRs' [AERB/NPP-PHWR/SG/D-21] [11] can be referred.

6.2.2 Primary Containment System and its Sub-compartments

The primary containment system and its sub-compartments should be described in this section following the unified format presented in Section 2.3 of this guide. Internal link should be provided to the description of the civil and structural design aspects in Section 3.7 (Chapter 3 of SAR).

The design bases for containment should be described considering the following:

a) The postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single failure of active component) that determine the design pressure requirements for the containment and containment internal structures (sub compartments) should be discussed. Credit should be taken only from seismically qualified equipment for accident mitigation in containment safety analysis. The maximum calculated accident pressure should be stated, and the bases for establishing the margin between this pressure and the design pressure should be discussed.

b) The postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single active failures) that determine the design pressure requirement for the internal structures of pressure-suppression-type containments (if applicable) with reference to the design evaluation should be discussed.

c) The bases for establishing the containment depressurization rate should be discussed and justified with reference to the assumptions used in the analysis of the offsite radiological consequences of the accident. The bases for containment re-pressurisation rate should be also discussed.

In addition, discuss the functional capability and frequency of operations of the systems provided to maintain the containment and sub-compartment atmospheres within prescribed pressure, temperature, and humidity limits during normal plant operation (e.g. containment cooling systems, containment clean up and ventilation systems, and containment purge systems).

6.2.3 Secondary Containment System

The secondary containment system includes the secondary containment structure and the safety-related systems provided to control the ventilation and cleanup of potentially contaminated volumes (exclusive of the primary containment) following a design basis accident. This section will discuss the secondary containment system following the unified structure presented in Section 2.3 of this guide. The ventilation systems (i.e., systems used to depressurize and clear the secondary containment atmosphere) should be described or referred to Section 6.2.5.

In addition, discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment filtration systems and escaping directly to the environment. Include a list of potential bypass leakage paths and provide an evaluation of potential bypass leakage paths considering realistic equipment design limitations. The following leakage barriers in paths that do not terminate within the secondary containment should be considered:

a) isolation valves in piping that penetrate both the primary and secondary containment barriers,
b) seals and gaskets on penetrations that pass through both the primary and secondary containment barriers, and
c) welded joints on penetrations (e.g., guard pipes) that pass through both the primary and secondary containment barriers.

6.2.4 Containment Energy Removal Systems

Systems to remove heat from the reactor containment should be provided to rapidly reduce the containment pressure and temperature following a loss-of-coolant accident or main steam line break to maintain them at acceptably low levels. The systems provided for containment heat removal may include fan cooler and spray systems or passive systems. The design and functional capability of these systems should be described in this section following the unified format presented in Section 2.3 of this guide.

6.2.5 Fission Product Removal and Control Systems

This section should include a detailed discussion of the operation of all fission product control systems following a design basis accident. Fission product control systems are considered to be those systems whose performance controls the release of fission products following a design basis accident. These systems are exclusive of the containment isolation system and include all filter system, containment spray and purge system. The ventilation systems (i.e., systems used to depressurize and clear the secondary containment atmosphere) should also be described in this section, if not described in section 6.2.3.

Both anticipated and conservative operation should be described. Reference should be made to other SAR sections, where appropriate. In addition, the following specific information should be presented to demonstrate the performance capability of these systems:

a) considerations of the coolant pH and chemical conditioning in all necessary conditions of system operation,
b) effects of postulated design basis loads due to fission products on filters; and

c) effects of design basis release mechanisms of fission products on filter operability.

6.2.6 Combustible Gas Control system

The design bases of the combustible gas control systems (i.e., the conditions under which combustible gas control may be necessary) and the functional and mechanical design requirements of the systems should be described. The design bases should include, as appropriate:

a) generation and accumulation of combustible gases within the containment,
b) capability to uniformly mix the containment atmosphere as long as accident conditions require and to prevent high concentrations of combustible gases from accumulating locally,
c) capability to monitor combustible gas concentrations within containment regions and to alert the operator in the main control room of the need to activate systems to reduce combustible gas concentrations,
d) capability to prevent combustible gas concentrations within the containment from exceeding the concentration limits,
e) capability to remain operable, assuming a single failure,
f) the sharing of combustible gas control equipment between nuclear units at multi-units sites,
g) capability to transport portable hydrogen re-combiner units after a loss-of-coolant accident,
h) protection of personnel from radiation in the vicinity of the operating hydrogen re-combiner units,
i) capability to purge the containment as a backup means for combustible gas control; and
j) methodology adopted to calculate the energy release rates due to combustion reaction.

6.2.7 Mechanical Features of the Containment

This section should describe in detail the following mechanical features of containment following the description in Section 2.3 of this guide:

a) Containment Isolation System.
b) Penetrations.
c) Airlocks, Doors, and Hatches.
d) Structural behavior monitoring systems.

The design requirements for the containment isolation barriers considering the following should be described:

a) extent to which the quality standards and seismic design classification of the containment isolation provisions follow the specified requirements,
b) provisions made to ensure that closure of the isolation valves will not be prevented by debris that is being carried by the escaping fluid,
c) operability of valves and valve operators in the containment atmosphere under normal plant operating conditions and postulated accident conditions,
d) qualification of closed systems inside and outside the containment as

isolation barriers,
e) qualification of a valve as an isolation barriers,
f) required isolation valves closure times,
g) demonstration of mechanical and electrical redundancy to preclude common mode failures; and
h) primary and secondary modes of valves actuation.

6.2.8 Containment Leakage Testing

The leakage testing requirements for the reactor containment, containment penetrations, and containment isolation barriers should be specified. This section should present a proposed testing program along with the test acceptance criteria. Describe the testing sequence for the containment structural integrity test and the containment leakage rate test. The maximum allowable containment integrated leakage rate during containment integrated leakage rate test (ILRT) should be specified. Also describe the measures that will be taken to ensure the stabilization of containment conditions (temperature, pressure, humidity) prior to containment leakage rate testing.

Discuss the pretest requirements, including the requirements for inspecting the containment, taking corrective action and retesting in the event that structural deterioration of the containment is found. Also discuss the criteria for positioning isolation valves and the requirements for venting or draining of fluid systems prior to containment testing. Fluid systems that will be vented or opened to the containment atmosphere during testing should be listed along with the identification of systems that will not be vented.

Further, aspects related to in-service inspection of containment liner (as applicable) should be provided.

6.2.9 Fracture Prevention of Containment Pressure Vessel

This section should present relevant information on the protection of the containment (primary and secondary) against over-pressure and under-pressure, if any.

6.3 Habitability/Survival Ventilation Systems

This section should present relevant information on the habitability/survival ventilation systems. The survival ventilation systems are the engineered safety features provided to ensure that essential plant personnel can remain in the main control room, supplementary control room and on site emergency support center, as applicable, and can take actions to operate the plant safely in operational states and to maintain it in a safe shutdown condition under accident conditions. The system for the control room should include shielding for radiation protection under accident conditions, air purification systems, protection against toxic gases, smoke, steam and control of climatic conditions. The storage capacity for food and water provided in the onsite emergency support centre should be described. Details of personnel protecting equipment such as fresh air masks, breathing apparatus sets etc., available at control rooms should be described in this section.

6.4 Other Engineered Safety Features

This section(s) should present relevant information on any other engineered safety features, additional safety systems/features and complementary safety features implemented in the plant design. Examples include, but are not limited to:

a) Passive decay heat removal systems.
b) Emergency borating system.
c) Containment filtered venting system.

The list of these systems will depend on the type of plant under consideration.

6.5 First of A Kind (FOAK) Systems

This section should include information on first-of-a-kind (FOAK) systems. For a new SSC (important to safety/safety related), design adequacy and safety is proven by a combination of research and development programs and relevant experience from similar applications. An adequate qualification program should be established to verify that the new design meets all applicable safety expectations.The discussions should include detailed technical inputs, details of experimentation and its results to evaluate the performance and reliability of the FOAK systems. A list of tests to be carried out in the different commissioning phases should also be included in this Chapter or referenced in Chapter 14 on Commissioning.

For each FOAK system, the detailed description should include:

a) description of the FOAK system,
b) design evolution (R&D),
c) applicable code and guides,
d) safety analysis,
e) demonstration of performanceand Mock-up test details,
f) interfacing with other safety systems and related issues,
g) design features including description of instrumentation/monitoring as appropriate and acceptance criteria,
h) list of commissioning tests,
i) surveillance programme; and
j) maintenance/In-service inspection requirement.

6.6 In-service Inspection for Class 2 and 3 Components

The applicant should discuss the ISI program for Class 2 and 3 components. The discussion should include the standards and rulesapplicablefor each of the areas listed below:

a) Accessibility.
b) Examination Techniques and Procedures.
c) Inspection Intervals.
d) Examination Categories and Requirements.
e) Evaluation of Examination Results.
f) System Pressure Tests.

3.7 Chapter 7. Instrumentation and Control

This chapter should provide relevant information on the instrumentation and control (I&C) systems. The information provided in this chapter should emphasize those instruments and their associated equipment that constitute the regulation/protection systems and those systems relied upon by operators to monitor plant conditions and to shutdown the plant and maintain it in a safe shutdown state after a design basis accident (DBA). These should also include radiation instrumentation and its monitoring and control function. 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 operation of the safety related instrumentation and control systems or create challenges which were not considered in the safety analysis of the plant. AERB Safety Guide on 'Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants' [AERB/NPP-PHWR/ SG/D-20] [12] can be referred for guidance.

7.1. General Principles and Design Approach

This section should describe I&C system architecture, 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 this section should describe:

a) safety classification and seismic categorization,
b) I&C system design basis including reliability; and
c) defence-in-depth, redundancy and diversity strategy.

Identify the systems that are identical to those of a NPP of similar design that has recently (last) received construction consent or an operating license; identify those that are different and discuss the differences and their effects on safety-related systems.

Safety criteria as well as applicable regulatory guides and standards should be identified for individual I&C systems in this section. The same can be presented in a tabular form.

In addition, provide logic diagrams, piping and instrumentation diagrams, and location layout drawings of all control systems in this chapter.

7.2. Reactor Protection System

Specific information on the following 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 accident analysis chapter (Chapter 15-Accident Analysis),
c) any interfaces with the actuation system for engineered safety features (including the use of shared signals and parameter measurement channels),
d) any interfaces with non-safety-related instrumentation, control or display

systems, together with 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,
h) bypasses, permissive and interlocks,
i) actuated devices,
j) power supply requirements,
k) information display requirement; and
l) testing requirement.

Considerations of instrumentation installed to prevent or mitigate the consequences of the following (as applicable) should be described:

a. Spurious control rod withdrawals
b. Loss of plant instrument power/air systems
c. Loss of cooling water to vital equipment
d. Plant load rejection
e. Turbine trip.

7.3. Engineered Safety Features Actuation System

This part of the SAR should provide relevant information on the engineered safety features (ESFs) actuation systems following the structure described in Section 2.3 of this guide.

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 ESFs.
b) Where the actuation logic for ESFs 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 ESFs, time delays in system operation and measurement uncertainties and how these relate to the assumptions made in the accident analyses chapter (Chapter15-Accident Analysis).
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 ESFs.

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.

In addition to postulated accidents and failures, it should include considerations of (1) loss of plant instrument power/air systems and (2) loss of cooling water to vital equipment. Also describe the method for periodic testing of I&C of ESF equipment and the effects on system integrity during testing. Reliability analysis should be presented to demonstrate that system design meets its reliability targets and single failure criteria.

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 (long term sub criticality), 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 provisions made in the design for the remote shutdown capability outside the main control room, such as remote shutdown stations / supplementary control room, in order to achieve and maintain hot and cold shutdown conditions should be described. The design of remote shutdown stations/supplementary control room should provide appropriate displays so that the operator can monitor the status of the shutdown. Logic diagrams, piping and instrumentation diagrams, and location layout drawings of all safe shutdown systems and supporting systems should be provided.

In addition to postulated accidents and failures, along with engineering evaluation, it is to be demonstrated that all these systems have sufficient capability to achieve and maintain safe shutdown condition. It should include, as applicable, considerations of instrumentation installed to permit a safe shutdown in the event of the following:

a) loss of plant instrument power/air systems,
b) loss of cooling water to vital equipment,
c) plant load rejection; and
d) turbine trip.

7.5. Information Systems Important to Safety

This section should provide relevant information on the systems for safety related display instrumentation and the computerized plant information system following the structure described in Section 2.3 of this guide. In addition, specific information on the following should also be provided:

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 mission time for which the reliable operation of the sensors is required.

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, 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. This should cover:

a) Any particular system needed for the management of severe accidents
b) Leak detection systems; monitoring systems for vibrations and loose parts
c) 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).

It should be demonstrated that the operator has sufficient information to perform required manual safety functions (e.g., ensuring safe control rod patterns, manual engineered safety feature operations, unanticipated post-accident operations, and monitoring the status of safety equipment) and sufficient time to make reasoned judgments and take action where operator action is essential. All instruments required for post-accident monitoring of reactivity, cooling and containment performance should be described.

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 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.

7.7. Control Systems not Important to Safety

This section should provide brief information on the control systems not important to 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); and
c) A demonstration that such systems do not challenge the operation of safety related systems.

It should be demonstrated that these systems are not important to safety and are capable of coping with all failure modes of the control systems.

7.8. Diverse Instrumentation and Control Systems

A description of the diverse I&C systems, as applicable, 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 diverse manual controls and display provisions should be addressed. Logic diagrams, piping and instrumentation diagrams, and location layout drawings of all diverse I&C systems should also be included.

The following should be demonstrated:

a) Requirements for reduction of risk from anticipated transients without scram (ATWS),
b) The adequacy of manual controls and displays that support operator actions to place the nuclear plant in a hot shutdown condition and to perform reactivity control, core heat removal, reactor coolant inventory control, containment isolation, and containment integrity actions.

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:

a) description on how the performance requirements of all supported systems are met,
b) the potential hazards to the system, including inadvertent actuations, error recovery, self-testing, and surveillance testing,
c) unauthorized access control,
d) redundancy and diversity requirements,
e) fail safe design of the protection systems,
f) system testing and surveillances,
g) status of the data communication systems in the design of bypass and inoperable status indications,
h) 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,and
i) exposure of the system to seismic hazards.

It should be demonstrated that the data communication systems conform to the relevant recommendations in codes, guides and standards applicable to these systems. The means and criteria for determining if a function has failed as a result of communications failure should also 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 should be described. The results of formal design review (human factors review) performed in developing or upgrading the layout of the control room, should be summarized in this section (Refer Chapter 18).

7.11. Supplementary/Backup 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 (Refer Chapter 18). The electrical design standards for equipment signals routed to the supplementary control room should be described or referenced. 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 independent of the MCR. The mechanisms for the transfer, of control and communications from the MCR to the supplementary/backup 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 Qualification

If digital instrumentation and controls systems are used, 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 should be included. The following topics should be addressed:

a) Design criteria to be applied to the proposed system.
b) I&C design as applicable to the individual sections of this chapter.
c) Defense-in-depth and diversity in a reactor trip system or an ESF actuation system as relevant for the combined ability of I&C systems to cope with common-cause failure.
d) Functional requirements.
e) Life-cycle process planning (the computer system development process, particularly the software life-cycle activities for digital systems).
f) Life-cycle process requirements, documenting the computer system functional requirements.
g) Software life-cycle process design outputs and identification of 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 above mentioned 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 detailed description should include information on the original design and test of the commercial equipment [13].

3.8 Chapter 8. Electric Power Supply Systems

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

8.1 General Principles and Design Approach

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

a) 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 center controlling the grid should be described, together with the provisions for communications between the dispatch center, the remote major load centers and the generating plants. The principal means of regulating the voltage and frequency of the external grid should be described. A simplified line diagram showing the main grid interconnections should be provided.
b) Derivation of auxiliary power supply at different voltage levels for station process loads, classification of plant specific electric power supply systems based on availability and reliability requirements [14].
c) Substantiation of the functional adequacy of the safety related electrical power systems and assurance that these systems have adequate redundancy, physical separation, independence and testability in conformance with the design criteria.
d) In addition, description and justification of the design bases, criteria, regulatory guides, standards, and other documents to be used in the design of electrical systems/equipment important to safety. In doing so, describe (and provide a positive statement regarding) the extent to which the design conforms to the appropriate standards and regulatory guides. Wherever alternative approaches are used, justify it by describing the methods used. Also equipment qualification, safety classification and seismic classes should be included.

In addition, preliminary system diagram/schematic, and location layout drawings should be provided in this chapter.

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 Section 2.3 of this guide. 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 specific design provisions necessary to cope with frequent grid failures should also be described.

Provide an analysis of the stability of the grid. This analysis should include the worst-case disturbances for which the grid has been analyzed and considered to remain stable. AERB Safety Guide 'Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactor' [AERB/SG/D-11; 2002] [14] can be referred for further details.

8.3 Onsite Power Systems

8.3.1 AC Power Systems

This subsection should provide relevant information on the plant specific AC power system as described in Section 2.3 of this guide. It should include a description of the on-site AC power systems, including the diesel or gas turbine driven systems, internal supporting 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 and sequentially. Electrical equipment protection, including the provisions to bypass this protection under accident conditions, should be described.

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 or gas turbine driven systems 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 under-frequency value for coastdown of the reactor coolant pumps are justified and the minimum number of engineered safety feature trains to be energized simultaneously is ensured, as applicable.
e) Equipment qualification such as environmental qualification, seismic qualification and independent validation & verification (IV & V) are addressed.

Provide analyses to demonstrate compliance with the codes and indicate the extent to which it has followed the recommendations of regulatory guides and other applicable criteria.

This section should include the following electrical power system analysis (provide the assumptions and conclusions that demonstrate the acceptance criteria) and distribution system studies:

a. Load Flow/voltage regulation studies and under-/overvoltage protection.
b. Short-circuit studies.
c. Equipment sizing studies.
d. Equipment protection and coordination studies.
e. Insulation coordination (surge and lightning protection).
f. Power quality limits.
g. Monitoring and testing.
h. Grounding.

AERB Safety Guide 'Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactor' [AERB/SG/D-11; 2002] [14] can be referred for further details.

8.3.2 DC Power Systems

This subsection should provide relevant information on the DC power system as described in Section 2.3 of this guide. 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.

Provide analyses to demonstrate compliance with the codes and indicate the extent to which it has followed the recommendations of regulatory guides and other applicable criteria.

This section should include the following electrical power system analysis (provide the assumptions and conclusions that demonstrate the acceptance criteria) and distribution system studies:

a) System redundancy requirements.
b) Conformance with the single-failure criterion.
c) System independence.
d) System capacity and capability.

AERB Safety Guide 'Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactor' [AERB/SG/D-11; 2002] [14] may be referred for further details.

8.4 Cabling and Raceways

In this section the cables and their raceways including cable supports, wall and floor penetrations, fire protection 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 earthquake loads. The buses and cables should also be sufficiently fire resistant/retardant to prevent the propagation of fires. At least three classes of cables should be identified: (1) control and instrumentation cables, (2) medium voltage power cables (e.g. 650 V or less), (3) high voltage power cables (e.g. 33kV or less). Special attention should be given to the qualification of cables that have to withstand conditions inside the containment during and after an enveloping event such as 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 also be 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.

8.6 Lighting System

This section should provide the description of the normal and emergency lighting system including the illumination levels at various locations under normal, offsite power supply failure and station black out conditions. The requirements of lighting fixtures in areas such as battery rooms and hydrogen cylinder areas should be incorporated.

8.7 Station Blackout

This section should describe how the design demonstrates compliance with applicable code/specific requirements and should indicate to the extent the requirements are followed. This section should specify the duration of time that a plant should be able to cope with an SBO and describe how redundancy and reliability of emergency onsite power sources are factored in determining an appropriate SBO duration which the plant should be capable of withstanding or coping with and recovering from. This section should also provide the target reliability levels chosen for emergency onsite ac power sources and a reliability program that provides reasonable assurance that reliability targets will be achieved and maintained.

8.8 Design Extension Condition Feature

This section should list and describe feature of the electrical power system provided to meet design extension conditions.

3.9 Chapter 9. Plant Auxiliary Systems

This chapter should provide relevant information on plant specific auxiliary systems that are essential for safe shutdown of the plant or for protection of the health and safety of the public. The design basis and description of each system, information related to system logics, instrumentation and control, system operation, system testing and inspection requirements should be included in SAR following the structure described in Section 2.3 of this guide. The description should also include safety and seismic classification of the system. 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. Radiological considerations associated with operation of each auxiliary system under normal operation and accident conditions, where applicable, should be summarized in the chapter and appropriate reference should be provided.

Even 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.

The plant auxiliary systems can be different for different 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. This chapter can be structured accordingly to the specificities of the design as the number of systems and their description may vary from reactor to reactor type.

A few examples of auxiliary system are fuel storage and handling (both new and spent fuel, as well as spent fuel cooling and cleanup), water, air and gas systems, ultimate heat sink, process auxiliaries, air conditioning, heating, cooling and ventilation systems and other auxiliaries (e.g. fire protection system, communication systems and emergency diesel generator cooling and auxiliaries).

Identification of shared systems for multiunit sites and analysis of failures of such systems should be provided.

9.1 Fuel Storage and Handling Systems

This section should provide relevant information on the fuel storage and handling systems. It should include details regarding quantity of fuel to be stored and the proposed arrangements, considering all failure scenarios, for the shielding, handling, inspection, storage, cooling, transfer and transport of nuclear fuel.

9.1.1 New Fuel Storage and Handling System

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, radiological protection, criticality prevention, fuel integrity and arrangements for transport etc.

9.1.2 Spent Fuel Storage and Handling System

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, including special provisions to deal with failed fuel, fuel chemistry, fuel cooling, fuel accounting systems, and arrangements for fuel consignment and transport.

9.1.3 Spent Fuel Pool Cooling and Cleanup/Purification System

The description should include the requirements for continuous or intermittent cooling, quantity of spent fuel to be cooled, requirements for pool water temperature and cleanliness from fission and corrosion products, leak monitoring, makeup requirements, level and radiation shielding requirements. A description of the cooling system and cleanup system, including schematic should be provided. The safety evaluation should include the capability for the spent fuel cooling during normal, AOOs, DBAs and DECs. Provisions to ensure that pool water will not be lost at a rate greater than the makeup capability and ability to maintain acceptable pool water conditions, radiological considerations (details should be presented in chapter12 on Radiation Protection) should be described.

9.1.4 Fuel Handling Systems

For fuel handling system (FHS), along with performance and load handling requirements, handling control features, provision to prevent fuel handling and cask drop accidents should be included. A description of the FHS system, including all components for transporting and handling fuel from the time it reaches the plant should be provided. An outline of the procedures used in new fuel receipt and storage, reactor refueling operations and spent fuel storage and shipment should be provided. The safety evaluation should demonstrate that the system design meets the applicable redundancy and diversity requirements. It should be demonstrated that the FHS design precludes inadvertent operations equipment malfunctions or failures that could prevent safe shutdown of the reactor or cause a release of radioactivity. Analysis should be presented to demonstrate that the individual subsystems and components, including control and interlocks, are designed to meet the single failure criterion without compromising with the capability of the system to perform its safety function.

9.2 Water Systems

This section should provide relevant information on the water systems associated with the plant structured in a format described in Section 2.3 of this guide. It should include but not limited to the following systems:

a) Service water system.

b) Component cooling water system/active and non-active water supply system.
c) De-mineralized water production and make-up system.
d) Ultimate heat sink system.
e) Condensate storage provisions.
f) Chilled water system.
g) Potable water systems.
h) Sanitary water systems.
i) Plant water system (Raw water/ Fresh water supply system).
j) Condenser cooling water System.
k) Turbine generator auxiliary cooling system.

For service water system, it should describe the capability of the service water system to meet the single failure criterion (when this system is safety related), the ability to withstand adverse environmental occurrences. It should also include the requirements for normal operation, during and subsequent to post accident conditions including loss of onsite and offsite power. The ability of the system to detect and prevent excessive leakage of radioactive material to the environment should also be described. Preventive measure for long term corrosion and organic fouling that may degrade the system performance and safety implications related to sharing (for multi-unit facilities) should be brought out.

9.3 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 Section 2.3 of this guide. It should include, for example, information on:

a) Process sampling system including post-accident sampling system
b) Equipment draining and floor drainage system
c) Chemical and volume control system (if not covered in other chapter)
d) Standby liquid control system (if not covered in other chapter).

For process and post-accident sampling system, describe the sampling system for various plant fluids. The design bases should include consideration of sample size and handling to ensure that a representative sample is obtained, provisions for isolation of the system and means to limit reactor coolant losses and requirements to minimize, to the extent practical, hazard to plant personnel. The points from which sample will be obtained should be delineated. The evaluation of the sample system should provide assurance that representative samples will be obtained and that sharing (for multiunit facilities) will not adversely affect plant safety.

For equipment draining and floor drainage system, describe the drainage systems for collecting the effluent from high activity and low activity liquid drains from various specific equipment items and buildings. Design considerations for precluding back flooding of safety related compartment should be discussed. Areas where drainage system is used to detect leakage from safety systems should be identified. Design consideration for preventing transfer of contamination fluids to non-contaminated drainage systems should be discussed. An evaluation of radiological considerations for normal operation and postulated spill and accidents including the effect of sharing (for multi-unit plants), should be presented in Chapter 11 on Radioactive Waste Management and Chapter 12on Radiation Protection.

For chemical and volume control system (CVCS) and boron recovery system (BRS), the design bases should include consideration of (1) the capability to vary coolant chemistry for control of reactivity and corrosion and (2) the capability for maintaining the required reactor coolant system inventory and reactor coolant pump seal water requirements. Items to be considered include the maximum and normal flow rates, charging rates for normal operation and maximum leakage conditions, boric acid storage and pH requirements for reactivity control, water chemistry requirements, and boric acid and primary water storage requirements in terms of maximum number of startup and shutdown cycles.

9.4 Air and Gas systems

This section should provide relevant information on air and gas systems following the structure described in Section 2.3 of this guide. The air systems that provide station air for service and maintenance uses should be described in this section, including compressed air systems (instrument, service and mask) and service gas (Nitrogen, Argon, Hydrogen, etc.) systems. A description should also be 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. Details of provision of back up air supply system for important equipment should be provided. The description should include maintenance of air cleanliness to ensure system reliability, the capability to isolate if required and safety implications related to sharing (for multi-unit plants). Supply, exhaust, sampling and safety provision for gas systems under various failure conditions should also be described.

9.5 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 Section 2.3 of this guide. In general, the description should include the requirements for meeting the single failure criteria, seismic design criteria, requirements for the manual or automatic actuation of system components or isolation dampers, fire dampers, ambient temperature limits, preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, monitoring normal and abnormal radiation levels within the area, differential pressure to be maintained and measured, and the requirements for the treatment of supply and exhaust air.

The following subsystems should be covered as applicable:

a) containment ventilation ,
b) control room HVAC,
c) spent fuel pool area HVAC,
d) auxiliary and radwaste area HVAC,
e) turbine building HVAC,
f) steam generator building HVAC; and
g) engineered safety features’ ventilation system.

9.6 Fire Safety

The primary objective of the fire safety programme is to minimize both the probability and consequences of postulated fires. The planning of fire prevention and protection programme should normally start at the plant design stage itself and carried through construction, commissioning and operation phases. Fire safety requirements which are to be addressed in the safety report are listed below. This section should provide relevant information on the fire protection systems following the structure described in Section 2.3 of this guide. It should justify the provisions made to ensure that the plant design provides adequate fire protection.

The SAR should state the basic requirements used for the design of the fire water supply and distribution systems and use of fire water system as back up to any of the cooling system. It should also specify any particular seismic requirements imposed on the design of each type of fire protection system used in the plant. 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. Chapter 15-Accident Analysis). Where appropriate, the provisions to ensure safety of personnel from fire or steam may also be described in this section.

9.6.1 Fire Protection Systems

The design bases for the fire protection system should be provided. The design should include adequate provisions for defense-in-depth in the event of fire, and should provide fire prevention, fire detection, 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 and /or diverse trains. Qualification criteria followed for passive fire protection systems should be elaborated. AERB Safety Standard 'Fire Protection Systems for Nuclear Facilities' [AERB/NF/SS/FPS (Rev.1); 2010] [15] and AERB Safety Guide 'Fire Protection in Pressurised Heavy Water Reactor based Nuclear Power Plants' [AERB/SG/D-4; 1999] [16] should be referred for details.

9.6.2 Fire Hazard Analysis

All areas containing safety systems and other locations, which constitute a significant fire hazard should be analyzed for postulated fire conditions to determine the fire resistance of the fire area boundaries and the requirement of the fire extinguishing systems and fire barriers. Layout drawings should be prepared for locations of fire extinguishers and fire barriers. Fire areas and fire zones should be incorporated in the building layouts and made part of the analysis report. Fire hazard areas should be identified where flame-proof/explosion-proof equipment are to be provided. Analysis should also demonstrate that the fire does not incapacitate the safety systems. Results of the analysis should be presented in this section.

Analysis should be revised periodically as design and construction progresses and before and during major plant modifications. Fire Hazard

Analysis report should be prepared as per the guidelines outlined in Appendix-A of AERB Safety Standard ‘Fire Protection Systems for Nuclear Facilities’ [AERB/NF/SS/FPS (Rev.1); 2010] [15].

9.7 Emergency Diesel Generator Supporting Systems

Supporting systems for the emergency diesel generator should be covered in this section. Main parts of emergency diesel generator system are covered in Chapter 8. The following subsystems should be addressed in this section:

a) Diesel generator fuel oil storage and transfer system.
b) Diesel generator external cooling system.
c) Diesel generator starting air system (provide internal link if covered in Chapter 8-Electric Power Supply Systems).
d) Diesel generator lubrication system (provide internal link if covered in Chapter 8-Electric Power Supply Systems).
e) Diesel generator combustion air intake and exhaust system (provide internal link if covered in Chapter 8-Electric Power Supply Systems).
f) Fire protection measures.

9.8 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.

9.9 Overhead Heavy-load Handling System

The overhead heavy-load handling system with respect to critical load handling should be described in this section. Inadvertent operations or equipment malfunctions during heavy load handling could have following safety concerns:

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

Necessary information includes parameters defining the load that, if dropped, would cause the maximum 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, but not limited to:

a) Reactor building crane.
b) Fuel building crane.

3.10 Chapter 10. Steam and Power Conversion System

This chapter 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 steam and power conversion system in normal operational states and under accident conditions.

b) A description of the main steam line piping and the associated relief and control valves, the main condensers, the main condenser evacuation system, the turbine gland sealing system, the turbine bypass system, atmospheric steam discharge system, the circulating water system, the condensate cleanup system, the condensate and feedwater system and the steam generator blowdown system should be provided. A description of the water chemistry programme, together with a discussion of the materials of the steam, feedwater and condenser systems should also be provided. Information on measurement system for stresses, displacement of important section of pipelines, effect of pipe whip or jet impingement, Steam leak monitoring system, LBB criteria as applicable should be presented.

Enough information should be provided to allow understanding in broad terms of what the secondary plant (steam and power conversion system) is, but emphasis should be on those aspects of design and operation that do or might affect the reactor and its safety features or contribute toward the control of radioactivity. The capability of the system to function without compromising directly or indirectly the safety of the plant under both normal operating and transient situations should be shown by the information provided. 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, the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems should be summarized in this chapter and presented in detail in Chapter 11 on Radioactive Waste Management and Chapter 12 on Radiation Protection

In describing individual parts of the steam and power conversion system, the information provided should follow the structure specified in Section 2.3 of this guide.

10.1 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 indicate safety-related system design features.

10.2 Main Steam Supply System

In this section the main steam supply system and main steam-line piping should be described along with a process flow diagram covering process and instruments, system components and interconnected piping. On the process flow diagram the physical division between the safety-related and non-safety-related 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 nuclear steam supply system (NSSS) to the power conversion system and various safety-related and non-safety-related auxiliaries.

For the BWR direct cycle plant, the main steam system extends from the outermost 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, PHWR and FBR plants, the main steam system extends from the secondary side connections of the steam generators up to and including the turbine stop valves and includes the isolation valves, safety and relief valves, connected piping of larger diameters, up to and including the first isolation valves that are either normally closed or capable of automatic closure during all modes of operation, and the steam-line to the auxiliary feed water pump turbine (if provided).

10.3 Feed Water Systems

Both main and auxiliary feed water systems should be described in this section, including the capability to supply adequate feed water 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 presented, including component failure and the effects of equipment malfunction on the RCS. 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 should also be included, as applicable.

An analysis should demonstrate the capability of the system to preclude hydraulic instabilities (characterized as water hammer) from occurring for all modes of operation. An analysis to demonstrate the system's capability to perform its safety function when subjected to a combination of environmental occurrences, environmental conditions, pipe break, and loss of power during normal and accident conditions should be performed. In addition, an analysis should be performed to demonstrate the system's capability to perform its safety function under various operating conditions.

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

a) 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 feed water piping.
b) Provide a summary of the criteria for routing or isometric drawings showing the routing of the feed water piping system from the steam generators to the restraint that is closest, on the upstream side, to the feed water isolation valve that is outside containment.
c) Describe the piping system analyses, including any forcing functions, or the result of test programmes performed to verify that uncovering of feed water 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), associated system and equipment (including moisture separator and reheater), use of extraction steam for feedwater heating, and control functions that could influence operation of the

RCS should be described. In addition, process flow diagram 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 all plant states. 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 safety classifications for TG system components, equipment, and piping.

10.5 Turbine and Condenser Systems

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

a) main condenser,
b) condenser air extraction system,
c) circulating water system,
d) condensate system,
e) condensate cleanup system,
f) turbine auxiliary systems (turbine gland sealing system, turbine bypass system); and
g) generator auxiliary systems.

10.6 Steam Generator Blowdown System

The steam generator blowdown system and its design basis should be described in this section. This should elaborate its ability to maintain optimum secondary-side water chemistry in recirculating steam generators 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 Protection against Postulated Piping Failures for Main Steam and Feed water Lines

This section should describe the scope of the either leak before break or break preclusion concept implementation in the main steam and feed water lines or postulation of pipe rupture locations and provisions made for protections against consequences of pipe rupture. 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).

The relevant description should include how leak before break concept has been implemented, as applicable.

3.11 Chapter 11. Radioactive Waste Management

This chapter should describe:

a) The capabilities of the plant to segregate, collect, minimize, reduction of volume, handle, process, condition, store, monitor and dispose of liquid, gaseous (if not described elsewhere), and solid wastes that may contain radioactive materials.
b) The instrumentation used to monitor the release of radioactive wastes, interlink should be provided if not covered in Chapter 7 on Instrumentation and control systems.
c) Compliance with relevant rules/codes.
d) Effluent discharge limits considered.

The information should cover all the normal and off-normal operational states of the reactor including decommissioning. The proposed radioactive waste treatment systems should have the capability to maintain releases of radioactive materials so that the prescribed limits for occupational exposure and public exposure are not exceeded and ALARA principle is followed.

For all kinds of waste management, handling, conditioning, storage and disposal should be described as follows:

Management of waste: Measures to manage or contain the waste produced at all stages of the lifetime of the plant should be described, including methodology to categorize and segregate waste, as necessary.

Handling of radioactive waste: Measures to safely handle 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 and its safe transport from the point of origin to the specified storage point. A consideration of the possible need to retrieve waste having significant amount of long lived radio nuclides at some time in the future, including 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, if any, at all stages of the lifetime of the plant should be described. The quantities, types, activity levels and volumes of radioactive waste and the need to categorize and segregate 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 shielding, 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 safely dispose off 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 either to co-located Near Surface Disposal Facility (NSDF) or another specified location for long term storage, if necessary. Details on safety assessment, engineered disposal modules, access control, ground water monitoring, material handling, etc. relevant to NSDF should be addressed.

The sections 11.2 to 11.6 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 all 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 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.

11.1 Source Terms

A description of the main sources of radioactive 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 including the accumulation rates and the quantities, conditions and forms of radioactive waste with different states of aggregation and activity level, for operation and accident conditions of the plant. Information should also be provided on the methods and technical means for processing and/or conditioning, storage and transport of radioactive waste. 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 generation 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.

The SAR should identify all sources of release of radioactive material that are not normally considered part of the radioactive waste management systems, e.g., the steam generator blowdown system. Identify planned operations, including anticipated operational occurrences that may result in escape of radioactive materials. Consider leakage rates and concentrations of radioactive materials for both expected and design conditions. The bases for all values used should be provided. Describe changes from previous designs that may affect the release of radioactive materials.

11.2 Liquid Waste Management Systems

This section should describe the capabilities of the plant to segregate, collect, process, monitor, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, AOOs and accident conditions. Design features incorporated to reduce maintenance, equipment downtime, liquid leakage, and releases to the building atmosphere or to facilitate cleaning or otherwise improve radioactive waste operations should be described. Estimated volume of waste, activity concentration, treatment process adopted, selection of materials, drainage system for equipment and floor drains, buffer storage for off-normal waste/unavailability of effluent discharge system, etc. should also be addressed. Design aspects on radioactive waste recycling and approach towards waste minimization should be brought-out.

The design provisions incorporated to control the release of radioactive materials due to overflows from all liquid tanks outside containment that could potentially contain radioactive materials should be described. Discuss the effectiveness of both the physical and the monitoring precautions taken, e.g., dyke, level gauge, alarm indications and automatic diversion of wastes from tanks exceeding a predetermined level. The potential for operator error or equipment malfunctions (single failures) to result in uncontrolled releases to the environment should be discussed. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environment.

The seismic design criteria and analytical procedures for structures housing the liquid radioactive waste components should be provided along with the quality group classification for the liquid radioactive waste components and piping. Failure analysis along with mitigation measures for liquid/effluent storage tanks should also be described.

11.3 Gaseous Waste Management Systems

This section should describe the capabilities of the plant to control, collect, process, handle, store, delay, purification& decay process if any, monitor and dispose of gaseous radioactive waste generated as the result of normal operation and AOOs. The gaseous radioactive waste system components and their design parameters, e.g., flow, temperature, pressure, and materials of construction, stand-by systems/ equipment should be listed. Provide an evaluation indicating the capabilities of the system to process surges in waste flows associated with anticipated operational occurrences including purging of containment, refueling and equipment downtime.

The seismic design criteria and analytical procedures for equipment support elements and structures housing the gaseous waste treatment system should be provided along with the quality group classification for the gaseous waste treatment components and piping.

Design features incorporated to reduce maintenance, equipment downtime, leakage, and gaseous releases of radioactive materials to the building atmosphere or to facilitate cleaning or otherwise improve radioactive waste operations should be described.

The design provisions incorporated to control the release of radioactive materials in gaseous effluents as the result of equipment malfunction or operator error should be described. Control logics for containment isolation, its redundancy, treatment processes, filters replacement criteria, provision for condensate collection, etc. should be described. Discuss the effectiveness of monitoring precautions taken, i.e., automatic termination of waste release from waste gas storage tanks when the release exceeds a predetermined level. The potential for an operator error or equipment malfunction (single failures) that may result in uncontrolled releases of radioactivity to the environment should be discussed. Systems should include monitoring provisions for radio-nuclides like FPNG, tritium, Iodine, Argon-41 and particulates.

This section should also address atmospheric dilution factor available at the site, proposed discharge limits on daily activity discharges and their concentrations during normal operation, derived annual dose to public at fence post boundary.

For systems where the potential for an explosion exists, any equipment that is not designed to withstand the pressure peak of the explosion should be identified and justification for the same should be provided. Process instrumentation (including gas analyzers) and design features provided to prevent explosions should be described along with provisions to ensure that seals will not be permanently lost following an explosion.

The section should include a description of each gaseous waste subsystem and the process flow diagrams indicating processing equipment, normal flow paths through the system, equipment capacities, and redundancy in equipment. For multi-unit stations, those subsystems that are shared should be indicated. All equipment and components that will normally be shared between subsystems should be identified. For each subsystem, tabulate or show on the flow diagrams the maximum and expected inputs in terms of flow and radioactivity content for normal operation, including anticipated operational occurrences. The bases for the values used should be provided. Indicate the composition of carrier and blanket gases, and describe the segregation of streams containing hydrogen, if appropriate.

11.4 Solid Waste Management Systems

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

The SAR should provide the design objectives and design criteria for the solid radioactive waste handling and treatment /conditioning systems in terms of the types of wastes, the maximum and expected volumes to be generated & handled, and the isotopic and curie content. Design provisions on volume reduction (like compaction, incineration, fusing, etc.) to minimize the disposal area should be indicated. Acceptance criteria for the waste immobilization method adopted should be defined. The seismic design criteria and analytical procedures for structures housing the solid radioactive waste system should be provided along with the quality group classification for the solid radioactive waste components and piping.

Solid waste subsystem to be used for processing dry filter media (ventilation filters); contaminated clothing, equipment, tools, and glassware; and miscellaneous radioactive wastes that are not amenable to solidification prior to packaging should be described. Tabulate the maximum and expected waste inputs in terms of type, sources of waste, volume, and isotopic and curie content. The bases for the values used should be provided. Describe the method of packaging and equipment to be used. The provisions to be used to control airborne radioactivity due to dust during compaction and baling operations should be described. Discuss the methods of handling and packaging large waste materials and equipment that has been activated during reactor operation (e.g. core components).

Containers to be used for packaging wastes (particularly for High Level Waste (HLW)) and their compliance with applicable regulations should be described. Provisions for sealing, decontaminating, and moving the containers to storage areas should be discussed along with the potential for radioactive spills due to dropping containers from cranes, monorails, etc. Provisions for collecting and processing decontamination liquids and spillage should be described. The provisions for waste storage prior to shipping, including the storage capacity and the expected onsite storage time should be described. Layout drawings of the packaging, storage, and shipping areas should be provided.

The maximum and expected annual volumes and the curie and isotopic content of wastes to be shipped offsite for each waste category should be indicated as applicable.

Following essential aspects should be addressed in detail:

a) Post closure safety assessment of Near Surface Disposal facility (NSDF).
b) Adequacy of area earmarked for NSDF (onsite or off-site).
c) Material handling facility at NSDF (including weather proof power feeding arrangement to crane).
d) Firefighting devices at NSDF.
e) Access road, area drainage, lighting.
f) Access control and all around fencing.
g) Bore hole for monitoring ground water at NSDF.
h) Types of disposal modules.
i) Dewatering arrangement for disposal modules.
j) Special disposal/storage modules for activated core components and core instruments along with its shielding criteria.
k) Design provision to address design basis flood level of NSDF.
l) Seismic criteria and water proofing methodology for disposal/ storage modules.
m) Permissible background radiation level at NSDF.
n) Characterization to identify and record the various radio nuclides content of waste packages, their total activity, specific activity, radiation field and record maintenance system. This record maintenance system should cover tagging of each waste package and its location in disposal module to provide necessary information for future use.

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 operation, AOOs and accident conditions.

Provide system descriptions for radiation detectors and samplers used to monitor and control releases of radioactive materials generated as the result of normal operations, including anticipated operational occurrences, and during postulated accidents. Those aspects of the design that relate to removing airborne radioactivity from equipment, cubicles, corridors, and operating areas normally occupied by operating personnel and into the effluent control systems should also be described.

For continuous process and effluent radiation monitors, provide the following information:

a) location of monitors, type of monitor, sensitivity, and measurement made (e.g., gross beta gamma or isotopic analysis),
b) types and locations of annunciators, alarms, and automatic controls and actions initiated by each,
c) provisions for emergency power supplies,
d) set points for alarms and controls and bases for values chosen; and
e) main out fall (MOF) sampling and monitoring system.

11.5.1 Environmental Impacts during 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 as applicable. Radiological impacts on surroundings should be considered that include:

a) radiation from the buildings and facilities in which radioactive materials are handled,
b) radioactive nuclides in gaseous discharges from controlled area devices; and
c) 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 in liquid discharges in surface, ground or marine water bodies should also be taken into account.

11.5.2 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 release of radioactive substance beyond the plant, should be addressed. Warning signals together with the activation levels settings, if any should be specified. Comparative review of the new tools and techniques if any, which have been deployed for monitoring of real time situations should be provided.

11.6 Other Aspects of Waste Management

SAR should include a detailed safety assessment of the waste disposal facility. In addition, the following aspects (as applicable) related to waste management should also be described as subsequent subsections:

a) Safety assessment of disposal facility.
b) Resin transfer and fixation system.
c) Laundry system.
d) Decontamination system.
e) Incineration system.
f) Disposal/storage of Core components & Instruments.
g) Waste characterization.

3.12 Chapter 12. Radiation Protection

This chapter should provide information on the policy, methods and provisions for radiation protection to restrict exposure of a personnel deployed on O&M activities during, normal operation, and anticipated operational occurrences (including refueling, purging, fuel handling and storage, radioactive material handling, processing, use, storage, transport and disposal, maintenance, routine operational surveillance, in-service inspection and calibration). The expected occupational radiation exposures to personnel during normal operation and AOOs should also be described.

This chapter 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 (radiation protection training, routine radiation surveys, radiological work permits etc.) have been made to reduce the amount of unnecessary radiation sources.

12.1 ALARA Considerations

This section should provide a description of the responsible organization’s policy, design 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.

How ALARA design guidance (both general and specific) is given to the individual designers should be described. How the design is directed toward reducing the need for maintenance of equipment and to reducing radiation levels and time spent where maintenance, In-service inspection, Surveillance and other operational activities are required. Describe any mechanisms that provide for design review by a competent professional in radiation protection such as Health Physicist. Indicate the extent to which design considerations given in regulatory guides [17] will be followed; if not, then describe the specific alternative approaches to be used.

This chapter should describe the methods to be used to develop the detailed operational plans and procedures for ensuring that occupational radiation exposures are as per ALARA principle. The impact of these operational plans and procedures on the design of the facility and how such planning has incorporated information from operating plant experience, other designs, etc. should also be described. The normal radiation fields in various radiation areas viz. normal full time occupancy area, accessible area and shutdown accessible area should be mentioned. The section should provide classification of the radiation areas based on their occupancy during normal operation and in AOOs.

12.2 Radiation Sources

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

12.2.1 Contained Sources

The contained sources of radiation that are the bases for the radiation protection design should be described in the manner needed as input to the shield design calculation. The sources contained in equipment of the radioactive waste management systems should be described. In this section, descriptions should be provided for other sources such as the reactor core, the spent fuel, during fuel handling and in storage pool, various auxiliary systems, the steam lines and turbine system (including reheaters, moisture separators, etc.). For the reactor core, the method used to determine radiation levels external to the biological shield at locations where occupancy is envisaged should be described. For other sources, the description should tabulate sources by isotopic composition or gamma ray energy groups, activity (curie content), and geometry, as well as provide the basis for the values. The source location in the plant should be specified so that all important sources of radioactivity can be located on plant layout drawings. Provide additional details (and any changes) of source description that are used to develop the final shield design.

12.2.2 Airborne Radioactive Material Sources

The sources of airborne radioactive material in equipment cubicles, corridors, and operating areas normally occupied by operating personnel should be described in the manner required for design of personnel protective measures and dose assessment. Those airborne radioactive sources that have to be considered for their contribution to the plant effluent releases through the radioactive waste management system or the plant ventilation system should be described in Chapter 11-Radioactive Waste

Management. Any other sources of airborne radioactivity in the areas mentioned above that are not covered in Chapter 11 should be included and described here. Sources resulting from reactor vessel head removal, relief valve venting, and movement of spent fuel should be included. The description should include a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. The models and parameters for calculating airborne radioactivity concentrations should be provided.

12.3 Radiation Protection Design Features

This section should provide a description of the design features of the equipment and the facility that provides information on radiation sources and ensures radiation protection. It should provide information on the shielding for each of the radiation sources identified and describe the personnel protection features. It should also describe the monitoring instrumentation for fixed area radiation and continuous airborne radioactivity, and the criteria for their selection and placement.

The principles of radiation protection applied in the design should be stated. This should typically include:

a) Material selection to minimize induced radioactivity and surface protection to avoid deposition of activated corrosion products, Material handling facilities including remote handling, Proper slopes in the floor to route the leakages, Equipment hatches to facilitate their removal either for replacement or overhaul as necessary, Underground pipelines carrying active fluids with inspection chambers at intervals to prevent sub-soil contamination, basic philosophy of ventilation air flow, etc.
b) That no person receives doses of radiation in excess of the authorized dose limits as a result of normal plant operation.
c) The occupational exposures in the course of normal operation are as per ALARA principle.
d) All practicable steps taken to prevent accidents with radiological consequences.
e) All practicable steps taken to minimize the radiological consequences of any accident.

12.3.1 Design Features

The description should include those features that reduce need for maintenance and other operations in radiation fields, reduce radiation sources where operations must be performed, allow quick entry and convenient access, provide remote operation capability, or reduce the time required for work in radiation fields and any other features that reduce radiation exposure of personnel. It should include descriptions of methods for reducing the production, distribution, and retention of activation products through design and construction methods, material selection, water chemistry, decontamination procedures, features to facilitate easy decontamination such as easily washable surfaces, proper slopes to floor to route the decant, hatches of adequate size to facilitate easy removal of equipment, remote handling tools/tackles and provisions on equipment to facilitate their easy handling, etc.

This chapter should also provide layout and arrangement of the facility showing the locations of all sources described in Section 12.2. The layout should provide the radiation zone classification, including zone boundaries for both normal operational and refueling outage conditions. Other chapters as appropriate should be referred. The layouts should show shield wall thicknesses, controlled access areas, personnel and equipment decontamination areas, contamination control areas, traffic patterns, location of the health physics facilities, location of airborne radioactivity and area radiation monitors, location of control panels for radioactive waste equipment and components, location of the onsite laboratory for analysis of chemical and radioactive samples, and location of the counting room. Specify the design basis radiation level in the counting room during normal operation and anticipated operational occurrences. Describe the facilities and equipment such as fume-hoods, glove boxes, filters, special handling equipment, and special shields that are related to the use of sealed and unsealed source and by-product material.

12.3.2 Shielding

Provide information on the shielding for each of the radiation sources identified in Chapter 11 and Section 12.2, including the requirements and shielding design details for penetrations, the material, method by which the shield parameters (cross sections, build-up factors, etc.) were determined, and the assumptions, codes, and techniques used in the calculations. Describe special protective features that use shielding, geometric arrangement (including equipment separation), or remote handling to ensure that occupational radiation exposures will be as per ALARA principle in normally occupied areas.

12.3.3 Ventilation

Provide information on ventilation arrangement, air changes and provision of filters.
The personnel protection features incorporated in the design of the ventilation system should be described. Describe here any ventilation system protective features not covered in Chapter 9 or Chapter 11. Include those aspects of the systems that relate to controlling the concentration of radioactivity in the areas mentioned above. The SAR should also provide the criteria established for the air-changes and changes due to filters and adsorbers in the air cleaning system.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The section should provide relevant details of the arrangements for the monitoring of all significant radiation sources, in all activities throughout the lifetime of the plant. This should include adequate monitoring provisions to cover operational states, design and design extension condition, and where appropriate, severe accidents. This should include information on the auxiliary and/or emergency power supply and the range, sensitivity, accuracy, precision, calibration methods and frequency, alarm set points, recording devices, and location of detectors and alarms for the monitoring instrumentation.

The details of fixed area radiation and continuous airborne radioactivity monitoring instrumentation and the criteria for selection and placement should be described. This should also include environmental discharge monitors provided in Stack Monitoring Room (SMR). The SMR instrumentation should be capable of indicating separately the type of activity being discharged.

12.4 Dose Assessment

In this section the objectives and criteria for design dose rates in various areas and an estimate of the annual person-Sv doses associated with major functions such as operation, normal maintenance, calibration, radwaste handling, refueling, and in-service inspection should be provided. The estimated annual dose from onsite radiation sources such as the auxiliary building, the reactor building, at the boundary of the restricted area, at the site boundary, and, for multi-unit plant, at various locations in a new unit constructions area should be provided. Annual dose from stored radioactive wastes and from radioactive effluents (direct radiation from the gaseous radioactive effluent plume) during normal operating conditions should also be provided.

Estimated annual dose to construction workers due to radiation from the existing operating plant(s), and the annual person-Sv doses associated should also be provided. The basis, models, input data and assumptions for the above values should be provided.

This section should also provide details of the arrangements proposed to allow the assessment of doses to persons during all activities involving ionizing radiation throughout the full lifecycle of the plant. The arrangements should include provisions to address operational states and accident conditions.

Specify the radiation dose targets that are included in the design specification. This section should include the estimation of public dose from the operation of the plant throughout its lifetime.

12.5 Operational Radiation Protection Programme

This section should describe the administrative and organizational arrangements, 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) radiation protection rules and supervision of work,
c) monitoring of individuals and the workplace,
d) work planning and work permits,

e) protective clothing and protective equipment,
f) facilities, shielding and equipment,
g) health surveillance,
h) application of the principle of optimization of protection,
i) source reduction,
j) training and certification,
k) arrangements for response to emergencies; and
l) provisions for documentation and logging.

This section should identify methods to make, store and retain records of radioactive releases that will routinely be made from the site. Further, this section should identify the measures that will be taken to make appropriate data available to the authorities and the public.

12.6 Health Physics Programme

In this section, the administrative and organizational arrangements of the health physics program, including the authority and responsibility of each position identified should be described.

The selection of portable and laboratory equipment and instrumentation for performing radiation and contamination surveys, for airborne radioactivity monitoring and sampling, for area radiation monitoring, monitoring and sampling of process systems and for personnel monitoring during normal operation, anticipated operational occurrences, and accident conditions should be described. The instrument storage, calibration, and maintenance facilities should be described. The location of the health physics facilities (including locker rooms, shower rooms, offices, and access control stations), laboratory facilities for radioactivity analyses, protective clothing, respiratory protective equipment, decontamination facilities (for equipment and personnel), and other contamination control equipment and areas that will be available should be identified and described.

3.13 Chapter 13. Conduct of Operation

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 Responsible Organization's (RO) 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 and in compliance with technical specifications (Chapter 16) to protect the plant personnel, the public and the environment.

13.1 Organizational Structure of Operating Organization/Plant Management

This section should provide a description on arrangements of the operating organization and specify the functions and responsibilities of the different components within it. Provide an organization chart showing the title of each position, the number of persons assigned, the number of operating shift crews, and all licensed the positions. For multi-unit stations, the organization chart (or additional charts) should clearly reflect planned changes and additions as new units are added to the station.

The organization and responsibilities of review bodies within responsible organization (e.g. safety committees and advisory panels) should also be described. The description should cover the organizational structure, (including interfaces with external agencies), 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, reviewing and quality assurance 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 in the relevant field, for each identified position.

13.2 Training and Qualification/Licensing

Information should be provided to describe the staff training programme with the overall objective of developing, retaining and up-grading the competence, including refresher training and retraining, and also the applicable documentation system. Training programmes and facilities, including simulator facilities, should reflect the status, characteristics and behaviour of the plant units, and should be briefly described. This section should also provide details on the categories of personnel to be trained, including the full range of positions of O&M personnel. This section should also 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.

It should be demonstrated that a systematic approach to training is 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.

Where the licensing regime includes provision for the licensing of operators [17], the section should describe the system for licensing and explain the provisions that will be put in place to comply with these licensing requirements. The training/qualification/licensing programme should cover the following:

a) Full range of plant conditions (normal, abnormal and emergency).
b) Specific operational activities (e.g., operations, maintenance, testing and surveillance).
c) Full range of plant functions and systems, including those that may be different from those in predecessor plants (e.g., passive systems and functions).
d) Full range of relevant HSIs (e.g. MCR, remote shutdown panel, local control stations, and technical support centre) including characteristics that may be different from those in predecessor plants (e.g., display space navigation, operation of “soft” controls).

13.3 Emergency Planning

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, coordinated and effective. The emergency preparedness arrangements can either be presented as part of the SAR; or they are prepared as a separate document that is referenced in the SAR. The emergency preparedness arrangements should cover the full range of accidents (in particular, DEC that would have effects on the environment and the off-site areas where preparations for the implementation of protective measures as 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 should be described. The intervals foreseen for regular exercises to maintain adequate emergency preparedness should be established and justified.

13.3.1 Emergency Management

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

A general description should be provided on the emergency arrangements, maintaining and its up-keeping 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 coordinated 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. Where the exclusion area is traversed by a road/water body, the arrangements made or that are to be made to control traffic need to be described in this section.

Projection of the population within the EPZ including a discussion of the sources of information and methodology that supports the population projection needs to be provided. It specifically should address whether the projected population creates a significant impediment to the development of emergency plans over the required period, including how it would affect the evacuation time estimate. If a significant impediment is created, then the measures that would mitigate or eliminate the significant impediment need to be described.

The agency responsible for emergency preparedness and planning needs to be identified for situations involving real or potential radiological hazards in the State where the facility is to be located.This section should also list the emergency response plans/manuals that would be prepared.

The references or copies of letters of agreement (or other certifications) from the agencies, who have the authority for emergency planning responsibilities in the central and State governments should be provided.

13.3.2 Emergency Response Facilities

The areas, equipment, and materials to which the control room operator could require access during an emergency should be identified. Those spaces requiring continuous or frequent operator occupancy should be listed. The selection of those spaces included in the control room should be based on need during postulated emergencies. This information should be summarized in this section. 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/communicating data on plant conditions to the off-site emergency facility; This should include description of Onsite Emergency Support Centre (OESC).
b) An off-site emergency facility in which response personnel will assess information gained from on-site measurements, provide advice, support and co-ordinate with all emergency response organizations in order to inform and, if necessary, protect the public;
c) Off-site monitoring arrangements for communicating data and information to AERB need to be established.

Description of emergency response facilities should include details of any equipment, communications and other arrangements necessary to support the specific facilities’ assigned functions. How the habitability of these facilities are ensured during accidents should also be described.

13.3.3 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 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, on-going response actions.

It should be demonstrated that the response of the necessary instrumentation or systems at the plant is adequate to ensure the performance of the required safety functions during accident.

13.3.4 Emergency plan considerations for multi-unit sites

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

a) Address and assess the extent to which the existing site's emergency plan is credited for the new unit(s), including simultaneous accidents in more than one unit, how the existing plan would be able to adequately accommodate an expansion due to one or more additional reactors and include any required modification of the existing emergency plan for staffing, training, emergency action levels etc.
b) 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.
c) Describe any required updates to existing emergency facilities and equipment, including the alert notification system.
d) Incorporate any required changes to the existing onsite and offsite emergency response arrangements and capabilities with state and local authorities or private organizations.
e) If applicable, address the exercise requirements for co-located licensees.
f) Describe how emergency plans, including security, is integrated and coordinated with emergency plans of adjacent sites and other industrial activity including activities related to hazardous chemicals.

13.3.5 Maintaining emergency preparedness

This section should provide relevant information on the programmes and procedures employed for maintaining the emergency preparedness for its readiness.

13.4 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:

13.4.1 Maintenance, Surveillance, Inspection and Testing

In this sub-section the SAR should provide a description and justification of the arrangements that the responsible organization intends to have in place to identify, control, plan, execute, audit and review and improve maintenance, surveillance, inspection and testing practices that influence reliability and affect nuclear safety. The surveillance programme should be intended 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 also 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 adequate to ensure the inclusion of all relevant aspects of the operating limits and conditions (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/manufacturer.

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/shutdown systems and containment systems, owing to their importance to safety and the severity of the possible consequences of failure.

All testing that can affect the safety functions of a NPP should be identified. 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.

13.4.2 Core Management and Fuel Handling

The SAR should demonstrate that the responsible 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 refueling, tests/calculations are performed to confirm that the core management meets the safety requirements as per 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/fuel bundles/fuel assemblies or control rods (in core or during handling), so as to minimize the build-up of fission products and activation products in the primary coolant or in gaseous effluents.

13.4.3 Management of Ageing

This section should describe all parts of the plant that can be affected by ageing along with the proposals made for addressing the issues. This includes, among others, the responsible 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 in analysing the feedback of operational experience with respect to ageing [18 & 19].

13.4.4 Control of Modifications

This section 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 into account of the safety significance of the proposed modifications to allow them to be graded and referred to AERB, 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 responsible 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 and the plan for carrying out such activity should be reflected in the SAR. It should also be demonstrated that the requirements for configuration management are met in the implementation of the plant modifications.

13.4.5 Feedback of Operational Experience

In this section, a programme for the feedback of operational experience and its implementation should be presented. The programme should provide measures to ensure that plant incidents and events are identified, recorded, notified and investigated internally. This should also ensure its use to promote enhanced plant performance and safety culture through the adoption of appropriate countermeasures to prevent recurrences, and has a provision to inform AERB, wherever necessary. The programme should include consideration of technical, organizational and human factor aspects. Arrangements made for reporting and analysing low level events and near misses should also 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 and other plants (as applicable), 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 back fitting and modernization of the plant, including organizational changes, if necessary.

13.4.6 Monitoring 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 responsible organization is being implemented effectively and that the lessons are being learned from its own experience and from that of others, to enhance safety performances. 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 performances are used to enable senior management to detect and respond in timely way to any shortcomings and deteriorating in safety.

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

13.4.7 Documents and Records

This section should describe the details on the provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and disposing 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.

13.4.8 Management of Outages

This section should provide a description on the relevant arrangements for planning and conducting periodic shutdowns of the reactor as required by the operating cycle and other factors. 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 for ensuring safety during specific circumstances of outage, such as multiple activities, multiple agencies 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.

13.5 Plant Procedures

This section should describe administrative and operating procedures that will be used to ensure that routine operation, 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. The following categories of the procedures as described below should be covered.

13.5.1 Administrative Procedures

This section should provide a description of the general administrative procedures used 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.

13.5.2 Operating Procedures

This section should provide a description of the plant operating procedures. The information presented should be sufficient to demonstrate that the operating procedures for normal operation 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 start-up, power production, shutting down, cooldown, shutdown, load changes, process monitoring and fuel handling. It should be demonstrated that the principles of human factors engineering have been considered in the development and validation of the procedures.

13.5.3 Emergency Operating Procedures

This 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. 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.

13.5.4 Accident Management Guidelines

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

13.6 Nuclear Security

This Section of SAR will be a confidential document and to be submitted as per AERB manual on security. The Applicant should submit this document to AERB for reviewing the adequacy of Nuclear Security. The chapter, along with additional requirement will be reviewed at different stages of consenting process.

3.14 Chapter 14. Commissioning

Purpose of commissioning is to demonstrate the design intent in totality including validation of operating procedures and training of operating personnel. In this Chapter, the applicant should demonstrate that the plant will be commissioned in safe manner and will be suitable for service prior to its entering the operational phase. This should include the following aspects.

14.1 Scope of Commissioning Programme

This should include normal performance as well as demonstration of all objectives of design safety under Anticipated Operational Occurrences and Design Basis Accident Conditions by appropriate commissioning procedures and tests. Special tests (e.g. tests for FOAK systems), if required, to demonstrate a specific design intent should also be specified.

14.2 Commissioning Programme

The commissioning programme should be described in this Section. The tests required to validate the plant's performance as per the design criteria prior to the start of operation of the plant should be described. For this purpose a well-planned, controlled and properly documented commissioning programme should be prepared and made ready for implementation. A clear link between the plant design with respect to safety and the commissioning programme should be brought out.

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.

The operating procedures should be validated to the extent practicable as part of the commissioning programme, with the participation of the designated operating personnel. Following specific information should also be included in the commissioning programme:

a) A summary description on the overall schedule with broad sequencing of various plant systems, relative to the expected fuel loading date, for developing and conducting the major phases of the test programme, and hold points for regulatory review/consent.
b) Description of the major phases of the initial test programme and discussion of the overall test objectives, specific objectives to be achieved for each major phase and prerequisites for each major phase.
c) Description of the administrative controls that will govern the conduct of each major phase of the test programmes,

d) Utilization of available information on construction of the plant and other reactor operating experiences in the development of the initial test programme.
e) A summary description of pre-operational and start-up testing planned for each unique and first-of-a-kind (FOAK) system along with test acceptance criteria, additional measurements and precautions with reference to Section 6.5 of Chapter 6.
f) A list of test that are identified as part of First-Plant-Only-Test (FPOT) with adequate technical justification.
g) Procedure followed for acceptance of construction completion certificate (CCC) and system transfer document (STD) for all major structures, systems and components should be described.

14.3 Organization and Staffing

This chapter should present the details of the commissioning organization, including the appropriate interfaces between design, construction and responsible organizations during the commissioning period. This should include any provisions for additional personnel (such as equipment manufacturers) and their interactions with the commissioning organization. It should also be shown that sufficient number of qualified operating personnel at all levels will be directly involved in the commissioning process.

Description of the applicant's organizational units and any other organizations/ personnel that will manage, supervise/execute and review any phase of the test programme should be included.

The general plans for the assignment of additional personnel to supplement the plant operating and technical staff during each major phase of the test programme should also be brought out.

14.4 Regulatory Documents

List of all regulatory guides applicable to initial test programme that is to be used or alternative methods along with justification for their use during commissioning should be included.

14.5 Conduct of Commissioning and Testing

The processes established to develop and approve test procedures, to control test performance, ensure quality during test 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 or unexpected commission results.

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 should be described.

A description of how, and to what extent the plant operating and identified emergency procedures will be tested during the initial test programme should also be described.

Design provisions to meet specific intent of operation, maintenance, testing etc. e.g. Surveillance testing, provision of ISI etc. should also be brought out and tested/demonstrated as practicable.

14.6 Review, Evaluation and Approval of the Results

Measures to be established for the review, evaluation, and approval of test results for each major phase of the programme should be included in this section. Wherever acceptance criteria is not available, results of accident analysis or simulator simulations used for arriving at an acceptance criteria for any event/test should also be included.

The applicant's requirements pertaining to the evaluation, disseminating and archiving of test procedures/reports and test data following completion of the test programme should be described.

14.7 Quality Assurance Programme for Commissioning

The quality assurance program for commissioning should be described and demonstrate its compliance with the AERB Safety Code 'Quality Assurance in Nuclear Power Plants' [AERB/NPP/SC/QA (Rev-1); 2009] [20] and other applicable regulatory documents. The objectives and acceptance criteria used in developing detailed test procedures in consultation with the designers should be described. If applicant plans for contracting the work of planning, developing, or conducting parts of the commissioning program, the method by which the applicant will retain responsibility for and maintain control of such contracted work should also be described.

All the commissioning activities should be audited as per approved procedure and related records should be generated and maintained for further follow-up action. Requirements related to these aspects should also be described.

14.8 Revision for Technical Specifications, EOPs and Revision of Accident Analysis

After completion of the commissioning activities, applicable documents such as technical specifications for NPP operation, Safety Analysis Report, EOPs and operating procedures should be revised, if required prior to application for regular operation of the NPP. Requirements related to these aspects should be included in this section.

3.15 Chapter 15. Accident 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/doses. These analyses include deterministic safety analyses (used in support of normal operation, analyses of anticipated operational occurrences (AOOs), design basis accidents, beyond design basis accidents/Design Extension Conditions) and probabilistic safety assessment (PSA). However PSA is separately discussed in Chapter 19. The description may be supported by reference material, where necessary.

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.

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 Introduction and Applicable Reference Documents

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 objectives and principles of nuclear safety, radiation safety and technical safety applicable to the particular plant design, as previously identified in the Chapter 1-General Design Aspects. This should also bring out the implementation of defence in depth principle in design of the plant.

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 to their frequencies but also different safety aspects (challenges to the barriers against releases of radioactivity) of the same events.

The acceptance criteria should be well justified and documented. 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 and Accident Scenarios

The methods used to identify PIEs and accident scenarios should be described. Initiating events that can occur owing to human error should also be considered in the identification of PIEs. It should be demonstrated that the identification of PIEs and accident scenarios has been performed in a systematic way and has led to the development of a comprehensive list of events.

PIEs and scenarios should be classified in accordance with their anticipated frequencies and grouped according to 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, along with the justification, in each of the various event groups credited in the safety analyses, which does not include events having similar time of occurrence, plant response and radiological release fractions.

c) To allow for differing acceptance criteria for the safety analyses to be applied to differing event classes.

Typically the list of PIEs to be addressed in the SAR will cover AOOs and design basis accidents. Some of the design basis accidents or DEC may further develop, if additional faults are assumed, and lead to severe accidents involving significant core degradation and/or off-site radioactive releases. Hence, a list of scenarios to address design extension conditions with and without core melt should also be presented. The results of the DEC analysis could be used 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 refueling, 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 offsite power or the total loss of off-site power, should also be evaluated, with account taken of the possible interactions between plant maneuvers 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 [21] and presented in the SAR should include, among others, internal PIEs such as:

a) increase or decrease of heat removal,
b) increase or decrease of reactor coolant flow,
c) reactivity and power anomalies (including mis-positioning of a fuel bundle),
d) increase or decrease of the reactor coolant inventory,
e) fresh or spent fuel storage related events,
f) release of radioactive material from a subsystem or component; and
g) malfunction of support/auxiliary systems.

An additional event category is ATWS (depending on the reactor type). ATWS events are events in which the reactor scram system (the reactor protection and reactivity control systems) is postulated to fail to operate following anticipated transients such as a loss of feed water, loss of load, turbine trip, inadvertent control rod withdrawal, loss of condenser vacuum in PWRs, and a closure of main steamline isolation valves in BWRs.

Even though ATWS events are beyond-DBEs and are subject to PSA, the physics and thermal-hydraulic phenomena of the plant response to ATWS events should be evaluated. Therefore, ATWS events are included as a category of postulated accident.

In addition, a set of internal PIEs, where appropriate, 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, derived from other considerations should be taken into account.

The set of external PIEs to be considered should include those due to, where appropriate: fires, floods, earthquakes, volcanic eruption, 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 in this chapter or Chapter19 on Probabilistic Safety Assessment.

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.

The environmental risks of accidents involving release of activity that can be postulated for the plant under review should be addressed. The scope of the section should also cover the offsite dose consequences for design basis accidents as well as for selected severe accidents.

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 DEC is generally less conservative than that of design basis accidents. It is acceptable that best estimate codes are used for deterministic analyses of DEC 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. Plant data used for development of the plant models should be provided 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 [21].

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.

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:

a) 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.
b) Plant parameters are maintained within the boundaries specified by the plant limits and conditions.

All possible conditions of normal operation should be analysed/addressed. 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) 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; and
i) handling of fresh and irradiated fuel.

15.5.3 Analysis of AOOs and Design Basis Accidents

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 fuel handling, 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 and operating mode of reactor 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 which are important to the outcome of the safety analysis should be identified. These would typically include:

a) reactor power and its distribution,
b) core temperature,
c) cladding oxidation and/or deformation ,
d) pressures in the primary and secondary system,
e) containment parameters,
f) temperatures and flows,
g) reactivity coefficients,
h) reactor kinetics parameters and the worth of reactivity devices,
i) core void,
j) water inventory in primary system; and
k) water inventory in SG/SG level.

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.

In some cases different analyses may be necessary (e.g. thermal hydraulic and structural analysis) for a single PIE in order to demonstrate that different acceptance criteria are met. In such cases, 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 sub-section 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) Boundary conditions and Assumptions :
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:

a. Normally operating plant systems and support systems.
b. Normally operating plant instrumentation and controls.
c. Plant and reactor protection systems including set points and time delay in actuation.
d. Engineered safety features and their actuation set points and time delay in actuation.
e. 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 additional single failure (if any) 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/Event evaluation:

A discussion of the modeled plant behaviour, highlighting the timing of the main events (initial event, any subsequent failures, times at which various safety groups/mitigating systems are actuated and time at which a safe long term stable state (or safe shutdown) 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 including core and system performance and physical barrier performance can be obtained within the context of the acceptance criterion being considered.

Core and System Performance:

In evaluating fuel cladding, core and system performance, parameters such as power, heat flux, pressure of the RCS, minimum DNBR/CPR, fluid inventories of the RCS and pressure difference across core, fuel temperatures and flow rates in the emergency core cooling system, secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory 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.

Physical Barriers Performance:

The status of the physical barriers and the fulfillment of the safety functions should also be discussed. The evaluation of the parameters that may affect the performance of the barriers that restrict or limit the transport of radioactive material from the fuel to the public should be discussed. Some of the key parameters like RCS pressure, steam line pressure, containment pressure, relief and/or safety valve flow rate, flow rate from the RCS to the containment system, if applicable should be provided to represent the physical barriers performance.

(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. While it is desirable to have radiological consequences of each PIE, it is acceptable to have radiological consequences of the governing PIE in each category of events. In such case justification for considering the PIE as governing should be included. The results presented should typically include or referenced are:

a) Data and assumptions used to estimate radioactive source from postulated accidents.
b) Data and assumptions used to estimate activity released.
c) Dispersion data (points of release, distances to applicable receptors, atmospheric dispersion factors (χ/Q) etc.).
d) Dose data (dose calculation method, dose conversion assumptions, peak concentrations in containment).

(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.

(h) Summary of analysis:

This should include a statement about meeting the acceptance criteria. 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.

15.5.4 Evaluation for Design Extension Conditions without Core Melt

Analysis should be performed and presented to demonstrate the capability of the design to mitigate DEC without resulting in a core melt. The choice of the events of this class to be analysed may be made on the basis of regulations, prevailing international practices, 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 single initiating event (unless they are taken into account in the design basis accident at the design stage), such as:

a) Extended station blackout
b) Anticipated transient without scram (ATWS)
c) Design basis events with degraded performance of the protection system or engineered safety features
d) 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. This should also cover events related to shutdown condition.

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

a) The objective of the analysis of DEC 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.

15.6 Evaluations of Severe Accident/Design Extension Conditions with Core Melt

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 modeled 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 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. 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.

15.6.1 Identification of Severe Accident Scenarios

The following should be addressed in this section:

a) Bases.
b) List of scenarios, including that are associated with shutdown condition.

15.6.2 Severe Accident Prevention

Provide a deterministic evaluation to show how the plant's severe accident preventive features would cope with the events.

15.6.3 Severe Accident Mitigation

The following should be addressed in this section:

a) Severe accident progression, both In and Ex-Vessel.
b) Severe accident mitigation features for external reactor vessel cooling, hydrogen generation and control, core debris coolability, high-pressure melt ejection(as applicable), fuel-coolant interactions, core concrete interaction, containment bypass (including steam generator tube rupture and intersystem LOCA), equipment survivability, and other severe accident mitigation features.

15.6.4 Containment Performance Capability

This section should provide an overview of the containment design and address the containment performance goals identified

15.6.5 Accident Management

Describe those actions taken (as per AMG) during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins, (3) maintain containment integrity as long as possible; and (4) minimize offsite releases.

This should also include provisions regarding decision support system and on-site emergency support center with adequate infrastructure and training of plant operating and technical staff.

15.6.6 Consideration of Potential Design Improvements

Any consideration for potential design improvements that are planned at this stage should be brought out in this section.

15.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with Spent Fuel Pool

This section should present the safety analysis performed for PIEs and accident scenarios associated with spent fuel pool. Specific operating modes related to fuel handling (e.g. emergency core unloading) should be also considered. It should be demonstrated that the relevant acceptance criteria regarding maintaining subcriticality, heat removal, structural integrity, shielding and confinement of radioactive gases released from irradiated fuel in the spent fuel pool are complied with.

The format and content of the analyses presented in this part of the SAR should be consistent with the presentation of the analyses for AOOs and design basis accidents, taking into account differences in systems involved, large thermal inertia of the spent fuel pool, more stringent acceptance criteria, and specific pathways for releases of radioactive substances.

In addition the Chapter 15 of SAR should also contain the following Appendices.

Appendix 15A
A summary (in tabular form) of the computer codes used, as well as the reactivity coefficients (e.g., moderator density, moderator temperature, and Doppler coefficients) and initial thermal power assumed in the analysis of each transient or accident should be provided.

Appendix 15B
The reactor trip functions (in tabular form), ESF functions, and other equipment available to mitigate each transient and accident should be provided.

Appendix 15C
A summary(in tabular form) of the trip setpoints and the total delay times of the reactor protection system and ESF actuation system assumed in the analyses of the transients and accidents should be provided. The table should also include the trip setpoint values specified in the Technical Specification.

Appendix 15D
All single failures (in tabular form) considered to determine the limiting single failure used in each transient or accident analyzed

Appendix 15E
The limiting single failure (in tabular form) selected for each transient and accident analyzed should be provided in a tabular form.

Appendix 15F
A list (in tabular form) of non-safety related system and equipment used to mitigate transients and accidents should be provided.

3.16 Chapter 16. Technical Specifications for Operation

The Operational Limits and Conditions (OLCs) and administrative requirements form an important part of the basis on which the applicant is authorized to operate the plant. The OLCs and administrative requirements can either be presented as part of the SAR; or they are prepared as a separate document that is referenced in the SAR. The applicant submits a proposed technical specification for operation along with bases for the review of AERB, and later modified and approved before becoming a part of operating license. A summary statement of the bases or reasons for all specifications, other than those dealing with design features and administrative controls should be included.

The submission of the document during SAR stage should be consistent with the format and content given in the AERB Safety Guide 'Operation Limits and Conditions' [AERB/SG/O-3; 1999] [22]. The following points given should be addressed in the document:

a) Technical specification for operation should clearly state the facility operating conditions (e.g. power operation, refueling) to which it applies.
b) Each specification should be as complete as possible, without any ambiguity and should include preliminary numerical values, graphs, tables, and other data.

c) The Technical Specification for Operation should be consistent with safety analysis of individual NPP and the design provisions. It should take into account the uncertainties in the process of safety analysis, including test and measurement limitations. The bases/ justification for each clause should be substantiated with proper reasons for its adoption and the background information, as appropriate.
d) The specifications should contain numerical values of limiting parameters and operability conditions of systems and components. The corresponding surveillance requirements to ensure that parameters remain within specified limits and that systems and components are operable should also be specified. The actions to be taken in case, the limits and conditions are not fulfilled should also be clearly established.
e) Essential administrative aspects, such as minimum shift composition etc.are also to be covered by these conditions. Reporting requirements of operational events and significant events should also be covered.
f) The selection of values for technical specifications should be done by (1) deterministic methods, or (2) probabilistic and reliability methods. Probabilistic and reliability methods should be utilized only when suitable justification is presented, and on a case-by-case basis.

16.1. Use and Application

This section should describe the way how the OLCs have been developed, their scope and range of applicability. 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. A list of definitions of terms, which are not defined in regulatory documents and specific to this Chapter, should be included in this Section.

16.2. Safety Limits

This Section should specify clearly the limits on important process parameters, within which operation of the reactor is shown to be safe by conservative analysis or assessment.

Safety limits are limits on process variable within which the operation of nuclear power plant is found to be safe. In case of violation of one or more of them, unacceptable degradation of integrity of items important to safety and/or unacceptable release of radioactivity are likely to occur. If safety limits are exceeded, the nuclear reactor should be immediately shutdown and maintained in cold depressurized state, the reactor should be restarted only after review and approval of AERB.

16.3. Limiting Safety System Settings (LSSS)

LSSS to provide sufficient margin below safety limit should be described. They are selected for the parameters included in safety limits and for other parameters or a combination of parameters which could contribute to pressure or temperature transients.

16.4. Limiting Conditions for Normal Operation, and Surveillance Requirements

Limiting condition for normal operation to provide acceptable margins between the normal operating parameters and the established limiting safety systems settings should be described. They should include the pre-requisite of a system configuration and operating personnel, i.e. minimum operable equipment, minimum staffing and prescribed action to be taken by operating staff.

Detailed OLCs for operation should contain numerical values of limiting parameters and operability conditions of systems and components.

The corresponding requirements for surveillance, maintenance and repair to ensure that these parameters remain within acceptable limits and that systems and components are operable should be specified and described in this section. Wherever appropriate, such requirements should be justified by means of a PSA. The actions to be taken in the event of non-fulfillment of the operational limits and conditions should also be clearly established.

16.5. Administrative Requirements

The relevant administrative requirements should be described in this section. Reporting requirements for operational events should also be covered. If the content under this section is prepared as a separate document the same should be referred.

16.6. 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 accounting 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.

16.7. Documentation

Appropriate documentation of operation is necessary to provide objective evidence of compliance with operational limits and conditions and to make sure that the information which may be necessary for evaluating or investigating any deviations from an operational limit or condition is available. The applicant should refer AERB Safety Guide 'Operational Limits and Conditions for Nuclear Power Plants' [AERB/SG/O-3; 1999] [22] for this purpose.

3.17 Chapter 17. Management of Safety and Quality Assurance

This chapter should describe and evaluate the responsible organization's management structure and introduces management of safety as an integral component of the management for maintaining safety. It should describe the procedures and processes that have been put in place to achieve satisfactory control of all aspects of safety throughout the lifetime of the plant.

In order to provide assurance that the design, construction, commissioning and operation of the proposed nuclear plant are in conformance with applicable regulatory requirements and with the design bases specified in the consent application, it is necessary that the applicant establish a Quality Assurance (QA) programme. The programme must meet the requirements of AERB Safety Code 'Quality Assurance in Nuclear Power Plants' [AERB/NPP/SC/QA (Rev.1); 2009][13]. In this, the applicant should provide a description of the QA programme to be established and executed during the design and construction, pre-operational testing and operation of the nuclear plant, in the SAR

17.1 Management of Safety

17.1.1 Specific Aspects of Management of Safety Processes

The measures employed to ensure the implementation and observance of the management of safety should be presented and justified. In particular following aspects of management of safety should be described:

a) Implementation of the management system, including safety culture, grading in the system and its documentation.
b) Responsibilities of senior management for the development and implementation of an effective management system.
c) Integration of elements of management systems, including those regarding safety, health, environment, security, quality, human-and-organizational-factor, societal and economic.
d) Resource management, including management of human resources, infrastructure and the working environment.
e) Measurement, assessment and improvement of the management system of nuclear installation.

17.1.2 Organisational Structure, Responsibility and Authority

The SAR should describe clearly the organisational structure, authorities and duties of senior management and organisations giving assurance that management system and QA programme is established and executed in a satisfactory. The SAR should provide organisation charts and lines of responsibilities in the NPP including those of the corporate organisation, the design group, the procurement group and the construction group. The SAR should also provide information on the arrangement made for resource management, including management of human sources, infrastructure and the working environment. The description should also present other functional organisations performing activities affecting quality in design, procurement, manufacturing, construction and installation, testing, inspection and auditing with clear delineation of their responsibility, authority and relationship to corporate management. In addition, a single overall project organisation chart should be included showing how the major organisations or companies working directly for the applicant on the project interrelate with one another.

17.1.3 Consideration of Safety Culture

This sub-section should present the responsible 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 site.

17.1.4 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 responsible organization is being implemented effectively and that the lessons are being learned from its own experience and from that of others, to enhance safety performances. 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 performances are used to enable senior management to detect and respond in timely way to any shortcomings and deteriorating in safety. This should also include a description of the way in which the responsible organizations intends to identify any development of the organization that could lead to the degradation of safety performances and should justify the appropriateness of the measures planned to prevent such a degradation.

17.1.5 Management of Organizational Changes

This section should elaborate the evaluation and classification of the organisational changes according to their importance to safety along with justification. The implementation of such changes should be planned, controlled, communicated, monitored, tracked and recorded to ensure that the safety is not compromised.

17.2 Quality Assurance Programme

17.2.1 QA Policies/Goals/Objectives

The corporate QA policies, goals, and objective should be summarized in this section of SAR. It should also describe how disputes involving quality are resolved.

17.2.2 Items Governed by QA Program

The SAR should identify the safety-related structures, systems, equipment and components to be controlled by the QA programme.

17.2.3 Planning

The SAR should provide a summary description of advanced planning that demonstrates control of quality-related activities including management and technical interfaces between construction organisation, the design organisation, manufacturer of items important to safety and the applicant nearing the end of design and construction phase and during pre-operational testing and, plant handover to plant management for operations and subsequent assurance of quality during operation.

17.2.4 Control of Documentation and Records

The SAR should describe the measures that assure that sufficient records are maintained both in soft form as well as hard form to furnish evidence of activities affecting quality. The SAR should also describe period for their storage, contingencies in case of data loss also needs to be described. The SAR should describe how the content of such records:

a) Includes at least the following: test logs; results of reviews, drawings, inspections, tests, audits, monitoring of work performance and material analyses, and such data as qualification of personnel, procedures and equipment.
b) Identifies the type of operation, the inspector or data recorder, the results, the acceptability and action taken in connection with any deficiencies noted.
c) Provides sufficient information to permit identification of the record with the item or activity to which it applies.

17.2.5 Training, Qualification/Certification

The SAR should describe the programme that provides adequate orientation and training of personnel performing activities affecting quality to assure that suitable proficiency is achieved and maintained. The SAR should describe how the orientation and training programme would assure that:

a) Personnel performing activities affecting quality are appropriately trained in the principles and techniques of the activity being performed.
b) Personnel performing activities affecting quality are instructed as to purpose, scope, and implementation of governing manuals, policies and procedures.
c) Appropriate training procedures are established.
d) Proficiency of personnel performing activities affecting quality is maintained.

The SAR should also describe the qualification requirements for position or positions responsible for assuring effective implementation of the QA programme of the applicant and of his major contractors.

17.2.6 Conformance to Regulatory Requirement

The QA programme in the SAR should cover each of the requirements brought out and describe the extent to which the QA programme will conform to provision of AERB Safety Code ‘Quality Assurance in Nuclear Power Plants’ [AERB/NPP/SC/QA (Rev.1); 2009] [20] in sufficient detail to permit a determination as to whether all requirements of the Code will be satisfied. The SAR should also demonstrate that the roles and responsibilities defined in AERB Safety Codes 'Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants'[AERB/NPP-PHWR/SC/D (Rev-1); 2009] [6], 'Design of Light Water Reactor based Nuclear Power Plants'[AERB/NPP-LWR/SC/D; 2015] [7] and 'Nuclear Power Plant Operation' [AERB/NPP/SC/O (Rev-1); 2008] [23] are covered appropriately in QA organization and QA programme.

17.2.7 Grading

This section should describe how a graded approach based on the relative importance to nuclear safety of each item, service or process would be used. The graded approach should reflect a planned and recognised difference in the applications of specific QA requirements in design, procurement and construction, commissioning and operation.

17.2.8 Procedures and Work Instructions

This section of the SAR should describe how the QA programme is documented by written policies, which would indicate how QA programmes for major activities such as design, procurement, construction etc. will be implemented.

17.2.9 Management Review

This section ofthe SAR should describe the measures that assure that there is regular management review of the QA programme to assess its effectiveness and the adequacy of its scope. The SAR should also provide the mechanism for measurement, assessment and improvement of the QA programme. The following should also be addressed in different sub-sections:

a) QA Program review.
b) Self assessment.
c) Independent assessment.
d) Improvement.

17.3 Quality Assurance in Design

17.3.1 Responsible Design Organisation

The RO should identify the design authority which will be responsible for specifying the system requirements and for approving the design outputs on its behalf and organisation(s) responsible for the detailed design.

17.3.2 Measures for Design Control

This section of SAR should describe the design control measures that assure that (1) applicable design requirements and design bases for safety related SSCs are correctly translated into specifications, drawings, procedures and instructions, (2) appropriate quality standards are specified in design documents and (3) deviations from such standards are controlled. The procedures that are established for generation, review, approval, issue and retrieval of design documents should also be described.

17.3.3 Verification & Validation

The SAR describe measures that assure verification or checking of design adequacy, such as design reviews, use of alternate calculation methods, or performance of a qualification testing programme using appropriate models and test conditions. SAR should identify the positions or organisation responsible for design verifications or checking and should describe measures that assure that verifying or checking process is performed by individuals or groups other than those who performed the original design.

17.3.4 Design Interfaces

This section of the SAR should describe measures for identifying and controlling design interfaces, both internal and external and for coordination between participating design organisations for review, approval, release, distribution, collections and storage of documents involving design interfaces and changes. SAR should describe how these measures will assure that the design documents are controlled in a timely manner to prevent inadvertent use of superseded design information.

17.3.5 Design Changes

The SAR should describe measures that will assure that design changes, including field changes are subject to the same design controls that were applied to the original design and are reviewed and approved by the organisations that performed the original design.

17.4 Quality Assurance in Procurement

17.4.1 Procurement Specifications

The SAR should describe measures that assure that procurement documents and changes thereto, for materials, equipment and services whether purchased by the applicant or contractors or subcontractors correctly include or refer the following as necessary to achieve quality:

(a) Applicable regulatory code and design requirements.
(b) Quality assurance programme requirements.
(c) Requirements, for supplier documents such as instructions, procedures, drawings, specifications, inspection and test records, and supplier QA records to be prepared, submitted or made available for purchaser review or approval.
(d) Requirements for retention, control and maintenance of supplier QA records.
(e) Provisions for purchasers, right of access to suppliers facilities and work documents for inspection and audit; and.
(f) Provision for supplier to report the non-conformances w.r.t. procurement requirements.

17.4.2 Design Documents Control

The SAR should describe (1) measures that clearly delineate the control responsibilities and action sequence to be taken in preparation, review, approval and issuance by competent personnel of procurement documents and (2) measures that assure that changes or revisions of procurement documents are subject to the same review and approval requirements as the original documents.

17.4.3 Vendor Evaluation

SAR should describe measures that assure (1) that procurement document require suppliers to have and implement a documented QA programme for purchased materials, equipment, and services to an extent consistent with their importance to safety, (2) that purchaser has evaluated the supplier before the award of the procurement order or contract to assure that the supplies can meet procurement requirements and (3) that procurement documents for spare or replacement items will be subject to control at least equivalent to those used for the original equipment.

17.4.4 Use of Commercial Grade Items

SAR should describe provisions that assure standard commercial off-the-shelf (COTS) items receive adequate attention, review and selection.

17.4.5 Conformance to Procurement Documents

This section should describe those measures that assure that materials, equipment, and services purchased by the applicant or contractors and subcontractors will conform to procurement document requirements.

17.5 Quality Assurance in Construction

This section of SAR should describe the measures established for the QA programme for all construction activities such as receiving and storing components, civil construction, erecting, installing, cleaning, flushing, inspecting, testing, modifying, repairing, maintaining and carrying out pre-operational testing.

Site construction activities should be planned and documented in adequate detail and approved by designated persons taking into account sequence of activities, the need for the use of special procedures for implementation, verification and acceptance, the need for special equipment for construction, inspection or testing and the need for calibration of instrument, training and qualification of personnel.

SAR should describe how the following are addressed in the applicant's QA programme.

a) selection of contractors,
b) review of contractors' QA programme,
c) approval of sub-contractors and suppliers,
d) document control,
e) control of design change information,

f) housekeeping during construction and installation,
g) activities requiring special cleanliness control,
h) industrial safety,
i) control of materials and equipment,
j) control of measuring and test equipment,
k) verification of construction work; and
l) hand over and transfer of responsibilities.

17.6 Quality Assurance in Special Processes

SAR should describe measures established to control special processes such as welding, heat treatment, non-destructive testing, etc. and to assure that they are accomplished by qualified personnel using written procedures qualified in accordance with applicable codes, standards, specifications, or other special requirements. SAR should describe those measures that assure that qualifications of special processes, personnel performing special processes and equipment are kept current and that records thereof are maintained.

17.7 Quality Assurance in Inspection

17.7.1 Establishment of Inspection Programme

SAR should describe the measures that assure that a programme for inspection is established and implemented by or for the organisation performing the activity to verify conformance with documented instructions, procedures and drawings for accomplishing the activity.

17.7.2 QA Plans Identifying Hold/Witness Points

SAR should describe the system whereby appropriate documents will identify any mandatory inspection hold points that require witnessing or inspection by the applicant or his designated representative and beyond which work may not proceed without the consent of his designated representative.

17.8 Test Control

17.8.1 Scope

SAR should describe a test programme that:

a) identifies all testing required to demonstrate that structures, systems and components will perform satisfactorily in service,
b) test is conducted by trained and appropriately qualified personnel in accordance with written test procedures that incorporate or refer the requirements and acceptance limits contained in applicable design documents; and
c) includes testing that will be performed under construction permit.

17.8.2 Provisions in Test Procedure

The SAR should describe the measures that assure that test procedures have provisions for assuring that:

a) all prerequisites for the given test have been met,
b) adequate test instrumentation equipment, qualified manpower are available; and
c) test is performed under suitable environmental conditions and with adequate test methods.

17.8.3 Control of Measuring and Test Equipment

SAR should describe the measures established for control of measuring and testing equipment.

17.9 Quality Assurance in Handling, Storage and Shipping

SAR should describe the measures established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. SAR should describe the measures for specifying and providing, when necessary for particular products, special protective environments such as inert gas atmosphere, moisture control and temperature control.

17.10 Control of Non-conforming Materials, Parts or Components

SAR should describe the measures established to control materials, parts or components that do not conform to requirements in order to prevent their inadvertent use or installation. The SAR should describe measures/methods that provide for, as appropriate, identification, documentation, segregation, disposition and notification to affected organisations. SAR should describe measures that control further processing, delivery or installation pending proper disposition of the deficiency. The SAR should describe measures established by the applicant:

a) for contractors to report to him those non-conformances concerning departures from procurement requirements that are disposed off "use as is" or "repair"; and
b) to make such non-conformance reports part of the documentation required at NPP site or to include a description of the non-conformance and its disposition on certificates of conformance that are provided to the site prior to installation or use of material or equipment at site.

SAR should state whether periodic analysis of non-conformance reports is performed to show quality trends and whether such analyses are forwarded to the management.

17.11 Preventive and Corrective Action

SAR should describe the measures that assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment and non-conformances are promptly identified and corrected.

SAR should describe how, in the case of significant conditions adverse to quality, the cause of the condition is determined, corrective action is taken to preclude repetition and the problem with its determined cause and corrective action and its acceptance is documented and reported to appropriate levels of management.

17.12 Audits

SAR should describe the programme of the applicant and of the principal contractors for conducting comprehensive, planned and periodic audits to verify compliance with all aspects of QA programme and to determine the effectiveness of the programme.

3.18 Chapter 18. Human Factor Engineering

This chapter should demonstrate that human factors engineering (HFE) and human–machine interface issues have been adequately taken into consideration in the development of the design, in order to facilitate interaction between the operating personnel and the plant. This should be valid for all plant states and for all plant locations where such interactions are anticipated. Information related to HFE and human-machine interface considered in the design for construction and commissioning should also be discussed.

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

18.1 Operating Experience Review

The section should describe the applicant's operating experience review (OER) methodology with respect to HFE-related safety issues.

18.2 Staffing and Qualifications

This section should describe the objectives of staffing and qualifications analyses, and the scope of the analyses performed as per AERB Safety Guide 'Staffing, Recruitment, Training, Qualification and Certification of Operating Personnel of Nuclear Power Plants' [AERB/SG/O-1; 1999] [24]. (Refer section 13.2)

The objective 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).

18.3 Human Reliability Analysis (HRA)

This section should describe the objectives and the 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.

18.4 Human-System Interface (HSI) Design

The objective of this section is to document the various Human Factors Engineering principles applied to the design of control room (main control room and supplementary control room), control boards and control consoles. Design basis for selection of alarms, controls and displays is required to be mentioned. Environmental factors namely illumination, temperature and noise should be described. SAR should also describe the process by which HSI design requirements are developed.

The design should provide adequate information to the operating staff to ensure safe state of NPP during normal operation as well as information required for handling transients and accidents.

The design should aim to promote the success of operator actions in light of the time available, the expected physical environment and the psychological pressure (operational stress) on the operator (please refer AERB/NPP-LWR/SC/D; 2015 [7] for details on the requirements with respect to operator actions).

18.5 Procedure Development

This section should demonstrate that the 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. SAR should describe the objectives and scope of the procedure development programme. This section should address the procedures of (1) plant system operations (which includes startup, power and shutdown operations), (2) test and maintenance, (3) generic technical guidelines for EOP's (4) abnormal and emergency operations, (5) accident management guidelines, (6) alarm response.

18.6 Training Programme Development

The section of SAR should describe (Refer section 13.2) the objectives and the overall scope of training programme.

18.7 Verification and Validation of HFE Results

This section of SAR should demonstrate that the verification and validation 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.

18.8 Design Implementation

The objectives and scope of the design implementation should be described. The scope should include the following considerations:

a) Confirmation that the as-built design conforms to the verified and validated design that resulted from the HFE design process.
b) Confirmation that the as built HSI procedures, and training conform to the approved design.

18.9 Human Performance Monitoring

The section should describe the objectives and scope of the human performance monitoring programme. The programme description should address how the programme provides reasonable assurance that the following criteria are met:

a) 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.
b) Changes made to the HSIs, procedure, and training do not have adverse effects on personnel performances (e.g., changes do not interfere with previously trained skills).
c) Human actions can be accomplished within established time and performance criteria.

3.19 Chapter 19. Probabilistic Safety Assessment

A Probabilistic Safety Assessment (PSA) of a nuclear power plant provides a comprehensive, structured approach for identifying failure scenarios and deriving numerical estimates of the risks to workers and members of the public. It is important to be ensured that the technological and methodological issues in PSAs are treated adequately and PSA has been carried out to an acceptable standard and that it can be used for its intended applications.

This chapter should provide the objective of the Level 1 PSA and Level 2 PSA, definitions of core damage/source term/large release, the methods and modelling assumptions used, component failure data and PSA results obtained and insights gained. These should include the relevant details for internal events and external events, and all operating modes. This chapter should provide the following information.

19.1. Level 1 PSA (Internal events, Full Power):

a) General Methodology

Methods and Modelling assumptions used in PSA, Details on the PSA standards, Component failure database, and Details about the computer codes used should be provided.

b) Description of plant systems

Brief description of the plant systems with a reference to design features and system operation should be provided so that this chapter of the SAR becomes stand alone.

c) **Initiating event analysis**

Approach for identification of IEs, Initial list, IE categorisation, Screening approach, Final list of PIEs, Grouping of IEs and Quantification of IE frequencies should be provided.

d) **Accident sequence analysis**

This section of SAR should describe the approach adopted in accident sequence modeling covering the aspects including, Definition of core damage, consequence categorisation, end state classification, frontline-support system dependency matrix and Event tree diagrams

e) **System analysis**

System functional description with simplified system schematic diagram, system success criteria (performance criteria such as flow, pressure, temperature, response time etc. and hardware requirements such as no. of trains, no. of components)

This should cover the following details for each system. System functional description with simplified system schematic diagram, system success criteria (performance criteria such as flow, pressure, temperature, response time etc. and hardware requirements such as no. of trains, no. of components), fault tree analysis, MCS analysis, Unavailability analysis.

f) **Human Reliability Analysis**

The description in this section should include the overview of human Reliability analysis (HRA), performance shaping factors (PSFs), References to the basis of available time and operator action time, man machine interface and quantification ofhuman error probabilities(HEP)

g) **PSA results**

The following aspects should be discussed as part of PSA results:

i) System unavailabilities.
ii) Contribution of PIEs to CDF.
iii) Contribution of Accident Sequences to CDF.
iv) Minimal Cut Sets (MCS) list for Accident sequences of different CDF categories (Top 100 cutsets).
v) Importance Analysis Results (Basic events as well as CCF events).
vi) Uncertainty and sensitivity analysis.

19.2. Level 1 PSA - External Events

This section should provide Hazard Analysis which should include both Internal and external hazards (sometimes referred to as external events, even when internal hazards are included). These hazards often create extreme environments common to several plants systems. This section should address the identification of internal and external hazards and the screening carried out to eliminate those which are unimportant contributors to the core damage frequency.

An Internal hazards include but not limited to internal fires, internal floods and missiles etc. Similarly, external hazards include but not limited to earthquakes, external floods, external fires, high winds, aircraft crash etc. The plant walk down for Hazard analyses should be described in this section.

The information submitted should contain the following as applicable to the hazard considered:

a) Identification of Hazard source
b) Qualitative screening
c) Plant boundary definition
d) Equipment damage criteria
e) Hazard analysis (Seismic, Flood etc.)
f) Fragility analysis (Seismic, Flood etc.)
g) Initiating Event frequency calculation
h) Plant damage state evaluation
i) Sequence and Systems analysis
j) Risk quantification
k) PSA Results
   i. Contribution to CDF
   ii. MCS List (Top 100 cutsets).

19.3. Level 1 PSA (Shut down and Low power):

The initiating events occurred during low power and shutdown modes usually make a significant contribution to the core damage frequency.

Relevant information as provided for Level 1 PSA (internal events, full power) should also be described in this section appropriately.

This section should provide the analysis carried out for shutdown and low power state and should include the following:

a) General methodology.
b) Plant description withdefinition of shutdown state.
c) Selection of IEs.
d) Consequence categorization.
e) Accident Sequence Analysis.
f) System Modeling.
g) Human Reliability Analysis.
h) PSA Results.

19.4. Level 2 PSA

This section should provide assessments of the probability of occurrence of severe core damage states, and the risk of large off-site releases requiring short term off-site response, especially those associated with early containment failure; and identification of systems for which design improvements or operational procedures could reduce the probability of severe accidents or mitigate their consequences.

a) General Methodology

Definitions of plant damage, source term, large release, methods and modelling assumptions used in PSA, details on the PSA standards, component failure database, and details about the computer codes used should be provided.

b) PSA Level 1 Modifications and Results

This should include modifications of level 1 PSA, grouping of IEs and refinement of PDS categorization.

c) Plant Damage State Analysis

Basis for binning of plant damage statesand quantification of PDS frequencies should be provided.

d) Accident Progression for Severe Accidents

Accident progression and Analysis of progression of severe accidents

e) Containment analysis

This should include identification of containment performance features, stages of accident progression and mitigation in the containment, Containment failure modes.

f) Containment Event Tree analysis

Containment Event tree development and containment event tree analysis should be described.

g) Containment system/ESFs Analysis

System functional description with simplified system schematic diagram, system success criteria (performance criteria such as flow, pressure, temperature, response time etc. and hardware requirements such as no. of trains, no. of components) should be discussed.

h) Source term assessment

This should include definition of source term, description of release categories, and source term assessment for different accident sequences.

i) PSA Results

The following aspects should be discussed as part of PSA results:

a) Severe accident sequence evaluation.
b) Release categorization.
c) Frequency of release categories Contribution of PIEs to LERF.
d) Contribution of Accident Sequences to LERF.
e) MCS list for Accident sequences of different LERF categories (Top 100 cutsets).
f) Importance Analysis Results (Basic events as well as CCF events).
g) Uncertainty and sensitivity analysis.

3.20 Chapter 20. Decommissioning

20.1 Introduction

Decommissioning of the plant will become necessary either at the end of the lifetime of the plant or earlier if the operator so decides. This chapter of SAR should contain the proposals anticipated at this point for the eventual decommissioning of the plant. The principles associated with the decommissioning should be presented followed by the explanation how these principles are implemented in the design.

In addition, this section should provide the information on the documentation required and regulations to be followed. AERB Safety Code ‘Management of Radioactive Waste’ [AERB/NRF/SC/RW; 2007] [25] and AERB Safety Guide ‘Decommissioning of Nuclear Power Plants and Research Reactors’ [AERB/NPP-RR/SG/RW-8; 2009] [26] can be referred.

20.2 Decommissioning Concept

This section should briefly discuss the envisaged 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 expected type, volume and activity of radioactive waste produced during the decommissioning phase,
c) envisaged options for decommissioning,
d) adequate documentary control and maintenance of records,
e) organizational provisions in place to preserve the institutional knowledge that will be needed at the decommissioning phase; and
f) envisaged site end point to be reached following decommissioning.

20.3 Decommissioning Plan

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

a) engineering studies for decommissioning, if necessary,
b) strategy for decommissioning, including the identification of a staged approach to decommissioning, if appropriate,
c) the development of a programme for bringing the plant to the desired condition for implementation of identified/envisaged approach to decommissioning,
d) ensuring confinement/containment wherever required for identified duration,
e) the development of a programme for ensuring that services (heating, electricity and water supply) will be available to support the work; and
f) the broad programme for providing adequate facilities for the sorting, processing, transport, storage/disposal, of the radioactive waste arising during decommissioning as per the envisaged decommissioning approach.

20.4 Provisions for Safety during Decommissioning

This section should provide a short description of the measures necessary to ensure safety during decommissioning. The measures 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 permit easier decontamination, to facilitate the access of personnel and machines and removal of waste, and to ensure the collection of important data should be described.

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6.2.2.14 Feedback and comparison with similar design
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6.2.3 Secondary Containment System

6.2.3.1 System / Equipment Functions
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6.2.3.5 Interfaces and interaction with other equipment or systems
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6.2.4 Containment Energy Removal Systems

6.2.4.1 System / Equipment Functions
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6.2.4.5 Interfaces and interaction with other equipment or systems
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6.2.5 Fission Product Removal and Control Systems

6.2.5.1 System / Equipment Functions
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6.2.6 Combustible Gas Control system

6.2.6.1 System / Equipment Functions
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6.2.6.15 References

6.2.7 Mechanical Features of the Containment

6.2.7.1 Containment Isolation System

6.2.7.1.1 System / Equipment Functions
6.2.7.1.2 Safety design bases
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6.2.7.1.5 Interfaces and interaction with other equipment or systems
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6.2.7.1.9 Radiological aspects
6.2.7.1.10 Performance and safety evaluation
6.2.7.1.11 Built-in safety features
6.2.7.1.12 System commissioning
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6.2.7.1.14 Feedback and comparison with similar design

6.2.7.1.15 References

6.2.7.2 Penetrations

6.2.7.2.1 System / Equipment Functions
6.2.7.2.2 Safety design bases
6.2.7.2.3 Description
6.2.7.2.4 Materials
6.2.7.2.5 Interfaces and interaction with other equipment or systems
6.2.7.2.6 System / Equipment Operation
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6.2.7.2.15 References

6.2.7.3 Airlocks, doors, and hatches

6.2.7.3.1 System / Equipment Functions
6.2.7.3.2 Safety design bases
6.2.7.3.3 Description
6.2.7.3.4 Materials
6.2.7.3.5 Interfaces and interaction with other equipment or systems
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6.2.7.3.10 Performance and safety evaluation
6.2.7.3.11 Built-in safety features
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6.2.8 Containment Leakage Testing

6.2.8.1 System / Equipment Functions
6.2.8.2 Safety design bases
6.2.8.3 Description
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6.2.8.5 Interfaces and interaction with other equipment or systems
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6.2.8.9 Radiological aspects
6.2.8.10 Performance and safety evaluation
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6.2.8.13 Compliance with applicable clauses of AERB design code
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6.2.9 Fracture Prevention of Containment Pressure Vessel

6.2.9.1 System / Equipment Functions
6.2.9.2 Safety design bases
6.2.9.3 Description
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6.2.9.11 Built-in safety features
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6.3 Habitability Systems/Survival ventilation systems

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6.3.2 Safety design bases
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6.3.10 Performance and safety evaluation
6.3.11 Built-in safety features
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6.4 Other Engineered Safety Features

6.4.1 System / Equipment Functions
6.4.2 Safety design bases
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6.4.4 Materials
6.4.5 Interfaces and interaction with other equipment or systems
6.4.6 System / Equipment Operation
6.4.7 Instrumentation and control
6.4.8 Monitoring, inspection, testing, and maintenance
6.4.9 Radiological aspects
6.4.10 Performance and safety evaluation
6.4.11 Built-in safety features

6.4.12 System commissioning
6.4.13 Compliance with applicable clauses of AERB design code
6.4.14 Feedback and comparison with similar design
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6.5 First Of A Kind (FOAK) Systems

6.5.1 Description of the FOAK system
6.5.2 Design features and acceptance criteria
6.5.3 Design Evolution (R&D)
6.5.4 Applicable code and guides
6.5.5 Safety Analysis
6.5.6 Mock-up test and demonstration of performance
6.5.7 Interfacing with other safety systems and related issues
6.5.8 List of commissioning tests with relative time schedule and hold points
6.5.9 Summary description of pre operational/startup testing
6.5.10 Training and documentation
6.5.11 Surveillance programme
6.5.12 Maintenance/ In-service inspection requirement
6.5.13 Operation

6.6 In-service Inspection for Class 2 and 3 Components

6.6.1 Accessibility
6.6.2 Examination techniques and procedures
6.6.3 Inspection intervals
6.6.4 Examination categories and requirements
6.6.5 Evaluation of examination results
6.6.6 System pressure test

7. Instrumentation and Controls

7.1 General Principles and Design Approach

7.1.1 I&C system architecture,
7.1.2 I&C functions and functional allocation to individual systems
7.1.3 Classification
7.1.4 I&C system design basis including reliability
7.1.5 Defence-in-Depth, redundancy and Diversity Strategy

7.2 Reactor Protection System

7.2.1 System / Equipment Functions
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7.2.4 Materials
7.2.5 Interfaces and interaction with other equipment or systems
7.2.6 System / Equipment Operation
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7.2.8 Monitoring, inspection, testing, and maintenance
7.2.9 Radiological aspects
7.2.10 Performance and safety evaluation
7.2.11 Built-in safety features
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7.2.13 Compliance with applicable clauses of AERB design code
7.2.14 Feedback and comparison with similar design
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7.3 Actuation Systems for Engineered Safety Features

7.3.1 System / Equipment Functions
7.3.2 Safety design bases
7.3.3 Description
7.3.4 Materials
7.3.5 Interfaces and interaction with other equipment or systems
7.3.6 System / Equipment Operation
7.3.7 Monitoring, inspection, testing, and maintenance
7.3.8 Radiological aspects
7.3.9 Performance and safety evaluation
7.3.10 Built-in safety features
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7.3.12 Compliance with applicable clauses of AERB design code
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7.3.14 References

7.4 Systems Required for Safe Shutdown

7.4.1 System / Equipment Functions
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7.4.4 Materials
7.4.5 Interfaces and interaction with other equipment or systems
7.4.6 System / Equipment Operation
7.4.7 Monitoring, inspection, testing, and maintenance
7.4.8 Radiological aspects
7.4.9 Performance and safety evaluation
7.4.10 Built-in safety features
7.4.11 System commissioning
7.4.12 Compliance with applicable clauses of AERB design code
7.4.13 Feedback and comparison with similar design
7.4.14 References

7.5 Information Systems Important to Safety

7.6 Interlock Systems Important to Safety

7.7 Control Systems not Required for Safety

7.8 Diverse Instrumentation and control system

7.9 Data Communication Systems

7.10 Main Control Room

7.11 Supplementary Control Room

7.12 Digital instrumentation and control system application

8. Electric Power

8.1 General Principles and Design Approach

8.2 Offsite Power Systems

8.2.1 System / Equipment Functions
8.2.2 Safety design bases
8.2.3 Description
8.2.4 Materials
8.2.5 Interfaces and interaction with other equipment or systems
8.2.6 System / Equipment Operation
8.2.7 Instrumentation and control
8.2.8 Monitoring, inspection, testing, and maintenance
8.2.9 Radiological aspects
8.2.10 Performance and safety evaluation
8.2.11 Built-in safety features
8.2.12 System commissioning
8.2.13 Compliance with applicable clauses of AERB design code
8.2.14 Feedback and comparison with similar design
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8.3 Onsite Power Systems

8.3.1 AC power systems

8.3.1.1 System / Equipment Functions
8.3.1.2 Safety design bases
8.3.1.3 Description
8.3.1.4 Materials
8.3.1.5 Interfaces and interaction with other equipment or systems
8.3.1.6 System / Equipment Operation
8.3.1.7 Instrumentation and control
8.3.1.8 Monitoring, inspection, testing, and maintenance
8.3.1.9 Radiological aspects
8.3.1.10 Performance and safety evaluation
8.3.1.11 Built-in safety features
8.3.1.12 System commissioning
8.3.1.13 Compliance with applicable clauses of AERB design code
8.3.1.14 Feedback and comparison with similar design
8.3.1.15 References

8.3.2 DC power systems
8.3.2.1 System / Equipment Functions
8.3.2.2 Safety design bases
8.3.2.3 Description
8.3.2.4 Materials
8.3.2.5 Interfaces and interaction with other equipment or systems
8.3.2.6 System / Equipment Operation
8.3.2.7 Instrumentation and control
8.3.2.8 Monitoring, inspection, testing, and maintenance
8.3.2.9 Radiological aspects
8.3.2.10 Performance and safety evaluation
8.3.2.11 Built-in safety features
8.3.2.12 System commissioning
8.3.2.13 Compliance with applicable clauses of AERB design code

8.3.2.14 Feedback and comparison with similar design
8.3.2.15 References

8.4 Cabling and Raceways

8.4.1 System / Equipment Functions
8.4.2 Safety design bases
8.4.3 Description
8.4.4 Materials
8.4.5 Interfaces and interaction with other equipment or systems
8.4.6 System / Equipment Operation
8.4.7 Instrumentation and control
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8.4.9 Radiological aspects
8.4.10 Performance and safety evaluation
8.4.11 Built-in safety features
8.4.12 System commissioning
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8.4.14 Feedback and comparison with similar design
8.4.15 References

8.5 Grounding and Lightning Protection

8.5.1 System / Equipment Functions
8.5.2 Safety design bases
8.5.3 Description
8.5.4 Materials
8.5.5 Interfaces and interaction with other equipment or systems
8.5.6 System / Equipment Operation
8.5.7 Instrumentation and control
8.5.8 Monitoring, inspection, testing, and maintenance
8.5.9 Radiological aspects
8.5.10 Performance and safety evaluation
8.5.11 Built-in safety features
8.5.12 System commissioning
8.5.13 Compliance with applicable clauses of AERB design code
8.5.14 Feedback and comparison with similar design
8.5.15 References

8.6 Lighting System
8.7 Station Blackout
8.8 Design Extension Condition Features

9. Plant Auxiliary Systems

9.1 Fuel Storage and Handling Systems

9.1.1 New Fuel Storage and Handling Systems

9.1.1.1 System / Equipment Functions
9.1.1.2 Safety design bases
9.1.1.3 Description
9.1.1.4 Materials
9.1.1.5 Interfaces and interaction with other equipment or systems
9.1.1.6 System / Equipment Operation

9.1.4.11 Built-in safety features
9.1.4.12 System commissioning
9.1.4.13 Compliance with applicable clauses of AERB design code
9.1.4.14 Feedback and comparison with similar design
9.1.4.15 References

9.2 Water Systems

9.2.1 Service Water System
9.2.2 Component Cooling Water System
9.2.3 De-Mineralized Water Production and Makeup System
9.2.4 Ultimate Heat Sink
9.2.5 Condensate Storage Provisions
9.2.6 Chilled Water System
9.2.7 Potable Water Systems
9.2.8 Sanitary Water Systems
9.2.9 Plant Water System (Raw water/ Fresh water supply system)
9.2.10 Condenser Cooling water System
9.2.11 Turbine Generator Auxiliary Cooling System
9.2.12 Active Process Water

9.3 Process Auxiliary Systems

9.3.1 Process and Post Accident Sampling System
9.3.2 Equipment draining and Floor Drainage System (non-active and active drainage system, leakage collection system)
9.3.3 Chemical and Volume Control System (if not covered in other chapter)
9.3.4 Standby Liquid Control System (if not covered in other chapter)

9.4 Air and Gas Systems

9.4.1 Compressed Air System
9.4.2 Service Gas System

9.5 Heating, Ventilation and Air Conditioning (HVAC) Systems

9.5.1 Containment Ventilation
9.5.2 Control Room HVAC
9.5.3 Spent Fuel Pool HVAC
9.5.4 Auxiliary and Radwaste Area HVAC

9.6 Fire Safety

9.6.1 Fire Protection System
9.6.2 Fire Hazard Analysis

9.7 Emergency Diesel Generator Supporting Systems

9.7.1 Diesel generator fuel oil storage and transfer system
9.7.2 Diesel generator cooling water system
9.7.3 Diesel generator starting air system
9.7.4 Diesel generator lubrication system
9.7.5 Diesel generator combustion air intake and exhaust system

9.8 Miscellaneous Auxiliary Systems

9.9 Overhead Heavy Load Handling System

9.9.1 Reactor Building Crane

9.9.2 Fuel Building Crane

10. Steam and Power Conversion System

10.1 Role and General Description

10.2 Main Steam Supply System

10.2.1 System / Equipment Functions
10.2.2 Safety design bases
10.2.3 Description
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10.2.5 Interfaces and interaction 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
10.2.11 Built-in safety features
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10.2.14 Feedback and comparison with similar design
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10.3 Feed water systems

10.3.1 Main feed water 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 and interaction 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
10.3.1.11 Built-in safety features
10.3.1.12 System commissioning
10.3.1.13 Compliance with applicable clauses of AERB design code
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10.3.1.15 References

10.3.2 Auxiliary feed water 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 and interaction 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
10.3.2.11 Built-in safety features

10.3.2.12 System commissioning
10.3.2.13 Compliance with applicable clauses of AERB design code
10.3.2.14 Feedback and comparison with similar design
10.3.2.15 References

10.4 Turbine Generator

10.4.1 System / Equipment Functions
10.4.2 Safety design bases
10.4.3 Description
10.4.4 Materials
10.4.5 Interfaces and interaction with other equipment or systems
10.4.6 System / Equipment Operation
10.4.7 Instrumentation and control
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10.4.9 Radiological aspects
10.4.10 Performance and safety evaluation
10.4.11 Built-in safety features
10.4.12 System commissioning
10.4.13 Compliance with applicable clauses of AERB design code
10.4.14 Feedback and comparison with similar design
10.4.15 References

10.5 Turbine and Condenser systems

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 and interaction 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
10.5.1.11 Built-in safety features
10.5.1.12 System commissioning
10.5.1.13 Compliance with applicable clauses of AERB design code
10.5.1.14 Feedback and comparison with similar design
10.5.1.15 References

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 and interaction 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
10.5.2.11 Built-in safety features
10.5.2.12 System commissioning

10.5.2.13 Compliance with applicable clauses of AERB design code
10.5.2.14 Feedback and comparison with similar design
10.5.2.15 References

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 and interaction 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
10.5.3.11 Built-in safety features
10.5.3.12 System commissioning
10.5.3.13 Compliance with applicable clauses of AERB design code
10.5.3.14 Feedback and comparison with similar design
10.5.3.15 References

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 and interaction 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
10.5.4.11 Built-in safety features
10.5.4.12 System commissioning
10.5.4.13 Compliance with applicable clauses of AERB design code
10.5.4.14 Feedback and comparison with similar design
10.5.4.15 References

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 and interaction 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
10.5.5.11 Built-in safety features
10.5.5.12 System commissioning
10.5.5.13 Compliance with applicable clauses of AERB design code
10.5.5.14 Feedback and comparison with similar design
10.5.5.15 References

10.5.6 Turbine auxiliary systems

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 and interaction 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
10.5.6.1.11 Built-in safety features
10.5.6.1.12 System commissioning
10.5.6.1.13 Compliance with applicable clauses of AERB design code
10.5.6.1.14 Feedback and comparison with similar design
10.5.6.1.15 References

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 and interaction 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
10.5.6.2.11 Built-in safety features
10.5.6.2.12 System commissioning
10.5.6.2.13 Compliance with applicable clauses of AERB design code
10.5.6.2.14 Feedback and comparison with similar design
10.5.6.2.15 References

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 and interaction 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
10.5.7.11 Built-in safety features
10.5.7.12 System commissioning
10.5.7.13 Compliance with applicable clauses of AERB design code
10.5.7.14 Feedback and comparison with similar design

11.3.7 Radiological aspects
11.3.8 Built-in safety features
11.3.9 Performance and safety evaluation
11.3.10 Discharge limits and doses
11.3.11 Safety classification and seismic categorization
11.3.12 PSI / ISI requirements
11.3.13 Code compliance

11.4 Solid Waste Management System

11.4.1 System description
11.4.2 Design parameter and bases
11.4.3 Solid Waste Categorisation
11.4.4 Estimated Volumes of Solid Waste
11.4.5 Waste Segregation
11.4.6 Waste Conditioning and process details
11.4.7 Waste interim storage
11.4.8 Solid waste disposal
11.4.9 Solid waste transportation
11.4.10 Near Surface Disposal Facility (NSDF)
11.4.11 Equipment selection & description
11.4.12 Logics, instruments and controls
11.4.13 Built-in-Safety features
11.4.14 Storage of activated core components & Instruments
11.4.15 Safety assessment and dose limits
11.4.16 Waste assaying
11.4.17 Code compliance

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 Environmental Impact during Plant Operation
11.5.2 System / Equipment Functions
11.5.3 Safety design bases
11.5.4 System description
11.5.5 Materials
11.5.6 Interfaces and interaction with other equipment or systems
11.5.7 System / Equipment Operation
11.5.8 Instrumentation and control
11.5.9 Monitoring, inspection, testing, and maintenance
11.5.10 Radiological aspects
11.5.11 Performance and safety evaluation
11.5.12 Monitoring/ sampling frequency
11.5.13 Record keeping
11.5.14 Code compliance

11.6 Other Aspects of Waste Management

11.6.1 Safety assessment of disposal facility
11.6.2 Resin transfer and fixation system
Sub divisions as mentioned above in 11.2 (as applicable)

11.6.3 Laundry system
Sub divisions as mentioned above in 11.2 (as applicable)

11.6.4 Decontamination system

Sub divisions as mentioned above in 11.2 (as applicable)

11.6.5 Incineration system
Sub divisions as mentioned above in 11.2 (as applicable)

11.6.6 Disposal/ storage of Core components & Instruments
Sub divisions as mentioned above in 11.2 (as applicable)

11.6.7 Waste characterization

12. Radiation Protection

12.1 ALARA Considerations

12.1.1 Policy consideration
12.1.2 Design consideration
12.1.3 Operational consideration

12.2 Radiation Sources

12.2.1 Contained sources
12.2.2 Airborne radioactive material sources

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features
12.3.2 Shielding
12.3.3 Ventilation
12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.4 Dose Assessment
12.5 Operational Radiation Protection programme
12.6 Health Physics Program

13. Conduct of Operations

13.1 Organizational Structure of Operating Organization

13.1.1 Operating organization
13.1.2 Qualifications of nuclear plant personnel

13.2 Training and Qualification/Licensing

13.3 Emergency Preparedness

13.3.1 Emergency Management
13.3.2 Emergency response facilities
13.3.3 Assessment of accident progression, radioactive releases and the consequences of accidents
13.3.4 emergency plan considerations for multi-unit sites
13.3.5 Emergency preparedness (post Fukushima emergency preparedness)
13.3.6 Maintaining emergency preparedness

13.4 Operational Programme Implementation

13.4.1 Maintenance, surveillance, inspection and testing
13.4.2 Core management and fuel handling
13.4.3 Management of ageing
13.4.4 Control of modifications

15.3 Identification and Classification of PIEs and Accident Scenarios

15.3.1 Basis for classification of PIEs and accident scenarios
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

15.5 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
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 individual groups of PIEs/AOOs and design basis accidents (typical but not limited to)

15.5.3.1 Analysis of core cooling and system pressure for reactivity induced accidents
15.5.3.2 Analysis of core cooling and system pressure for a decrease of reactor coolant flow
15.5.3.3 Analysis of system pressure for increase of reactor coolant inventory
15.5.3.4 Analysis of core cooling and system pressure for increase of heat removal by the secondary circuit
15.5.3.5 Analysis of core cooling and system pressure for decrease of heat removal by the secondary circuit
15.5.3.6 Analysis of loss of electrical power supply
15.5.3.7 Analysis of core cooling for LOCAs
15.5.3.8 Analysis of primary to secondary circuit leakage (with and without iodine spiking)
15.5.3.9 Analysis of pressurized thermal shocks
15.5.3.10 Analysis of pressure-temperature transients in the containment
15.5.3.11 Analysis of radioactivity transport during bounding design basis accidents
15.5.3.12 Analysis of fuel handling events
15.5.3.13 Analysis of internal hazards
15.5.3.14 Analysis of natural external hazards
15.5.3.15 Analysis of man-made external hazards
15.5.3.16 Radioactive release from subsystem or component

15.5.4 Evaluation for Design Extension Conditions without Core Melt

15.5.4.1 Anticipated transients without scram (complex sequences)

15.5.4.2 DEC other than ATWS (complex sequences)

15.6 Evaluations of Severe Accident/Design Extension Conditions with Core Melt

15.6.1 Identification of severe accident scenarios
15.6.2 Severe accident prevention
15.6.3 Severe accident mitigation
15.6.4 Containment performance capability
15.6.5 Accident management
15.6.6 Consideration of potential design improvements

15.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with Spent Fuel Pool

Appendix 15A
(A summary (in tabular form) of the computer codes used, as well as the reactivity coefficients (e.g., moderator density, moderator temperature, and Doppler coefficients) and initial thermal power assumed in the analysis of each transient or accident should be provided.)

Appendix 15B
(The reactor trip functions (in tabular form), ESF functions, and other equipment available to mitigate each transient and accident should be provided.)

Appendix 15C
(A summary (in tabular form) of the trip setpoints and the total delay times of the reactor protection system and ESF actuation system assumed in the analyses of the transients and accidents should be provided. The table should also include the trip setpoint values specified in the Technical Specification.)

Appendix 15D
(All single failures considered to determine the limiting single failure used in each transient or accident analyzed)

Appendix 15E
(The limiting single failure selected for each transient and accident analyzed should be provided in a tabular form.)

Appendix 15F
(A list of non-safety related system and equipment used to mitigate transients and accidents should be provided in a tabular form.)

16. Technical Specifications for Operation

16.1 Use and Application
16.2 Safety Limits
16.3 Limiting safety system settings
16.4 Limiting conditions for normal operation, and surveillance requirements.
16.5 Administrative Requirements
16.6 Bases
16.7 Documentation

17. Management of Safety and Quality Assurance

17.1 Management of Safety

17.1.1 Specific Aspects of Management of Safety Processes
17.1.2 Organisational structure, responsibility and authority
17.1.3 Consideration of Safety Culture
17.1.4 Monitoring and Review of Safety Performance
17.1.5 Management of organizational changes

17.2 Quality Assurance Programme

- 17.2.1 QA policies/goals/objectives
- 17.2.2 Items Governed by QA program
- 17.2.3 Planning
- 17.2.4 Control of documentation and records
  - 17.2.5.1 Identification and retrievability of records
  - 17.2.5.2 Storage of Records
- 17.2.5 Training, qualification/certification
- 17.2.6 Conformance to regulatory requirements
- 17.2.7 Grading /graded approach
- 17.2.8 Procedures and work instructions
- 17.2.9 Management review

17.3 Quality Assurance in Design Control

- 17.3.1 Responsible design organisation
- 17.3.2 Measures for design control
- 17.3.3 Verification & validation
- 17.3.4 Design interfaces
- 17.3.5 Design changes

17.4 Quality Assurance in Procurement Control

- 17.4.1 Procurement specifications
- 17.4.2 Design documents control
- 17.4.3 Vendor evaluation
- 17.4.4 Use of commercial grade items
- 17.4.5 Conformance to procurement documents

17.5 Quality Assurance in Construction Control

- 17.5.1 Selection of contractors,
- 17.5.2 Review of contractors' QA programme
- 17.5.3 Approval of sub-contractors and suppliers
- 17.5.4 Document control
- 17.5.5 Control of design change information
- 17.5.6 Housekeeping during construction and installation
- 17.5.7 Activities requiring special cleanliness control
- 17.5.8 Industrial safety
- 17.5.9 Control of materials and equipment
- 17.5.10 Control of measuring and test equipment
- 17.5.11 Verification of construction work
- 17.5.12 Hand over and transfer of responsibilities

17.6 Quality Assurance in Control of Special Processes

17.7 Quality Assurance in Inspection

- 17.7.1 Establishment of inspection programme
- 17.7.2 QA plans identifying hold/witness points

17.8 Test Control

- 17.8.1 Scope
- 17.8.2 Provisions in test procedure
- 17.8.3 Control of measuring and test equipment

17.9 Quality Assurance in Handling, Storage and Shipping

17.10 Control of Non-conforming Materials, Parts or Components

17.11 Preventive and Corrective Action

17.12 Audits

18. Human Factor Engineering

18.1 Review of NPP Operating Experience

18.1.1 Objectives and scope
18.1.2 Methodology
18.1.3 Results

18.2 Staffing and Qualifications

18.2.1 Objectives and scope
18.2.2 Methodology
18.2.3 Results

18.3 Human Reliability Analysis

18.3.1 Objectives and scope
18.3.2 Methodology
18.3.3 Results

18.4 Human- System Interface Design

18.4.1 Objectives and scope
18.4.2 Methodology
18.4.3 Results

18.5 Procedure Development

18.5.1 Objectives and scope
18.5.2 Methodology
18.5.3 Results

18.6 Training Programme Development

18.6.1 Objectives and scope
18.6.2 Methodology
18.6.3 Results

18.7 Verification and Validation of HFE Results

18.7.1 Objectives and scope
18.7.2 Methodology
18.7.3 Results

18.8 Design Implementation

18.8.1 Objectives and scope
18.8.2 Methodology
18.8.3 Results

18.9 Human Performance Monitoring

18.9.1 Objectives and scope
18.9.2 Methodology
18.9.3 Results

19. Probabilistic Safety Assessment

19.1 Level 1 PSA (Internal events, Full Power)
19.2 Level 1 PSA - External Events
19.2.1 Level 1 PSA (External Events –Fire)

ABBREVIATIONS

AC	Alternate Current
AERB	Atomic Energy Regulatory Board
ALARA	As Low As Reasonably Achievable
AMG	Accident Management Guideline
AMP	Ageing Management Programme
AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CCC	Construction Completion Certificate
CDF	Core Damage Frequency
CRDM	Control Rod Drive Mechanisms
CRDS	Control Rod Drive Systems
DBA	Design Basis Accident
DBE	Design Basis Event
DBT	Design Basis Threats
DC	Direct Current
DEC	Design Extension Conditions
EIA	Environmental Impact Assessment
EOPs	Emergency Operating Procedures
EPZ	Emergency Planning Zone
ESF	Engineered Safety Features
FMEA	Failure Mode and Effects Analysis
FOAK	First Of A Kind
FSAR	Final Safety Analysis Report
HEP	Human Error Probability
HFE	Human Factor Engineering
HRA	Human Reliability Analysis
HSIs	Human System Interfaces
HVAC	Heating, Ventilation, Air Conditioning
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ISI	In Service Inspection
IHX	Intermediate Heat Exchanger
LM	Life Management
LOCA	Loss of Coolant Accident
LSSS	Limiting Safety System Settings
MCR	Main Control Room
MOF	Main Out Fall
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
LHGR	Linear Heat Generation Rate
O&M	Operation & Maintenance
OLCs	Operational Limits and Conditions
OER	Operating Experience Review
PIE	Postulated Initiating Event
P&IDs	Piping & Instrumentation Diagrams
PPS	Physical Protection System
PSA	Probabilistic Safety Assessment

PSAR	Preliminary Safety Analysis Report
PSF	Performance Shaping Factor
PSR	Periodic Safety Review
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RO	Responsible Organisation
SAR	Safety Analysis Report
SBO	Station Block Out
SL	Safety Limit
SMR	Stack Monitoring Room
SSC	Systems, Structures and Components
STD	System Transfer Document
TG	Turbine Generator
WBGT	Wet Bulb Globe Temperature

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		November 27 & 28, 2008
		April 2 &3, 2009
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		October 13 & 14, 2014
		November 11 & 12, 2014
		July 2 & 3, 2015
		July 17, 2015

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LIST OF SAFETY CODE, SAFETY GUIDES AND SAFETY MANUALS FOR REGULATION OF NUCLEAR AND RADIATION FACILITIES

Safety Series No.	Title
AERB/SC/G	Regulation of Nuclear and Radiation Facilities
AERB/NPP/SG/G-1	Consenting Process for Nuclear Power Plants and Research Reactors
AERB/NF/SG/G-2	Consenting Process for Nuclear Fuel Cycle Facilities and Related Industrial Facilities other than Nuclear Power Plants and Research Reactors
AERB/RF/SG/G-3	Consenting Process for Radiation Facilities
AERB/SG/G-4	Regulatory Inspection and Enforcement in Nuclear and Radiation Facilities
AERB/SG/G-5	Role of the Regulatory Body with respect to Emergency Response and Preparedness at Nuclear and Radiation Facilities
AERB/NRF/SG/G-6 (Rev.1)	Development of Regulatory Safety Documents for Nuclear and Radiation Facilities
AERB/SG/G-7	Regulatory Consents for Nuclear and Radiation Facilities: Contents and Format
AERB/SG/G-8	Criteria for Regulation of Health and Safety of Nuclear Power Plant Personnel, the Public and the Environment
AERB/NPP/SG/G-9	Standard Format and Contents of Safety Analysis Report for Nuclear Power Plants
AERB/NPP&RR/SM/G-1	Regulatory Inspection and Enforcement in Nuclear Power Plants and Research Reactors
AERB/NF/SM/G-2	Regulatory Inspection and Enforcement in Nuclear Fuel Cycle Facilities and Related Industrial Facilities other than Nuclear Power Plants and Research Reactors
AERB/RF/SM/G-3	Regulatory Inspection and Enforcement in Radiation Facilities