REGULATION ON ENSURING THE SAFETY OF NUCLEAR POWER PLANTS

Chapter One

GENERAL PROVISIONS

Article 1. (1) This Regulation defines the basic criteria and rules of nuclear safety and radiation protection of nuclear power plants (NPP), as well as the administrative provisions and the technical requirements for ensuring safety applicable for site selection, design, construction, commissioning and operation.

(2) The Regulation also settles the requirements for industrial and fire safety, emergency planning and emergency preparedness of a NPP, as long as they result from the implementation of the defence-in-depth concept.

(3) The Regulation covers the physical protection of a NPP only in terms of the safety and security interfaces.

Article 2. (1) A NPP shall be considered safe when the following provisions have been implemented altogether:

1. The radiation impact of a NPP in all operational states is maintained below the prescribed dose limits for internal and external exposure of the personnel and the public, and is kept as low as reasonably achievable;
2. Accidents without fuel melt do not cause radiation impact that necessitates implementation of protection measures for the public;
3. Accidents with fuel melt, resulting in early or large radioactive releases into the environment, are practically eliminated, while the other severe accidents (that are not practically eliminated) have only a limited radiation impact.

(2) During the design and operation of a NPP and the implementation of all relevant activities, measures shall be taken to:

1. control the radiation exposure of people and releases of radioactive substances into the environment;
2. limit the frequency of events the occurrence of which may result in loss of control of the core or of the fission chain reaction;
3. mitigate the consequences of such events if they occur.

Article 3. (1) The use of the defence-in-depth concept is the main prerequisite for prevention and mitigation of accident consequences and it shall be ensured with a suitable combination of:

1. an effective management system demonstrating the clear commitment of the NPP management to ensure priority to safety and development of a high level of safety culture;
2. selection of a suitable site and combining a conservative design with appropriate engineering solutions to provide diversity, redundancy and safety margins, mainly through the use of:

a) design, technology and materials of high quality and reliability;

b) systems and design parameters that control and limit the reactor installation operation;

c) appropriate combination of inherent and engineered safety features.

3. comprehensive operating, emergency operating and accident management instructions.

(2) The defence-in-depth concept shall be applied at all stages of the NPP lifetime. Depending on the activities performed, independent levels of safety shall be identified where no single technical, human or organisational error or fault may result in significant harmful consequences, and the combination of such errors or faults shall have a very low probability.

Article 4. (1) A nuclear power plant shall be designed, sited, constructed, commissioned and operated in such a way as to meet the safety objectives in the following areas:

1. normal operation, anticipated operational occurrences, and prevention of accidents;
2. accidents without nuclear fuel melt;
3. accidents with nuclear fuel melt;
4. independence between all levels of defence-in-depth;
5. safety and security interfaces;
6. radiation protection and radioactive waste management;
7. competent leadership and effective management for safety.

(2) The safety objectives during normal operation, anticipated operational occurrences, and prevention of accident are the following:

1. reducing the frequency of deviations from normal operation by enhancing the plant capability to remain stable within the operational limits and conditions;
2. reducing the potential of deviations from normal operation to escalate into accidents by enhancing the plant capability to control them.

(3) The safety objectives for accidents without nuclear fuel melt are to prevent fuel damage through engineering and administrative provisions by demonstrating that:

1. the probability of fuel melt has been minimized to the extent practicable taking into account all types of failures, external hazards and their realistic combinations;
2. such accidents will not induce off-site radiological impact or will not impose necessity of iodine prophylaxis, sheltering or evacuation as protective measures for the public;
3. the radioactive releases of all sources of ionising radiation have been minimized to the extent practicable.
4. at the stages of site selection and design, measures have been taken to decrease the impact of external hazards or malicious acts.

(4) The safety objectives for accidents with fuel melt are to reduce potential radioactive releases to the environment resulting from accidents (in the reactor and/or in the spent nuclear fuel pool), also in the long term (considering the time needed to maintain safety functions) by meeting the following criteria:

1. accidents with fuel melt, resulting in early or large radioactive releases to the environment shall be practically eliminated;
2. for accidents with fuel melt that cannot be practically eliminated, design provisions shall be implemented such that only limited protective actions in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the NPP, limited sheltering, no long-term restrictions in food consumption), and that sufficient time shall be available to implement these measures.

(5) In order to reach an overall strengthening of the defence-in-depth to the extent practicable, the independence between all levels of defence shall be enhanced, particularly by using the diversification principle.

(6) The safety objective under para. 1, item 5 is to design and apply the safety measures and security measures in a well-considered and integrated manner. Synergies between safety and security enhancements shall be sought.

(7) The safety objectives under para. 1, item 6 are reducing to the extent practicable by design provisions, for all operating states, the following:

1. individual and collective doses for the personnel;
2. radioactive discharges to the environment;
3. radioactive waste quantity and activity.

(8) The safety objective under para. 1, item 7 is to achieve competent leadership and effective management for safety, starting at the design stage. This implies that the operating organisation:

1. establishes effective management of safety of the NPP design, and have competent personnel and sufficient available technical and financial resources to bear full responsibility for ensuring safety.
2. implements such measures that allow the staff of all other organisations involved in site surveys, design, construction, commissioning and operation to demonstrate their awareness of safety issues related to their job or their personal role in ensuring safety.

Chapter Two

OPERATING ORGANISATION

Section I

General Requirements

Article 5. (1) The operating organisation shall bear the full responsibility for ensuring safety, including when other entities perform works or provide services to the NPP, as well as in relation to the activity of the specialised regulatory authorities in the field of using nuclear energy and ionising radiation.

(2) The operating organisation shall ensure safety during site selection, design, construction, commissioning and operation of an NPP in accordance with the requirements of the ASUNE (Act on the Safe Use of Nuclear Energy), the regulations for its application, and the conditions of the permits and licences for operation issued by the Chairperson of the Nuclear Regulatory Agency.

Article 6. (1) During site selection for a NPP the operating organisation shall be responsible for the scope of site characteristics surveys, the selection of competent contractors, the arrangement and coordination of all surveys and the use of the results obtained in the NPP design.

(2) During NPP design the operating organisation shall be responsible for the implementation of all applicable safety requirements, and the additional requirements to the NPP design prescribed by the Nuclear Regulatory Agency.

(3) During NPP construction, irrespective of the provisions of the contract with the main

contractor, the operating organisation shall be responsible for the coordination of the activities performed by the design, construction and installation organisations and equipment suppliers, as well as for the quality of the activities performed and materials used and their compliance with the detailed design and technical specifications.

(4) During NPP commissioning, irrespective of the provisions of the contract with the main contractor, the operating organisation shall be responsible to demonstrate full compliance of the performance of the structures, systems and components (SSC) and the power unit as a whole, with the design basis and safety analyses.

(5) Throughout the NPP operation, the operating organisation shall ensure safety during the implementation of all activities on the NPP site, including conduct of operation, accident prevention and mitigation of the consequences, nuclear material accounting and control, radiation monitoring of premises, the site, and the environment in area with dimensions substantiated in the technical design and the intermediate safety analysis report of the NPP. The operating organisation shall use the NPP only for the purposes it was designed and constructed for.

Article 7. (1) The organisational structure of the operating organisation that is responsible for implementation of the tasks at each of the stages in Article 5, para. 2, shall be justified from the point of view of ensuring safety and its priority over other needs.

(2) Responsibilities, authorities and lines of interaction shall be clearly defined for all personnel performing activities that have an impact on safety.

(3) Changes in the organisational structure that can be of significant importance to safety, shall be justified in advance, carefully planned and evaluated after implementation in terms of achieving the intents related to safety.

(4) The top management of the operating organization shall establish, maintain and demonstrate leadership and management for safety to promote a strong safety culture in the performance of activities at all stages of the life cycle of the nuclear power plant. Leadership for safety shall be demonstrated successfully at all organizational levels.

(5) The senior management of the operating organization shall ensure:

1. the implementation of the safety policy and the achievement of its objectives;

2. the consistency of all plans, strategies, goals and objectives with the safety policy, and management of their collective impact in such a way that safety is not compromised by other priorities;

3. decision-making at all levels taking into account the priorities and accountabilities for ensuring safety;

4. the acceptance of personal responsibility for safety by all persons in the organization.

(6) Managers at all levels shall:

1. develop competencies for leadership for safety, demonstrate commitment to safety and foster a strong safety culture;

2. promote values and expectations for safety by means of their decisions, statements and actions;

3. ensure that relevant professional knowledge, skills and experience of individuals under their responsibility are used in making decisions.

Section II

Management System

Article 8. (1) The operating organisation shall establish, implement, assess and continuously improve a management system that has the main objectives to ensure and enhance the NPP safety, and to encourage and support a high level of safety culture of the personnel.

(2) The management system shall integrate all elements of management in such a way as to allow all the requirements for protection of human health and environment, provision of security and quality assurance, financial aspects and other needs of the operating organisation to be considered coherently with safety requirements to avoid potential negative impact on safety.

(3) The requirements of paras. 1 and 2 are relevant to the whole plant lifetime as well as for the overall duration of the activities performed during all operational states and accident conditions.

Article 9. (1) The application of management system requirements shall be graded with the purpose of effective use of resources, on the basis of consideration of:

1. the complexity and significance of each activity and its products;
2. the hazards and the magnitude of potential impact associated with each activity and its products;
3. the possible consequences of incorrect performance of an activity or of failed product that is not fit for its purpose.

(2) The graded approach shall be applied to products and activities of each process.

Article 10. (1) The documentation of the management system shall include the following:

1. policy statements of the operating organisation;
2. description of the management system;
3. description of the organisational structure of the operating organisation;
4. description of the functional responsibilities, subordination, levels of authority and interactions with personnel who manage, perform and assess the work;
5. description of the interactions with external organisations;
6. description of processes and information that explain how work is to be prepared, reviewed, implemented, reported, assessed and improved.

(2) The documentation of the management system shall be understandable to those who use it, readable, readily identifiable, up to date and available at the places of their usage.

Article 11. (1) The management of the operating organisation shall adopt long-term objectives and strategies, short-term objectives and plans of the organisation consistent with the safety policy and shall disseminate them to the personnel in a way that ensures the understanding of the overall impact on safety.

(2) To assess the progress in implementation of the objectives, strategies and plans, performance indicators shall be applied at different levels of the operating organisation.

(3) Plans and strategies shall be periodically reviewed considering the quantitative indicators and, when necessary, measures shall be taken to correct deviations from them.

Article 12. The management system shall clearly define when, how and who shall make decisions for ensuring nuclear safety throughout the operation of the NPP.

Article 13. Management at all levels of the operating organisation shall demonstrate its commitment to the establishment, implementation, assessment and continuous improvement of the management system, and shall encourage all the personnel to take part in its implementation and continuous improvement.

Article 14. Within the management system, the operating organisation shall:

1. identify and provide for the necessary resources (personnel, infrastructure, working conditions, information and knowledge, suppliers, material and financial resources) for the implementation of all activities it is responsible for at the NPP site;
2. specify the requirements for the personnel qualification at all levels and provide training to achieve the required level of qualification;
3. define, provide for, support, and periodically reassess infrastructure and working conditions that are necessary for safe implementation of activities in compliance with the relevant requirements;
4. takes into account in an integrated manner the human and organizational factors that influence safety.

Article 15. (1) The management system shall define the processes needed for meeting the objectives of the operating organisation, and providing for the means necessary to meet all requirements and deliver the anticipated products. These processes shall be planned, implemented, assessed and continuously improved.

(2) The following shall be ensured during the development of each process:

1. All safety, health, environmental, security, quality, regulatory, legal and economic requirements that are applicable to the process, have been identified and implemented;
2. Hazards and radiation risk, and all necessary measures that mitigate their consequences, have been identified;
3. Interrelations with other processes have been identified;
4. Input data and initial conditions of the process have been specified;
5. The sequence of process implementation has been described;
6. The products of the process have been identified, as well as the inspection criteria and methods for these products;
7. The criteria for quantitative evaluation of the process have been identified.

(3) Activities and interactions among different contractors or groups involved in a process shall be scheduled, controlled and managed in such a way as to ensure effective communication and clear distribution of responsibilities.

Article 16. (1) The methods necessary to ensure the effectiveness of both the implementation and the control of the processes and activities shall be determined and implemented.

(2) All activities for inspection, testing, verification, validation and their success criteria, and the performance responsibilities, shall be specified for each process.

(3) For each process it shall be specified when and at what cases the activities under para. 2 are implemented by individuals or a team who are not involved in the performance of this process.

Article 17. (1) The documentation of the management system (policy statements, instructions, procedures, specifications, drawings, training aids and all other documents that describe processes, specify requirements or products) shall be controlled.

(2) Changes to documents shall be reviewed, assessments shall be documented, and changes shall be approved at the same management level at which the documents have been approved. Document users shall familiarise with them and shall use appropriate and correct documents.

Article 18. The management system shall specify the records that shall be kept and controlled. All records shall be clear, complete and compiled, identified and easily retrievable for the period of safe-keeping as specified for each record.

Article 19. (1) The management system shall specify the necessary control of processes and activities performed by external contractors while considering the full responsibility of the operating organisation to ensure safety.

(2) Selection of product and service suppliers shall be based on criteria established in advance and their performance shall be periodically assessed.

(3) Purchase requirements shall be clearly and explicitly stipulated in procurement documentation. Prior to using the supplied products, the operating organisation shall obtain evidence that these products meet the specified requirements.

(4) Prior to acceptance, performance or the operational use of the supplied products they shall be inspected, tested, verified and validated to prove that the activities and their products meet the stipulated requirements and that the products satisfactorily meet their designated function.

Article 20. (1) To confirm the capability of processes to achieve the anticipated results and identify possibilities for improvement:

1. the efficiency of the management system shall be controlled and assessed;
2. managers shall conduct self-assessment of the performance of processes or activities they are responsible for;
3. independent assessments shall be regularly performed on behalf of the operating organisation.

(2) Self-assessments and independent assessments shall cover leadership for safety and safety culture under Article 25, including underlying attitudes and behaviours.

Article 21. (1) The operating organisation shall establish an organisational unit authorised and with responsibilities to perform independent assessments. The personnel of this organisational unit shall be assigned depending on the NPP lifetime stage. The individuals performing the independent assessments may not self-assess their own work.

(2) Assessment results shall be analysed and all appropriate measures taken. Decisions made and reasons for actions taken shall be documented and disseminated to the NPP personnel.

Article 22. (1) The management system shall be reviewed at scheduled intervals. The reasons for non-compliances shall be established and timely corrective measures shall be taken to avoid their recurrence.

(2) Management system reviews shall identify the need for changes or improvement of policies, strategies, plans or processes;

(3) The potential safety impact of changes to the management system shall be analysed prior to their implementation. Changes with potential impact on safety shall be justified, planned, executed and evaluated using a graded approach.

Article 23. (1) Processes and products that do not comply with the specified requirements, shall be identified, isolated, controlled, documented and reported. The influence of non-compliances shall be assessed and such processes and products shall be either acceptable for further use, corrected, or discarded.

(2) The acceptance of processes and products under para. 1 shall be subject to approval, and in case they have been changed or corrected, they shall be tested to demonstrate their compliance with the requirements or anticipated results.

Article 24. (1) The management system improvement plans shall include plans for provision of adequate resources. The improvement activities shall be controlled during their implementation and their efficiency shall be checked.

(2) The personnel of the operating organization shall be trained in the relevant aspects of the management system to ensure its implementation and to foster its continuous improvement.

Article 25. The management personnel at all levels of the operating organisation shall consistently demonstrate, support and encourage such way of thinking and behaviour that leads to sustainable and high level of safety culture. The actions of the management personnel shall encourage a culture of open reporting, as well as critical attitude, awareness and preparedness for challenges and states that jeopardize safety.

Article 26. The management system shall provide for the means of systematic development, support and encourage the desired and anticipated way of thinking and performance, resulting in high level of safety culture. The adequacy and effectiveness of these means shall be assessed as part of the self-assessment and review of the management system.

Article 27. The operating organisation shall ensure that its suppliers and external contractors perform their safety related activities according to Articles 25 and 26.

Chapter Three

SITE CHARACTERISTICS

Section I

General Requirements

Article 28. The characteristics of potential NPP sites and of the selected site shall be assessed and documented as an integral part of the overall NPP safety analysis.

Article 29. (1) The following groups of characteristics shall be identified during the NPP site selection:

1. External hazards of natural origin that may affect the NPP;
2. External hazards of human induced origin that may affect the NPP;
3. Site characteristics that affect the NPP impact on the public and environment (dispersion of radioactive substances, population density).

(2) The site selection shall be based on a comprehensive weighted analysis of all characteristics and priority given to those that have a direct impact on the NPP safety and security.

Article 30. (1) The assessment of processes, phenomena, and factors of natural and human induced origin for the selected site shall confirm the possibility for implementation of protective measures to prevent their impact and meet the safety objectives under Article 4.

(2) In case the site under consideration for a new NPP is in close proximity to the site of an existing NPP, the impact of the existing nuclear facilities shall be taken into account.

Article 31. Siting of a NPP is not allowed:

1. in territories where this is prohibited by law, or on sites that do not comply with the requirements of environmental protection, radiation, fire safety and physical protection, or any other requirements stipulated by a normative act;
2. on sites where measures for practical elimination of large or early releases of radioactive substances into the environment as a result of external effects cannot be applied;
3. on sites of pronounced seismic activity combined with surface deformations;
4. on sites located up to 5 km away from a known capable fault or its branching where a rupture and/or deformation on or near the ground surface may be expected;
5. on sites with a potential danger of old or new landslides being activated;
6. on sites of nonconsolidated soils or with a potential for liquefaction, subsidence, collapse, terrain tilting, or erosion of slopes where the implementation of safety ensuring engineering measures is practically impossible;
7. on sites where karst, suffusion and karst-suffusion processes take place;
8. on sites within the zones of passage of snow avalanches or mud streams, and in areas of mud volcano activity;
9. on sites exposed to impact of tsunami waves;
10. on mine development sites the stability of which cannot be ensured;
11. on sites of active exchange of surface and groundwater.

Section II

Investigation of Natural and Human Induced Hazards for Site Selection

Article 32. The following engineering investigations and studies of the processes, phenomena and factors of natural origin that may impact the NPP safety shall be performed for the NPP region and the NPP site:

1. The tectonic activity characteristics shall be defined as follows:

a) location of faults, potential earthquake foci zones and geodynamic zones with respect to the NPP site, indicating the orientation and boundaries of potentially hazardous fault zones;

b) amplitudes, speed and gradients of the latest and contemporary movements of the earth crust, parameters of potential dislocations;

c) characteristics of the capable fault areas (geometric schemes, dislocation amplitudes and directions along the faults, data of the latest activity known);

2. The following shall be identified within the site boundaries:

a) characteristics of seismic movement (accelerations, speed, dislocations, response spectra) of an earthquake with frequency of occurrence not higher than 10^{-2} events per year (seismic level - 1), and for a safe shutdown earthquake (SSE) with frequency of occurrence not higher than 10^{-4} events per year (seismic level -2) at the level of the natural terrain of the site;

b) hazard of landslide displacements of the slopes considering the ground layer conditions and seismic fluctuations with an intensity of up to safe shutdown earthquake including, and also when the impact of ground-waters, tectonic deformations and contemporary geodynamic processes are considered;

c) possibility of karst, suffusion and karst- suffusion processes development;

d) presence of specific ground layers (biogenic, collapsing, uplift, saline, alluvial, human induced), the thickness of these layers and their physical-mechanical properties (deformation modules, strength properties, etc.);

e) areas of water-saturated unconnected layers susceptible to liquefaction during seismic impacts, and the boundary values of ground acceleration with potential for liquefaction;

f) rising of groundwater level and flooding of the site as a result of high level groundwater coming from dams, filtration from irrigated lands, water flows, precipitation, snow melting;

g) rare phenomena characteristics such as tornadoes (including their frequency of occurrence, intensity, maximum tangential values at the periphery and tornado forward movement speed, drop of pressure between the periphery and the centre of the tornado);

3. The following shall be defined for an NPP site: the maximum water level and duration of possible flooding due to precipitation, intensive snow melting, high level in water basins, ice blocking of rivers, avalanches and slides; the characteristics of maximum probable flooding from watercourses with a frequency not higher than 10^{-4} events per year combined with high tides and waves caused by winds;

4. For an NPP site situated at a sea, lake or dam coast the probability of occurrence and the maximum height of tsunami or seiches waves shall be studied, considering the seismic tectonic conditions, shore configuration, landslides and collapse in the water;

5. The impact on safety of other processes, phenomena and factors of natural origin (hurricane, extreme precipitation, icings, thunderstorms, dust-storms and sand-storms, erosion of river and water basin banks) shall be identified for an NPP site.

Article 33. (1) The NPP region and the respective site shall be investigated to identify sources of potential human induced hazards regardless of their frequency (recurrence).

(2) Sources of human induced hazards that may cause explosions, fires, releases of explosive, toxic, and corrosion-active substances shall be identified.

(3) All potential stationary and mobile sources of explosions, including industrial and military sites for production, processing, storage and transportation of chemicals and explosive substances, and ammunition dumps shall be analysed and the impact parameters shall be identified for the most dangerous explosion.

(4) All stationary and mobile potential sources of emergency release of chemically active substances, including industrial and military sites for processing, use, storage and transportation of toxic and corrosive substances shall be analysed.

(5) The impact parameters on an NPP and the respective probabilities shall be determined for events induced by:

1. explosions and fires, releases of explosive, inflammable, toxic and corrosive gases and substances from industrial facilities, ground and water conveyance facilities;

2. a contemporary passenger aircraft crash;

3. floods, including those related to reaching the waterfront as a result of breaking of dams located upstream the NPP site;

4. accidents of a water vessel along waterways and in harbour zones occurring together with explosions and fires, releases of dangerous chemicals, provided the NPP is situated within their range of impact;

5. electromagnetic emissions (fields);

6. external fires (forest areas, peateries, flammable liquids);

7. deformations and other factors arising on developing underground resource deposits, carrying out excavation works, including tunnel construction, mines and quarries exploitation and their accidental destruction;

8. water level fluctuations of the NPP water supply source.

Article 34. (1) The factors that affect the NPP impact on the public and the environment shall be considered during the NPP site selection.

(2) The aerologic, hydro-meteorological, hydrogeological and geochemical conditions of radionuclide dispersion, migration and accumulation, also the natural radiation background shall be investigated in the NPP region and predictions shall be made for changes in these conditions over the NPP operating lifetime.

(3) Atmospheric dispersion shall be assessed by taking into consideration slight wind, calm weather, air temperature, near-surface and altitude inversions, atmospheric stability, precipitation and fogs in the NPP region.

(4) The characteristics of radionuclide migration in surface- and ground-water and deposition of radionuclides at the bottom of water basins shall be defined considering the following:

1. possible radioactive contamination of drainage and groundwater;

2. radionuclides physical and chemical properties;

3. kinetics of geochemical reactions and possible changes in the mineralogical composition of layers;

4. lithological composition and thickness of water-bearing and watertight strata, the ground layers in the weathering zone and the soil layer;

5. sorption capacity of deposits, ground layers and soil layers with respect to radionuclides and hazardous chemical substances;

6. direction and speed of contaminated streams towards the release places (drain-pipes, water bodies, water intake wells, etc.);

Article 35. To ensure reliable and long-term residual heat removal of nuclear fuel, the extreme temperatures of water and air and their duration, air humidity, water flow rate, minimum water level, and quantity of algae shall be specified.

Article 36. (1) The current and future distribution of population and use of land and water sources within the NPP region shall be identified for the purposes of emergency planning.

(2) To determine the population distribution, data from the latest population census shall be used, revised in terms of the direction and distance to the NPP.

(3) The data used in the food chain for assessment of the NPP radiological impact on the population shall be provided by the study on the use of land and water sources by the population.

Article 37. (1) Studies, analyses and engineering shall be performed efficiently and in good quality at all stages of the NPP site investigation.

(2) Engineering investigations shall be carried out in sufficiently large regions and areas to cover all special features and spheres of influence that may be of importance for identification of natural and human induced hazard sources, and the characteristics of the studied events.

(3) To describe natural and human induced hazards, suitable design parameters shall be chosen or developed considering the uncertainties of data used for investigations and studies.

(4) A meteorological monitoring programme shall be developed and implemented for the NPP site to measure the basic meteorological parameters at appropriate altitudes and places. Data from at least one year shall be processed together with all related data from other sources.

(5) Results of field researches, laboratory tests, geotechnical analyses and other studies shall be summarized in a sufficiently detailed report allowing for independent assessment.

Chapter Four

DEFENCE IN DEPTH AND DESIGN BASIS

Section I

Defence in Depth Implementation in the Design

Article 38. The defence in depth concept shall be implemented in the design through provision of a number of physical barriers and several levels of protection for ensuring protection against the impact of ionizing radiation and consequences mitigation in case of preventive measures failure.

Article 39. (1) The number of the necessary physical barriers shall be specified on the basis of an assessment of quantities and isotope composition of radionuclides that might be released into the environment, the efficiency of the different barriers, their vulnerability to internal and external impacts, as well as the potential consequences in case of a failed barrier.

(2) The NPP design shall envisage independent physical barriers for each significant source of ionizing radiation. The assessment under para. 1 shall cover both all risks caused by all the nuclear fuel on the NPP site, and the risks due to other sources of ionizing radiation.

Article 40. (1) The levels of defence shall have the objective of preventing the following to the extent practicable:

1. conditions leading to breaking the integrity of the physical barriers;
2. failure of a physical barrier when challenged (under the conditions of item 1.);
3. failure of a physical barrier as a consequence of a failure of another physical barrier;
4. possibility of unfavourable consequences resulting from errors in the operation and servicing of structures, systems and components (SSCs).

(2) The purpose of the first level of defence shall be to prevent abnormal operation, failures of SSCs important to safety which necessitate a conservative layout, design, construction, maintenance and operation of the NPP in compliance with a management system and proven engineering practices. This objective shall be accomplished with the help of:

1. selection of appropriate design standards and materials;
2. quality control during component manufacturing, construction and commissioning;

3. decreasing the risk of internal hazards;

4. application of processes and procedures for NPP design, manufacturing of components, construction of the NPP, maintenance, surveillance and testing of important to safety SSCs;

5. the method of operation and consideration of operating experience;

6. detailed analyses of the operation, maintenance and management system.

(3) The purpose of the second level of defence shall be to detect and control deviations from normal operation to prevent anticipated operating occurrences escalating into accident conditions. The second level of defence shall require:

1. the design to provide systems and design features that control and limit the reactor facility operation;

2. efficiency of design systems and features to be verified by safety analyses;

3. development of operating procedures to prevent deviations from normal operation and anticipated operational occurrences, to mitigate their consequences and return the NPP to a safe state.

(4) The objective of the third level of defence shall be to prevent nuclear fuel damage and off-site release of radioactive substances; to bring the reactor installation to a safe state in case of anticipated operational occurrences and accident sequences, through the use of the inherent safety features, and safety systems and emergency procedures envisaged for that purpose.

(5) The purpose of the fourth level of defence shall be to control and manage accidents that have occurred at precedent levels of defence or were caused by extreme external events in order to return the reactor installation to a stable safe state and postpone in time the consequences of severe accidents. At this level, the most important task is to ensure the function for retaining radioactive substances within the containment, thus decreasing radioactive releases into the environment to a level as low as reasonably achievable.

(6) The purpose of the fifth and last level of defence shall be to mitigate the radiological consequences to the public caused by radioactive releases as a result of possible accident conditions. This requires that an adequately equipped emergency response centre, emergency plan and emergency procedures, and an off-site emergency response should be in place.

Article 41. (1) The implementation of the defence in depth shall ensure that each level of defence is independent and efficient at all times so that the loss or inefficiency of one level shall not affect the functionality of the other levels.

(2) Independence of SSCs performing safety functions at different levels of defence shall be ensured by simultaneously meeting the following conditions:

1. the capability to perform the required safety functions shall not be influenced by the operability or inoperability of SSCs which are part of safety functions at other levels of defence;

2. the capability to perform the necessary safety functions shall not be influenced by the consequences of postulated initiating events, internal and external hazards including, which require the functioning of the respective SSC.

(3) The design shall ensure adequate efficiency of the first two levels of defence to prevent the escalation into accidents of all failures or abnormal operation that may occur throughout the whole operational lifetime of the NPP.

(4) Systems and provisions for prevention of accidents with fuel melt shall be independent of systems and provisions specially designed to perform safety functions in case of a postulated

severe accident to such an extent as not to impede the implementation of these functions.

Section II

Design Basis

Article 42. The design basis shall specify the characteristics of a NPP and its SSCs necessary to perform safety functions for the purpose of:

1. ensuring safe operation within justified operational limits and conditions throughout the entire operational lifetime;
2. limiting of potential radiological impact within the boundaries of the NPP site so that in all operational states and accidents without fuel melt to avoid reaching the intervention criteria and taking protection measures for the public as stipulated in the Regulation pursuant to Article 123 of the Safe Use of Nuclear Energy Act;
3. prevention of accident progression and fuel melt in the reactor core and the spent fuel pool;
4. practical elimination of large or early releases of radioactive substances into the environment;
5. mitigation of consequences from potential releases from accidents that could not be practically eliminated, long-term localization of radioactive substances and maximum delay in time of eventual leakage.

Article 43. In all operating states of the nuclear facilities on the NPP site, the annual individual effective dose resulting from internal and external exposure of the public caused by the impact of all nuclear facilities on-site shall be maintained as low as possible and shall not exceed 0,15 mSv.

Article 44. (1) The NPP design shall have the necessary characteristics allowing for minimizing to the extent practicable the probability of nuclear fuel melt by considering all failures, external and internal hazards and realistic combinations of events.

(2) The safety assessment under para. 1 shall confirm that the mean value of the nuclear fuel melt frequency is less than 10^{-5} /year for a nuclear power unit based on the consideration of all operating states and all types of initiating events and hazards.

Article 45. The design basis shall incorporate design limits, operational states and emergency conditions, safety classification of SSCs, and significant methods and design assumptions applied in the design and safety assessment. The design basis shall be systematically specified and documented to reflect the actual state of the nuclear power unit.

Article 46. The continuous improvement principle as per Article176, para. 2 shall be applied in relation to the design basis. All reviews shall be carried out by using deterministic and probabilistic approach to identify the need and possibilities for improvement.

Article 47. The design limits shall include at least:

1. radiological and other technical acceptance criteria for all operational states and accident conditions;
2. criteria for protection of the fuel cladding integrity, including departure from nucleate boiling ratio, cladding temperature, nuclear fuel temperature, fuel rod tightness (integrity), and maximum allowable fuel damage during all operational states and accidents without nuclear fuel melting;

3. criteria for protection of the reactor coolant pressure boundary, including maximum pressure, maximum temperature, thermal and pressure transients and loads;

4. criteria for protection of the containment structure of the reactor installation, including temperature, containment pressure and allowable containment leak rates, also the necessary margins ensuring its integrity and leak tightness in case of extreme external events, severe accidents and in combinations of initiating events.

Article 48. (1) For all operational states and accidents without fuel melt, the NPP unit shall be capable of performing the following fundamental safety functions:

1. control of reactivity;
2. heat removal from the reactor core and the spent nuclear fuel;
3. confinement of radioactive substances to avoid releases into the environment.

(2) The NPP design shall include solutions aimed at mitigation of possible radioactive releases into the environment in case of an accident with fuel melt and the subsequent period, sufficient to maintain the fundamental safety functions as follows:

1. ensure subcriticality of the core to the extent practicable for a long time period and continuously maintain subcriticality in the spent fuel storage pools;
2. ensure residual heat removal from damaged fuel by independent and diversified systems and provisions, operable in the conditions of such an accident (including caused by an extreme external event);
3. maintain at all times the function "confinement of radioactive substances".

Article 49. (1) In order to specify all events that may have an impact on the safety of the reactor installation and the spent fuel pool, an initial list of all NPP states shall be developed - steady and transient states, anticipated operating events and initiating events as a result of single or multiple failures of SSCs, human errors, internal and external events and hazards.

(2) The selection of multiple failure events shall consider:

1. a postulated common cause failure or inefficiency of all trains of a safety system which performs a required safety function in the conditions of an anticipated operational occurrence or a postulated initiating event;
2. a postulated common cause failure of a safety system or a system important to safety performing a fundamental safety function in normal operational mode.

(3) The accidents with fuel melt in the reactor core and in the spent fuel pool that are not practically eliminated (severe accidents) shall be considered in the NPP design basis. Representative severe accident scenarios shall be identified and analysed to specify the boundary conditions for SSCs operation, accident management strategies and the possible safety improvement measures.

(4) The definition of the design basis shall take account of the possible internal events and hazards such as internal flooding, fires, explosions, and mechanical impacts caused by damaged high energy pipelines, impact of missiles from damaged components, load drops, in accordance with the requirements of Section IV of Chapter Five.

(5) External events shall be selected according to the requirements of Section IV, Chapter Five. The identification of the external events and the parameters of their impact shall take into account the interrelation and interfaces between safety and security according to Article 4, para. 1, item 5.

(6) The design basis shall cover possible combinations of single events, including internal and external hazards that can result in anticipated operational occurrences and accidents without fuel melt.

(7) The compiled preliminary list of initiating events and NPP states shall be reviewed by a combination of deterministic and probabilistic methods while considering relevant operating experience and safety assessments of other nuclear power plants. The results of research programmes and sound engineering evaluations shall be additionally considered for the selection of severe accident scenarios.

Article 50. The final list of events and accidents considered in the design shall cover bounding scenarios with the lowest margins to the acceptance criteria of the analysis results in order to define the boundary conditions according to which the safety important SSCs and their functional characteristics will be designed and manufactured. An indicative list of bounding scenarios and events to be considered in the NPP design is presented in an attachment to this Regulation.

Article 51. (1) To ensure appropriate reliability, efficiency and independence of the SSC important to safety, the following principles shall be applied in the design:

1. use of practically proven or experimentally tested and qualified components;
2. redundancy of the SSCs designated to perform their functions in single initiating events and multiple failure events;
3. diversity of SSCs performing one and the same safety function to protect against common cause failures;
4. fail-safe components to ensure that structures and systems perform their safety functions;
5. physical, structural or spatial separation of safety system trains;
6. functional isolation of interconnected circuits and systems.

(2) In the design of SSCs important to safety preference shall be given to solutions using inherent safety features (feedback control, thermal inertia and other natural processes).

(3) The human factor in the design is considered by:

1. automatic or passive devices to actuate and control the safety systems to an extent that does not require intervening of operators for a period of 30 minutes upon the initiating event occurrence;
2. technical features to preclude human errors and mitigate their consequences, including during the maintenance of SSCs important to safety.

(4) The failure of a system for normal operation shall not impede the performance of a safety function.

(5) Independent performance of the fundamental safety functions shall be ensured for each unit on multiple-unit sites.

Article 52. (1) All SSCs that are important to safety shall be identified and classified in safety classes according to their function and relation to safety.

(2) SSCs shall be classified by applying a structured approach based on a combination of deterministic and probabilistic methods, complemented by engineering judgement, where appropriate.

Article 53. (1) The process of SSCs safety classification shall cover as a minimum the following steps:

1. systematic identification of functions necessary for the performance of the fundamental safety functions for each operational or accident state;

2. categorization of the specified functions according to their importance to safety by using the results of the safety assessment;

3. specification and classification of SSCs that perform functions categorized as important to safety; the SSCs shall be allocated to safety classes according to the category of the function they perform;

4. specification and classification of other safety important SSCs designed for normal operation.

(2) The functions of SSCs under para. 1, item 2 shall be categorized by applying the principle for the least consequences as a result of the most frequent events, and by considering the following three factors:

1. the consequences as a result of a failure to perform a function;

2. the frequency of occurrence of the initiating event, combination of events or a common cause failure which require the performance of the respective function;

3. the contribution of the performed function for bringing reactor installation into a controlled or safe state.

(3) Safety-important structures, systems and components for normal operation that are to be considered in the course of classification shall be specified according to their significance for the:

1. protection of personnel and public from the effects of ionizing radiation;

2. prevention of failures that have not been considered in the design basis (including damage of the reactor pressure vessel);

3. decreasing the frequency of SSCs failures that may result in accidents;

4. mitigation of the consequences of external and internal hazards, considered in the design;

5. prevention of initiating events progression in case no other single failures have occurred.

(4) Structures, systems and components under para. 3 shall be directly classified in safety classes according to the consequences of their failure.

Article 54. (1) SSCs assigned to safety classes shall be designed, manufactured, installed, tested, operated and maintained in such a way as to ensure the quality and reliability required by the respective safety class.

(2) The following shall be specified for each safety class:

1. the appropriate standards and rules for design, manufacturing, installation and inspection;

2. the degree of redundancy, the need for emergency power supply, and qualification for operation under specific adverse environmental conditions;

3. the state of operability or inoperability of SSCs that is considered in the deterministic safety analysis;

4. the applicable quality requirements.

(3) The NPP design shall preclude the interference of individual SSCs important to safety, and shall ensure that a failure of an SSC of one safety class shall not cause a failure of an SSC of a higher safety class. The auxiliary systems, supporting SSCs important to safety, shall be assigned to the same safety class.

(4) The design shall consider an appropriate isolating device which shall be classified in a higher safety class for the cases of connecting SSCs of different safety classes, or of safety

classified SSCs to SSCs that are not safety related.

(5) SSCs that perform different functions shall be allocated to a safety class that corresponds to their most important function.

Article 55. (1) During the design and selection of structural materials of SSCs important to safety the impact on their characteristics and operational states operability throughout their entire lifetime shall be considered as well as the impacts under accident conditions when the performance of their functions is required.

(2) Equipment qualification procedures shall be developed and implemented to confirm that SSCs important to safety will be capable of performing their functions throughout their entire lifetime while considering possible environmental impacts and conditions (seismic impacts, impacts of temperature, pressure, humidity, vibrations, jet blasts, electromagnetic interference, ageing, irradiation and possible combination thereof) in all operational states and accident conditions.

(3) Working conditions of components of structures and systems important to safety shall be simulated by tests and full-scope experiments, or by alternative methods with proven equivalent effect in the cases these tests are practically not possible.

(4) Basis, methodologies, instructions and results of the classification and qualification of SSCs important to safety shall be systematically documented in a way allowing traceability and verification.

Article 56. (1) The design provisions for severe accident shall ensure implementation of the confinement safety function, and include all necessary measures to reliably maintain the integrity and leak-tightness of the containment in its capacity of the last barrier against spreading of radioactive substances into the environment. To this effect, technical means shall be provided for implementation of the following functions:

1. containment isolation in case of accidents and ensuring the leak-tightness of the containment penetrations;
2. temperature and pressure management in the containment, including filtration in case ventilation systems are provided;
3. monitoring and control of the concentration of combustible gases in the containment;
4. reducing the fission products in the containment;
5. diagnosis of the fuel status in the reactor and the spent fuel pool and provision of information for accident management decision-making by instrumentation qualified for severe accident conditions.

(2) Nuclear fuel melting shall be prevented in the cases where the containment isolation cannot be ensured within the required time (in reactor shutdown state and states with open containment), or in the cases resulting in containment bypass.

Article 57. (1) The design of SSCs and other technical provisions intended to perform safety functions during severe accidents shall implement the following principle requirements:

1. independence to the extent practicable from SSCs performing safety functions at other levels of defence-in-depth;
2. safety classification, seismic qualification and environmental qualification for the duration of the accident for which they are required to remain functional;
3. reliability commensurate with the performed function, which may require redundancy of

active components of the systems and of other engineered and measuring instruments.

(2) Independence of SSCs performing safety functions during severe accidents shall be ensured in terms of adequate power supply (direct and alternating current) for a justified period and considering possible natural phenomena and hazards.

(3) The effectiveness, capacity, and qualification of SSCs and the other technical means (including mobile equipment where it is provided) shall be proven and verified.

(4) When the accident management strategy requires the use of mobile equipment, permanent connecting points shall be provided and physical and radiological aspects considered. Appropriate procedures for qualification, maintenance, testing, inspections and personnel training shall be developed for the mobile equipment and the connection lines and points.

(5) The designs of all nuclear facilities at a single site shall be reviewed by applying a systematic approach to identify potential dependence on supply of process fluids and power supply or common service points. It shall be verified that common resources (personnel, technical provisions, materials) designated for severe accident management, will be sufficient and efficient for each nuclear facility.

(6) The autonomy of the nuclear facilities on the site in terms of supplies necessary for implementation of safety functions shall be analysed and ensured for a justified period, but not less than 72 hours.

Chapter Five

SAFETY ASSESSMENTS

Section I

General Requirements

Article 58. (1) Safety assessment is a systematic process that takes place along the life cycle of a NPP in order to determine the implementation of all applicable safety requirements in the design, including the extent to which the safety objectives of Article 4 have been met. Design and safety assessment shall be considered elements of a complex iterative process.

(2) Safety assessment shall be carried out during site selection, design, construction, commissioning, operation, implementation of modifications to the design and operation, periodic safety review and long term operation after the design lifetime.

(3) Safety assessment shall be performed on the basis of the results from performed safety analysis and additional scientific studies, analysis of the operational experience, as well as the data of applied proven technologies, design solutions and engineering practices.

Article 59. (1) Safety analysis shall be used as a method for assessing the NPP behaviour in a wide range of operational states and accident conditions, to confirm the adequacy of the design basis and design solutions, and to demonstrate the possibility of maintaining the NPP in a safe state.

(2) Safety analysis shall be performed using deterministic and probabilistic methods, while the level of safety achieved by the design shall be justified by a deterministic safety analysis.

Probabilistic analysis shall be used in the selection and categorisation of initiating events and accident sequences to complement the information on processes and the behaviour of the NPP, and to assess the contribution of the various safety aspects to the overall safety level.

(3) The results from the performed safety analysis shall be used to support the integrated decision making process on the NPP safety management.

Article 60. (1) The computer codes and mathematical models of the NPP used in the safety analysis shall be verified and validated for the respective application and the uncertainty of the calculated parameters shall be assessed. Verification and validation data shall be documented as part of the safety assessment.

(2) The computer codes and mathematical models shall be used only in the areas of application for which they have been validated.

(3) The input data and mathematical models used in the safety analysis shall be specific to the power unit and reflect the actual configuration of the SSCs. They shall be kept up to date both in the design process and during the operation of the NPP. When updating the data, models and calculations, account shall be taken of new data received, changes to the design and the operational procedures, and advanced methods and analysis tools.

(4) The deterministic and probabilistic safety analysis shall be carried out by experts who have undergone appropriate training to work with the relevant software and who possess the necessary analytical skills, knowledge and experience.

Article 61. (1) Safety assessment shall determine the possibility of deploying a NPP on the selected site based of the following criteria:

1. the scope of studies and investigation of the processes, phenomena and factors of natural and man-induced origin has been defined;
2. the phenomena and characteristics related to the site and the surrounding area have been adequately identified and taken into account;
3. the characteristics of the population in the area and the capabilities of the emergency plans for the period of operation of the NPP have been analysed;
4. the hazards associated with the site have been identified.

(2) During the identification of the hazards related to external events, the effects of the combination of these hazards with the hydrological, hydrogeological and meteorological site conditions shall be taken into account.

(3) Design measures for protection of the SSCs, engineering measures for protection of the site or administrative procedures shall be foreseen to ensure an acceptable risk associated with the identified hazards.

(4) The assessment shall confirm that all relevant factors have been taken into account in the analysis of potential radiological impacts on the population in the area surrounding the NPP in all operational states and accident conditions.

Article 62. (1) All safety functions of NPPs, including civil structures, systems and components, engineering and natural barriers, inherent safety features, as well as human activities necessary to ensure safety, shall be identified and evaluated in the safety assessment. The safety performance assessment shall cover all modes of normal operation (including start-up and shutdown), anticipated operational occurrences and accident conditions.

(2) The assessment of safety functions shall determine whether the reliability of their implementation is consistent with their importance to safety, which requires assessment of the reliability and qualification of the SSCs designated to perform these functions, their vulnerability to single failures and common cause failures, and the application of the principles of redundancy, diversity, independence, isolation and physical separation.

Article 63. (1) The assessment of the application of the defence in depth concept shall confirm that account has been taken of the possible initiating events at the respective levels of the defence in depth and that the fundamental safety functions defined in Article 48, para. 1 have been fulfilled.

(2) The assessment of the implementation of the defence in depth concept shall determine whether sufficient measures are in place to ensure:

1. detection and prevention of deviations from normal operation;
2. prevention the development of anticipated operational occurrences related to safety;
3. control and management of the accidents within the limits established by the design basis;
4. practical elimination of large or early releases of radioactive substances into the environment;
5. limiting in terms of time and place of the radiological consequences of fuel melt accidents that have not been practically eliminated.

(3) The independence of the levels of defence shall be assessed by an appropriate combination of deterministic and probabilistic safety analysis and engineering judgement. For each initiating event (starting at level 2), the SSCs required to meet the objectives of para. 2 shall be identified, and in the safety analysis, it shall be demonstrated that SSCs designed to function at a single level of defence are sufficiently independent from the SSCs provided for the other levels of the defence in depth.

(4) The safety assessment shall examine the necessary physical barriers to the release of radioactive substances and technical measures to protect the barriers and preserve their effectiveness. This process shall include an assessment of:

1. safety functions that have to protect the barriers;
2. the potential threats to their implementation;
3. the mechanisms of occurrence of these threats;
4. measures to prevent the occurrence of such mechanisms;
5. measures to mitigate the consequences of failure of a safety function;
6. assessment of the sufficiency of the barriers.

(5) The safety assessment shall address the design provisions to detect failures or bypass of each level of defence in depth. Particular attention shall be paid to internal and external events and hazards that could adversely affect more than one barrier or cause simultaneous failures of safety systems.

Article 64. The safety assessment shall be intended to determine the sufficiency of the design safety margins in normal operation, anticipated operational occurrences and accident conditions.

Article 65. (1) Assessment of the radiation protection measures shall be carried out for all operational states and accident conditions. Radiation protection measures at the operational states shall aim at achieving the following objectives:

1. limit exposure doses of the personnel and the public below the adopted statutory limits;

2. maintain radiation doses at the lowest possible level.

(2) The adequacy of the design protection measures in case of accident conditions shall be assessed in the light of the safety objectives under Article 4 related to limiting the duration and the area of implementation of public protective measures.

(3) The assessment shall confirm that sufficient technical and organisational measures have been envisaged to provide defence in depth of all sources of ionising radiation at the NPP, to monitor the radiation parameters of SSCs, the premises, the site and environment, and to control the personnel exposure.

(4) The assessment shall determine the correctness of the input data and the validity of the methodology used to calculate exposure doses of the personnel and population.

(5) Subject to assessment shall be the design provisions for sufficient space for inspection and maintenance, use of automated repair tools and non-destructive testing in high radiation areas, measures preventing spread of contamination, and the fulfilment of sanitary rules for protection of the personnel.

Article 66. (1) The following aspects shall be taken into account when assessing the loads on the SSCs due to external and internal impacts as a result of operational states and accident conditions:

1. the loads and load combinations for structures and components are commensurate with their safety class;
2. the expected frequency of occurrence of each load or load combination;
3. the stresses and deformations in the safety classified structures and components correspond to the specified loads and load combinations;
4. individual and cumulative degradation of structures and components, taking into account possible degradation mechanisms (plastic deformation, fatigue, ageing and their potential interaction).

(2) The total number of anticipated transients over the operational time and their frequency of occurrence shall be assessed on the basis of available documented data, operating experience, requirements of the operating organisation and site characteristics.

Article 67. The safety assessment shall confirm that the envisaged SSCs important to safety are with proven and conservative design that has considered the following engineering aspects:

1. the operating experience, including the results of the root causes analysis of operational events, has been taken into account, where applicable;
2. an appropriate classification and qualification system of the SSC has been developed and implemented, reflecting the importance of the performed safety functions, the severity of the consequences of their failure and the operability during anticipated operational occurrences and accident conditions;
3. the industry standards and design standards used for design, manufacturing and construction ensure:

(a) the SSC capability to perform the assigned functions, taking into account the manufacturing tolerances of the components, the accuracy of measurement of the parameters and the time delay of the control signals;

(b) the SSC capability to perform the assigned functions with low failure rate in accordance with safety analyses;

(c) the SSC capability to perform assigned functions under operational loads or loads caused by postulated initiating events;

4. the materials used are appropriate for the purpose with regard to the specified standards and the conditions that arise as a result of operational states and accident conditions considered in the design;

5. where practicable, the fail-safe principle has been applied or means of failure detection have been provided for;

6. the effects of ageing and wear out, as well as life-limiting factors such as cumulative fatigue and embrittlement, have been taken into account;

7. the necessary instructions and procedures defining the actions of personnel during normal operation, in case of deviations from the operational limits and conditions, in case of anticipated operational occurrences and in case of accidents are available and ensure an adequate level of safety.

Article 68. (1) The design solutions used in NPP evolutionary designs shall have been approved in previous applications at existing NPPs. Where this is not possible, safety shall be justified by use of results from ancillary research programmes or by the operating experience gained in other relevant applications.

(2) Based on the results and conclusions of the operating experience, the safety analysis and conducted research, the need and the benefit of improvement of the design beyond the established practice shall be reassessed. When introducing innovative or non-proven design solutions, compliance with safety requirements shall be demonstrated through a suitable ancillary programme for preliminary experimental testing and confirmation of the relevant features.

Article 69. The safety assessment shall systematically take into account human factors and man-machine interaction in the design of the NPP. For this purpose:

1. the actions assigned to operating personnel to ensure safety shall be identified and task analyses for making operational decisions shall be perform;

2. sufficient information and management tools to enable the operating staff shall be provided to:

(a) manage and control normal operation;

(b) easily assess the NPP's overall condition under normal operation, anticipated operational occurrences and accident conditions;

(c) control the state of the reactor and the state of all SSCs;

(d) identify important to safety changes in the NPP state;

(e) confirm the implementation of the intended automatic actions;

3. design working areas and the operating conditions so as to take into account the ergonomic principles and allow reliable and efficient task performance;

4. the NPP design is tolerant to human error to the extent practicable;

5. all operating actions that have to be performed for a short period of time are automated;

6. sufficient and reliable communication equipment is in place between the Main Control Room/Emergency Control Room, local control panels and Emergency Response Centre.

Article 70. (1) The safety assessment shall identify possible interactions among the NPP systems, between the NPP and other off-site industrial sites, and among the power units on the same site. Interactions among systems shall be taken into account in all operational states and accident conditions, including external hazards.

(2) The assessment shall take into account not only the physical interaction but also the

effects from operation, maintenance, failures or malfunctions of a system on the operating conditions of another system important to safety. Interactions between systems belonging to different levels of defence in depth shall be avoided by means of appropriate design solutions.

(3) The grid - NPP interaction shall be subject of evaluation in order to ensure the reliability of power supply to the systems that are important to safety.

(4) In case of extreme weather conditions or in case of an external hazard affecting more than one power unit on a site, an assessment of the capabilities of the common plant support systems, residual heat removal systems and ultimate heat sink systems to fulfil their safety function for a reasonable period of time shall be performed.

Article 71 (1) The operating organisation shall carry out an independent review of the safety assessment prior to using it or submitting it for regulatory review. The independent review shall be carried out by suitably qualified and experienced experts who have not participated in the safety assessment process. The scope of the independent review shall include:

1. a comprehensive review of the completeness of the safety assessment and the way in which it is implemented and presented;
2. a detailed overview of individual aspects of the safety assessment that have the greatest impact on safety;
3. review of the models and data used in the safety analysis to verify if they are up-to-date, representative and relevant.

(2) The design basis, the safety assessment and the technical and organisational measures ensuring the implementation of the defence in depth concept shall be documented in a preliminary, interim and final safety analysis reports (SAR) related to the authorisation process under the Act on the Safe Use of Nuclear Energy.

(3) The SAR shall confirm the fulfilment of safety requirements and shall be used to support safe operation, including during assessment of the consequences to safety from design modifications and operational practices.

(4) The operating organisation shall maintain the SAR updated in accordance with the changes made to the SSCs important to safety, the assessments and analyses carried out and the current safety requirements. The report shall be timely updated when there is new safety assessment information, including one regarding the site and NPP siting area characteristics.

Section II

Deterministic Safety Analysis

Article 72. (1) A deterministic safety analysis, including neutron, thermal-hydraulic, radiological, thermal-mechanical and strength calculations, shall be performed to identify the behaviour of the reactor installation and the spent fuel storage pool for events and states specific to the NPP.

(2) The NPP events and states defined in the design basis shall be grouped and analysed in separate categories with different acceptance criteria under Article 47 in order to demonstrate that the events with the highest frequency have no off-site radiological consequences, and events with potential consequences are very unlikely and fulfil the safety objectives under Article 4, paras 3 and 4. Depending on the expected frequency of occurrence and potential consequences, the

following categories shall be distinguished:

1. steady and transient states during normal operation;
2. anticipated operational occurrences;
3. accidents without fuel melt;
4. accidents with fuel melt.

(3) In the category of steady and transient states during normal operation, unit specific modes shall be analysed, such as: start-up; power operation; hot standby; hot shutdown; cold shutdown; refuelling; heating and cooling at maximum permissible speed; step load changes; load changes at different power ranges; reducing power from full load to house-load; boundary states defined by the operating limits and conditions.

(4) In the category of the anticipated operational occurrences, unit specific transients shall be analysed, that are typically associated with loss of operational and/or standby power supply, turbine-generator(s) trips, failures in the instrumentation and control systems, loss of one or several main coolant pumps.

(5) In the accidents without fuel melt category, individual postulated initiating events and multiple failure initiating events specific to the unit shall be analysed, including those resulting from internal and external hazards, from common cause failures and events that potentially affect all the nuclear facilities on the site.

(6) By means of the safety analysis it shall be demonstrated that accidents with nuclear fuel melt that lead to large or early releases of radioactive substances into the environment have been practically eliminated.

(7) The states, which have to be practically eliminated, shall be determined taking into account:

1. initiating events directly leading to a severe accident with large or early releases (destruction of major components such as the reactor pressure vessel);
2. dependent failures, internal and external events and hazards with the potential to cause large or early releases;
3. fuel meltdown scenarios that endanger the integrity of the containment;
4. fuel meltdown in a spent fuel storage pool (even when it is inside the containment) in view of the impossibility to practically implement accident management measures.

(8) For the accidents involving fuel meltdowns, that cannot be practically eliminated, it shall be demonstrated that the off-site radiation impact is limited and implementation of long-term measures for protection of the population are not necessary in accordance with the safety objective under Article 4, para. 4.

(9) The screening out of initiating events or accident sequences shall be justified and documented on the basis of compliance with technically valid criteria.

Article 73. (1) The deterministic safety analysis shall confirm the design basis of the NPP for the particular site and NPP siting area.

(2) The deterministic analysis shall demonstrate the possibility of controlling anticipated operational occurrences through the instrumentation and control systems, and individual initiating events through the automated action of the safety systems. When analysing such events a safety margin shall be ensured by applying a conservative approach and the following general rules:

1. the initial and boundary conditions shall be determined with reasonable conservatism;

2. the most unfavourable, independent of the initiating event, single failure shall be applied for an active or passive component performing a safety function (or a single personnel error) with the most adverse effect on the evolution of the event;

3. an additional failure (sticking) of the most effective control rod, conservatively identified with respect to the reactor scram efficiency in hot state shall be considered;

4. only operability of safety classified SSC, qualified to operate under the conditions of the particular event shall be considered;

5. the assumed efficiency of a safety classified SSC shall lead to the most adverse consequences;

6. the operability of normal operation systems shall be considered only if the effect of their operation degrades the consequences of the event;

7. any failure occurring as a result of the initiating event shall be considered as part of the postulated event;

8. operator's actions shall be assumed to be taken no earlier than 30 minutes after the beginning of the event;

9. uncertainties consider shall be accounted for by appropriate conservative assumptions, safety factors, parameters sensitivity analysis, or uncertainty assessment.

(3) Multiple failure event analyses shall be conducted to confirm the capability of the design to deal with common cause failures, to determine the need for additional measures to prevent melting of nuclear fuel, and to demonstrate sufficient margin to the occurrence of cliff-edge effects. The following general rules shall apply in the analysis of these events:

1. moderately conservative or realistic assumptions, hypotheses and arguments shall be used that are reasonable for the purpose of analysis;

2. the conclusions and results of the probabilistic safety analysis shall be applied;

3. failures and events that may occur in all operating states shall be considered;

4. the geographic and spatial location of the NPP shall be considered, the capacity and diversification of the SSCs performing safety functions, and the feasibility of implementing the actions envisaged for accident management;

5. the uncertainties and their impact on the results shall be adequately considered.

(4) Analyses of accidents with fuel melt and their radiological consequences shall be performed using a realistic approach and justified assumptions, applying the following requirements:

1. scenarios which may occur in all operational states shall be considered;

2. successful accident management actions of plant staff and emergency response teams shall be considered;

3. the conclusions and results of the probabilistic safety analysis levels 1 and 2, and of applicable experimental data shall be used;

4. the characteristics of the phenomena arising from severe accidents and the associated uncertainties shall be considered;

5. the possibilities for interaction with other nuclear facilities on the site shall be determined.

Article 74. The evolution of accident sequences shall be predicted and analysed until reaching predefined end states of the NPP for each category of events, including low power events and shutdown state of the reactor installation.

Article 75. (1) An analysis of the radiological consequences shall be performed for all operational states and accident conditions in order to assess the effectiveness of the protective barriers, determine the need to implement measures to protect the population in case of accidents, and assess the time available for their implementation.

(2) The assessment of the radiological situation for all operational states shall be carried out applying probabilistic distribution of the parameters of atmospheric dispersion that are typical for the area of the NPP siting.

(3) The assessment of the radiological consequences from accident conditions shall take into account the most adverse weather conditions typical for the area of the NPP siting. The consequences shall be determined for different periods of accidents evolution (related to the avertable dose intervention levels) considering all the mechanisms of migration of radioactive substances into the environment and all exposure pathways.

Article 76. The deterministic analysis of internal and external events and hazards shall determine the effectiveness of safety functions, taking into account event specific features. The level of detail of the analysis shall be consistent with the contribution of the event to the overall risk for the NPP, the number of physical barriers it threatens, and the possibility of the event causing simultaneous failures of safety systems.

Article 77 (1) To justify the effectiveness and the adequacy of the fire protection measures, a deterministic analysis of the fire risk shall be made by experts with qualification and experience both in the analysis of process systems and in the field of fire safety.

(2) The analysis under para. 1 shall be performed for all steady states and transients at normal operation considering the following assumptions:

1. occurrence of a single fire and its spread in each zone containing highly flammable materials;
2. dependent failures in the affected areas as a consequence of the fire;
3. combined effect from the fire and another initiating event which is likely to occur irrespective of the fire.

(3) The results of the fire hazard analysis shall indicate the possible consequences of the fire and the operation of the fire detection and fire suppression systems, including potential failures and spurious actuation.

Article 78. (1) The deterministic safety analysis shall be carried out according to methodologies developed in advance, which shall include the analysis assumptions and their basis, the individual implementation steps and justified criteria for the acceptability of the results.

(2) The analysis methodologies, the results of the safety analysis and the conclusions regarding the fulfilment of the acceptability criteria shall be documented in a verifiable and traceable way, paying special attention to the cases when engineering judgement have been used.

Section III

Probabilistic Safety Analysis

Article 79. (1) For the application of an integrated approach to the NPP safety assessment, a Probabilistic Safety Analysis (PSA) shall be performed that systematically identifies all factors which have a significant contribution to safety and the radiation risk to the population and the environment. The PSA shall be conducted at the following levels:

1. Probabilistic Safety Analysis Level 1 - it identifies the accident initiating events, and the

accident sequences, and assesses the nuclear fuel damage frequency;

2. Probabilistic Safety Analysis Level 2 - it identifies possible radioactive substances release pathways into the environment and assesses the large radioactive releases frequency;

3. Probabilistic Safety Analysis Level 3 - it assesses the risk to human health and other social risks such as soil, water and food contamination by radioactive substances and it is implemented by a decision of the Chairperson of the Nuclear Regulatory Agency.

(2) A PSA shall be implemented in order to achieve the following objectives:

1. perform a systematic analysis of the compliance with the main safety objectives and criteria, assess the frequency of occurrence of severe fuel damage and large radioactive releases into the environment, and determine the risk to the population;

2. prove, where possible, a sufficient margin until cliff-edge effects occur.

(3) The scope of PSA shall include:

1. significant sources of radioactivity (nuclear fuel in the reactor core and the spent fuel pool) and all operational states of the power unit (including operation at full power, low power and shutdown state);

2. all significant initiating events, internal hazards (such as internal fires and floods) and external events and hazards (such as seismic impacts and extreme weather conditions) identified on the basis of appropriate selection criteria;

3. all functional dependencies resulting from spatial distribution and other possible causes for common cause failures;

4. realistically modelled behaviour of the power unit, taking into account the actions of the operating personnel in accordance with the operating and emergency procedures and justified time for the performance of the functions of the systems;

5. human error analysis, taking into account the factors which can influence the performance of the personnel in all operational states and accident conditions.

6. sensitivity analysis of results and uncertainty assessment.

(4) The PSA shall be performed using unit-specific data and in accordance with a contemporary proven methodology. The data, methodology and results of the analysis shall be documented in a traceable way and kept up-to-date in accordance with the operating organisation's management system.

Article 80. (1) The PSA shall have the necessary quality and level of detail for using the results in support of the deterministic analysis when making decisions in the design process and operation of NPPs with respect to:

1. demonstrating a balanced design in which there is no an initiating event which has a disproportionate impact on the overall risk of the NPP;

2. identifying the need for changes to the design and operational practices and assessing the adequacy of the proposed safety improvement measures;

3. assessment of the operating limits and conditions, emergency procedures and severe accident management guides;

4. assessment of the significance of the operational events;

5. development and validation of personnel training programmes, including full-scope simulator training scenarios;

6. assessment of the maintenance, surveillance and testing programmes for SSCs having a significant contribution to risk.

(2) For using the PSA results, the limitations of the analysis performed shall be identified. Each specific application shall be checked against the identified limiting factors and the impact of sensitivity and uncertainty of the results.

(3) When using the PSA to assess the requirements for periodic tests and the admissible system or component downtime, the analysis shall consider all the states of that system or component and the safety function that they perform.

(4) Where the PSA results indicate importance to safety for certain components, their reliability and operability shall be ensured and documented in the safety analysis report.

(5) The external events analysis shall take into account the impact of the external hazard on the reliability of the buildings and civil structures, the robustness of the systems and components and the possibilities for human action under such conditions.

Section IV

Analysis of External and Internal Events and Hazards

Article 81. (1) For all operational states of a NPP threats from external and internal hazards shall be removed or minimized, as far as reasonably achievable. The assessment of the impact of external and internal hazards shall be considered as a part of the plant safety demonstration and shall be carried out to define and confirm the design basis, the margins to cliff-edge effects, as well as the practically possible improvements for implementation of the fundamental safety functions in case of impacts exceeding the design basis.

(2) To assess the effectiveness and sufficiency of the plant protection against external and internal events, all hazard sources that may affect safety shall be considered and assessed in the design, originating from:

1. external natural phenomena, processes and factors specific to the site and the surrounding region;
2. external human made hazards;
3. internal hazards;
4. combinations of related external and internal hazards.

Article. 82. The assessment of external and internal events and hazards includes the following methodological steps:

1. identification of all hazard sources for nuclear power plants;
2. screening based on established criteria;
3. assessment of the impact parameters of the selected external and internal events to be considered in the design;
4. analysis of external and internal events with deterministic and probabilistic methods.

Article. 83. (1) The identification of the external hazards specific to the site and the surrounding region under Art. 82, item 1 and the justification of the completeness of the compiled list shall be carried out taking into account the requirements of Chapter Three, considering at least the following hazards:

1. geological and tectonic hazards;
2. meteorological hazards;
3. hydrological hazards;
4. biological phenomena;

5. external fires caused by off-site sources;
6. commercial airplane crash;
7. explosions caused by off-site sources;
8. off-site industrial and transportation accidents;
9. electrical and electromagnetic interference.

(2) The identification process of all internal hazards that might affect the SSC important to safety, under Article 82, item 1, shall consider premises and locations where permanent or temporary hazard sources are present. The list of internal hazards shall be justified in terms of completeness and relevance to the NPP design and shall be compiled taking into account at least the following hazards:

1. fires;
2. explosions;
3. missiles;
4. pipe breaks with consequential hazardous conditions;
5. flooding;
6. collapse of structures and falling objects;
7. electrical and electromagnetic interference;
8. release of hazardous substances.

(3) The screening of external and internal hazards under Article 82, item 2 shall be based on justified criteria established with conservative assumptions. The exclusion of hazards from subsequent assessment and analysis shall be performed only if it has been demonstrated with a high degree of confidence that it is physically impossible or extremely unlikely the event to affect the plant safety. External and internal hazards which, in combination with other events, may affect the safety of the NPP shall not be excluded from the analysis.

(4) The assessment of the impact parameters of the selected external and internal events under Art.82, item 3 shall be carried out using a deterministic approach and, where applicable, with probabilistic methods in terms of the analysis of the available data and the derivation of a relationship between the event severity (intensity and duration) and the exceedance frequency. The assessment shall take into account all individual hazard sources, their direct consequences, as well as credible indirect effects.

(5) The following rules shall apply to the assessment of the parameters of external events:

1. the assessment shall be based on all available site and region data, including historical data;
2. external events and hazards whose severity changes over time (climatic and other changes) shall be taken into account;
3. justified methods and assumptions shall be used, and the natural and model uncertainties affecting the assessment results shall be evaluated;
4. for natural hazards confidence levels shall be determined for the estimates of the event severity (maximum credible event severity).

(6) External events that are considered in the design shall be grouped for analysis into the following categories:

1. design basis events, which include single external events and combinations of causally or non-causally linked phenomena and processes, the frequency of which is at least 10^{-4} per year. Where it is not possible to calculate the frequency of occurrence with an acceptable degree of

confidence, a design basis event shall be chosen and justified to ensure an equivalent level of safety;

2. extreme events that are identified, assessed and analysed to determine the margins to cliff-edge effects and the potential measures for design improvement.

(7) The impact parameters of each design basis external event shall be determined conservatively, taking into account the results of the assessment of the relevant processes and phenomena. For airplane crashes and explosion blast waves a design basis event shall be defined and analysed.

(8) The design basis events of natural origin shall be compared to relevant historical data to verify that historical extreme events are enveloped by a sufficient margin. Specifically with regard to the selection of a seismic design basis event, an acceleration for seismic loading of at least 1 m/s^2 at the natural terrain elevation shall be determined with the response spectrum being at least equal to the relevant response spectrum for conventional building construction;

(9) The assessments of the internal hazard sources, the applied methods and input data as well as the utilisation of the results shall be justified and documented. Internal hazard sources shall, as far as practicable, be removed or their impact minimised until it can be demonstrated that the most severe physically possible impact is incapable of posing a threat to SSC important to safety or that the occurrence of an event induced by an internal hazard source is extremely unlikely with a high degree of confidence.

(10) The design basis internal events shall be defined based on the unit specific assessments of the internal hazards and shall address all internal hazards that have not been removed or minimised according to the requirements of paragraph 9. The impact parameters of the design basis internal events shall be defined conservatively, taking into account the most severe physically possible impact.

(11) The external and internal design basis events as defined in Art. 82, item 4 shall be analysed using deterministic methods, supplemented where applicable by probabilistic analyses, in order to demonstrate the protection against external and internal hazards. The analyses shall be documented and kept up to date.

Article 84. (1) Based on the deterministic analysis of external and internal design basis events, reliable protection concepts shall be developed to ensure conservative performance of the fundamental safety functions for any direct and credible indirect effects of the design basis events.

(2) The protection concept for external events shall:

1. provide safety margins in the design by applying a conservative approach;

2. rely primarily on passive measures, where applicable;

3. ensure that sufficient measures to cope with a design basis accident included remain effective during and following a design basis event;

4. take into account the development of the event over time and its predictability;

5. ensure that procedures and means are available to verify the plant condition during and following accidents considered in the design;

6. consider that events could cause multiple failures in safety systems and/or their supporting systems and could simultaneously challenge several power units at a site, the site infrastructure, the regional infrastructure and external supplies;

7. ensure that sufficient resources remain available at multi-unit sites considering the use of

common equipment (including mobile) and services;

8. not have an adverse impact on the protection against design basis events of other origin.

(3) The protection concept for external events shall take into account the available means for monitoring and alert in case of external hazards, as well as the defined intervention levels for implementation of protective measures and for execution of pre-defined actions following the event.

(4) The protection against internal hazards shall be developed and implemented on the base of the defence-in-depth concept and shall include provisions to prevent the occurrence of events induced by internal hazards, to detect these events and, if relevant, to control and mitigate their consequences.

(5) The protection concept for internal hazards shall be sufficiently reliable and conservative to ensure the implementation of the fundamental safety functions for any direct and credible indirect effects of the design basis internal events. In particular, the protection concept for internal hazards shall:

1. provide safety margins in the design by applying conservatism;
2. rely primarily on passive measures, as far as reasonably practicable;
3. ensure physical separation of the individual trains of safety systems to prevent propagation of the effects of internal hazards;
4. ensure that procedures and means are available to verify the conditions of the unit during and following an impact initiated by a design basis event;
5. minimize, as far as possible, the event propagation within the site;
6. ensure that sufficient resources remain available at multi-unit sites considering the use of common equipment and services;
7. not have an adverse impact on the protection against design basis events of other origin.

(6) SSCs identified as part of the protection concept with respect to design basis events (external and internal) shall be classified in a safety class and qualified for the conditions and effects of the relevant hazards. If there is a credible combination of the hazard under consideration with another internal or external event the SCCs of the protection concept shall remain effective in this combination.

(7) Systems and equipment designed for detection and monitoring shall be part of the protection concept for internal hazards. Where possible, intervention levels shall be defined to facilitate the timely implementation of protection measures.

Article 85. (1) Accidents with nuclear fuel melt resulting from external events that lead to large or early releases of radioactive substances into the environment shall be practically eliminated by demonstrating with a high degree of confidence that the occurrence of such events is extremely unlikely.

(2) Extreme external events and phenomena that are more severe than the design basis events but cannot be practically eliminated shall be identified and analysed using a realistic approach to define practically possible improvements related to such events. When possible, the hazards severity (intensity) shall be developed as a function of the exceedance frequency or other parameters related to the event. The assessment process shall take into account the following aspects:

1. definition of the event parameters for which the implementation of the fundamental safety functions cannot be ensured;

2. demonstration of sufficient margins to avoid cliff-edge effects;
3. identification of the most resilient means for ensuring the fundamental safety functions;
4. consideration the possibility that the event could cause multiple failures in the safety systems and/or their supporting systems, to simultaneously challenge several power units at multi-unit sites, the site infrastructure, the regional infrastructure and external supplies;
5. demonstration that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;
6. conducting on-site verification and walk-downs.

(3) Internal events more severe than the design basis shall be analysed to protect the fundamental safety functions and to identify reasonably practicable improvements, unless the most severe physically possible impact has been considered for the definition of the design basis events. Analyses shall take into account credible failures of the protection measures.

(4) In accordance with the results of the analyses under paragraph 3, organisational measures shall be taken to ensure the necessary equipment and procedures, the minimum number of personnel with the required knowledge, skills and fitness for work.

Article 86. (1) In order to fulfil the safety objective under Article 4, para. 3, the consequences of a modern commercial airplane crash (as a man-made event in the category of extreme events) shall be taken into account in the design of the NPP. The analysis shall demonstrate the provision of the fundamental safety functions that render and maintain the NPP in a safe state.

(2) The design measures for the protection of the systems necessary for rendering and maintaining the NPP in a safe state following the impact of the accident, and of the civil structures in which they are located shall take into account:
1. the effects of direct and secondary impacts on their mechanical resistance;
2. the effects of vibrations on their operability;
3. the effects of ignition and explosion of aviation fuel on their integrity.

(3) Civil structures or individual parts thereof containing nuclear fuel or SSCs performing key safety functions shall be capable of preventing penetration of aviation fuel therein. Aircraft fuel-induced fires shall be considered as different types of combinations of fireball and fiery surface. Secondary fires occurring as a result of the main fire shall also be considered.

(4) Consequence analysis shall be carried by:
1. using a realistic approach, taking into account the best characteristics of input materials, realistic assumptions about failures and modern analytical methods;
2. not considering other concurrent failures of SSCs and the NPP
3. assessing the sensitivity of the results for confirming availability of a sufficient margin until cliff-edge effects occurrence.

(5) The protection concept shall take into account the impact of the event on the ability of the personnel to perform the necessary actions and the possibility of external supplies.

Article 87. (1) The following aspects shall be taken into account in the assessment of the impact parameters, in the analysis of natural and human-induced external events and in defining the protection concepts:
1. careful use of generic conditional probabilities of failure in view of differences in failure mechanisms within the same type of NPP;

2. considering the large uncertainties in the parameters of the events in the assessment of the margins to cliff-edge effects occurrence;

3. considering the impact of external events both on systems and components, and on the reliability of buildings and civil structures;

4. the possibility of limiting the impact of external events through an appropriate site layout (particularly important for multi-unit sites or power units of different generations).

(2) The results of the analysis of external and internal events shall be considered in the design, in the operational and maintenance procedures of the SSC, which ensure the implementation of the fundamental safety functions, as well as in the training programs for personnel and emergency response teams.

Section V

Periodic Safety Review

Article 88. The operating organisation shall periodically review all safety aspects of the power units and the NPP as a whole in order to determine its compliance with the licensing basis, with the current safety requirements and safety standards and with the internationally recognised good practices. The Periodic Safety Review (PSR) shall aim at identifying non-compliances, assessing their significance for safety and planning measures to address the deviations, taking into account the accumulated operational experience and the latest advances in science and technology.

Article 89. The PSR shall be carried out at least once every 10 years and shall include as a minimum the following safety factors:

1. site characteristics considered in the design and, where appropriate, their reassessment using updated methods and data;
2. NPP design;
3. current status of the SSCs important to safety;
4. qualification of SSCs;
5. ageing management;
6. deterministic safety analysis;
7. probabilistic safety analysis;
8. analysis of internal and external events and hazards;
9. safe performance indicators and operating experience assessment;
10. effectiveness of the feedback from foreign experience and scientific research;
11. organisation, management system and safety culture;
12. operating procedures and emergency procedures;
13. human factors;
14. emergency planning;
15. interaction of nuclear facilities at one site;
16. radiation impact on the personnel, population and environment.

Article 90. (1) The PSR shall be performed according to an up-to-date, systematic and documented methodology. The PSR shall be planned and carried out at the following stages:

1. preparation of the review, including identification and agreement with the Nuclear Regulatory Agency of the general methodology and the implementation plan, as well as the training of the personnel to be involved in the review;

2. carrying out the review of the safety factors in accordance with the defined methodology;
3. analysis of the review outcomes, performance of global assessment and development of an integrated programme of measures for safety improvement.

(2) The general methodology for the implementation of the PSR shall contain the overall framework for the review, organised in the following main parts:
1. general methodological requirements that include the approach and the scope of review, the applicable requirements and standards, the methodologies for categorising non-compliances, for performing the global assessment and developing the integrated programme for implementation of measures to improve safety;
2. specific methodologies for reviewing the individual safety factors and assessing compliance with the applicable legislative, regulatory and licensing requirements and with the current safety standards and good practices;
3. a plan for management of the activities and a schedule for their implementation.

(3) When performing the summary assessment, all positive and negative findings shall be considered, and their cumulative effect on safety shall be determined to identify the practicable improvements taking into account the entire lifetime of the power unit.

(4) The results of the global assessment shall demonstrate full confidence in the safe operation of the power unit during the period until the next PSR, and indicate any problems that could limit the future safe operation of the unit and explain how these problems will be managed.

(5) The PSR report and the draft integrated programme to improve safety shall be submitted to the Nuclear Regulatory Agency in accordance with Article 37c of the Act on the Safe Use of the Nuclear Energy.

Chapter Six

REQUIREMENTS FOR DESIGN OF NPP AND PLANT SYSTEMS

Section I

General Requirements to NPP

Article 91. (1) SSCs important to safety shall ensure safe shutdown of the reactor and maintaining it subcritical, cooling of the reactor coolant boundary, residual heat removal, confinement of the radioactive substances within the prescribed limits for the operational states where applicable, for accidents without fuel melt, and for the impacts caused by internal and external events considered in the design.

(2) In accidents with fuel melt, SSCs important to safety shall be able to ensure the performance of the "confinement of radioactive substances" function; to that end they shall ensure and maintain the function of "residual heat removal" from the damaged fuel, and containment cooling. The spent fuel pools shall be maintained subcritical at all times, and the reactor core - in the long-term aspect.

(3) For the performance (or restoration) of the required safety functions during accident with fuel melt, both stationary systems and mobile equipment located on the NPP site can be

considered in the design.

Article 92. (1) SSCs important to safety, their structure, layout and operational condition shall provide possibility for testing, maintenance, repair, inspection and control, over the whole plant operating lifetime, without significant reduction in their functional availability. When SSCs important to safety cannot undergo in-service testing and inspection to a sufficient extent to detect potential failures, their reliability shall be ensured by another proven alternative method, or the design shall conservatively consider a higher failure rate.

(2) SSCs important to safety shall be designed, located and protected in a way that leads to reducing the rate and consequences of fire. The design options shall ensure the performance and maintaining of the main safety functions and control of the power unit condition.

Article 93. The safety systems shall operate so that any initiated actuation shall lead to complete fulfilment of the safety functions. Resetting of the safety systems to their initial state shall require consecutive actions of the operating personnel.

Article 94. (1) The NPP site shall be provided with facilities for personnel protection in case of an accident. They shall be located, protected and equipped so as to ensure habitability and personnel protection over a specified period of time.

(2) The design shall foresee at least one emergency response centre where the technical support teams for the operating personnel during accident shall operate, and where the emergency recovery activities on the NPP site shall be coordinated.

(3) The emergency response centre shall be physically separated from the main control room (MCR) and shall be suitably laid out, designed and protected, so as to preserve its functionality, habitability, and efficiency under the accident conditions that are to be managed (inclusive of severe accident and extreme external events of natural origin).

(4) The centre shall receive information on the power unit's status during the phases of accident progression and on the radiological conditions at the NPP site and its surroundings.

(5) The centre shall be equipped with devices and systems for communication with the main and supplementary control rooms, with the authorities of local self-government and the executive power relevant for accident management. The communication systems and devices shall be maintained available, periodically tested and the documentation shall be kept up-to-date.

Article 95. Prior to the beginning of a power unit commissioning, verification shall be made on the hardware and software, programmes and methodologies which are needed for:

1. functional capability inspection of the SSCs (including those inside the reactor) and their replacement after their design lifetime;
2. functional tests of the systems intended to prove their design characteristics;
3. verification of the signal sequences and actuation of systems and components, including of the emergency power supply;
4. in-service inspection of the base metal and weldings of the structures and pipelines;
5. verification of the metrological characteristics of the measuring channels for compliance with the design requirements.

Section II

Reactor Core Design and Features

Article 96. (1) The reactor core and associated reactor coolant system, and reactor safety systems shall be designed with appropriate safety margins to ensure that the specified design limits for fuel damage are not exceeded in all operational states and accidents without fuel melt with account taken of:

1. design operational states and their course;
2. thermal, mechanical and irradiation degradation of the core components;
3. physical-chemical interaction of the core materials;
4. limiting values of thermal hydraulic parameters;
5. vibrations and thermal cycles, material fatigue and ageing;
6. impact of coolant impurities and radioactive fission products on the fuel claddings corrosion;
7. irradiation and other impacts that deteriorate the mechanical characteristics of the core materials and fuel cladding integrity.

(2) The design shall specify the limits for damage of the fuel elements (in terms of amount and degree) and the associated coolant radioactivity according to reference isotopes.

Article 97. To ensure safe shutdown of the reactor, to maintain the reactor subcritical and to ensure core cooling, the reactor core and associated core internals located within the RPV shall be designed and installed in such a way as to withstand the static and dynamic loads expected in all operational states, accidents without fuel melt, and external events considered in the design.

Article 98. (1) The reactor core and its components that affect reactivity shall be designed in a way that any reactivity change caused by the control rods as well as reactivity effects shall not lead to fuel damage that exceeds the specified design limits and shall not cause any damage to reactor coolant pressure boundary in all operational states and accidents without fuel melt.

(2) The design shall prove that in all accidents with fast insertion of positive reactivity, specific energy threshold for fuel damage is not exceeded at any moment of the fuel cycle, and fuel melt is excluded.

(3) Accidents with fast insertion of positive reactivity, resulting in early or large radioactive releases to the environment shall be practically eliminated.

(4) For all accidents without fuel melt, changes in core geometry shall be limited so as to ensure conditions for long-term fuel cooling, insertion of control rods and ensuring core subcriticality for a long period of time.

(5) Reactivity feedback, determined by the parameters that affect reactivity, shall be negative in all possible critical states of the reactor, in all operational states and accidents without fuel melt.

(6) The design shall ensure that after anticipated operational occurrences and accidents without fuel melt, the core shall remain subcritical.

Article 99. (1) The neutron flux distribution shall be stable and shall require minimum intervention for maintaining the shape, level and stability of the neutron flux within defined design limits, for all core states, including after reactor shutdown, during and after core refuelling, and anticipated operational occurrences.

(2) The design shall provide for adequate means to measure neutron flux distribution so that the design limits in any axial or radial core part are not exceeded during reactor installation power operation.

Article 100. The reactor core and associated coolant system and safety systems shall be designed to enable adequate inspection and testing throughout the service lifetime of the plant.

Article 101. The characteristics of nuclear fuel, reactor structures and reactor coolant system components (including the coolant clean up system) and considering the operation of the other systems shall exclude accumulation of critical mass in accidents with fuel melt.

Article 102. The fuel elements and assemblies, with account taken of the uncertainties in data, calculations and assumptions in fabrication, shall be designed to withstand irradiation and the reactor core conditions in combination with all degradation processes that can occur in all operational states, such as:

1. differential expansion and deformation;
2. external pressure of the coolant;
3. additional internal pressure in the fuel element due to fission products;
4. irradiation of fuel and other materials in the fuel assembly;
5. variation in pressure and temperatures resulting from variations in power;
6. chemical effects;
7. static and dynamic loads, including flow induced vibrations and mechanical vibrations;
8. variations in heat transfer performance that could be a result of distortions or chemical effects.

Section III

Reactor Shutdown Systems

Article 103. (1) The design shall provide for means to control reactivity, that are capable of ensuring reactor shutdown in all operational states and accident conditions, and of maintaining core subcriticality even at the maximum value of the effective neutron multiplication factor.

(2) The effectiveness, speed of action of reactivity control equipment, and core criticality margins shall be such that the design limits for fuel damage are not exceeded.

Article 104. (1) The means for reactivity control shall consist of at least two independent and diverse systems; each one of these systems shall be capable, on its own, of maintaining the reactor subcritical by an adequate margin and with high reliability, considering the single failure principle or human error.

(2) The systems shall be adequate to prevent any foreseeable increase in reactivity leading to unintentional criticality during the reactor shutdown, or during core refuelling operations or other routine or non-routine operations in the reactor shutdown state.

(3) The system design shall consider possible internal or external system failures that could render part of reactivity control means inoperative, or that could result in a common cause failure of the whole system.

Article 105. (1) At least one of the systems shall perform reactor emergency shutdown functions and shall maintain core subcritical with sufficient margin, taking into account failure to activate the most effective control rod and maximum value of the effective neutron multiplication factor.

(2) In case the effectiveness of the emergency shutdown system is not sufficient to maintain the core subcritical for a long time, provisions shall be made for automatic actuation of another reactor shutdown system with adequate effectiveness to maintain the core subcritical, considering possible reactivity release.

(3) The reactor emergency shutdown system shall have at least two independent groups of rods. The control rods shall be actuated at any intermediate or operating position.

(4) Any possibility for positive reactivity insertion by means of reactivity control shall be excluded by technical means if the emergency shutdown system rods have not been inserted in operating position.

Article 106. (1) All emergency shutdown system rods shall have intermediate position indicators, end position annunciators and limit switches (end breakers), actuated where practicable directly by the control rod. The other reactivity control means shall be equipped with position indicators as a minimum.

(2) The design shall determine tests required to confirm the operability of the reactivity control means, envisaged for each operational state of the NPP.

Section IV

Instrumentation and Control Systems

Article 107. (1) The instrumentation and control systems shall measure the values of key parameters that could impact the chain fission reaction, the nuclear fuel integrity in the core and in the spent fuel pools, the reactor coolant system and the containment in all operational states and accident conditions.

(2) The instrumentation and recording equipment shall ensure sufficient information for monitoring the status of essential equipment and the course of accidents, for determining the amounts and locations of releases of radioactive substances in the environment, and for post-accident analysis.

(3) The changes in normal operation conditions which may affect safety, shall be accompanied by audible and visible indication in the main control room.

(4) The control systems shall be adequate for measuring the NPP parameters and shall be qualified for environmental conditions in normal operation, anticipated operational occurrences and accident conditions.

Article 108. (1) The normal operation control systems shall reliably maintain and control variations of process parameters within the operational limits.

(2) Control signals of technological systems and components important to safety, formed by the control systems or by the MCR remote control switches, shall be automatically recorded.

Article 109. (1) The control safety systems shall detect indications of potential accidents and shall automatically actuate the safety systems necessary for achieving and maintaining safe plant conditions.

(2) The control safety systems shall be designed so that:

1. a single failure does not result in loss of control function;
2. unavailability of any component or train does not result in loss of the required minimum redundancy;
3. they shall be capable of overriding unsafe impacts of the control systems for normal operation;
4. they shall prevent operator actions that could compromise the efficiency of the protection system in operational states and in accident conditions, but shall not counteract correct operator actions in accident conditions;
5. in the event of a failure of a control safety system they shall achieve safe plant conditions by implementing the fail-safe principle;
6. they shall actuate the safety systems so that operator action is not necessary within a certain period of time from the onset of anticipated operational occurrences or accident conditions;
7. they shall make relevant information available to the operator for monitoring the effects of the automatic actions;
8. they shall ensure continuous automatic diagnostics of the systems operability and diagnostics of the process components whose failures are initiating events for accidents.

Article 110. (1) The instrumentation and control systems for SSCs important to safety shall be designed with high functional reliability and possibility for periodic testing commensurate to the safety function to be performed.

(2) Principles such as functional diversity and diversity in component design, independence and redundancy shall be used in the design to the extent practicable to prevent the loss of safety functions.

(3) Periodic testing shall determine the functionality of sensors, input signals, logics, technical controls and technical and computer means of recording.

(4) When a safety system has to be taken out of service for periodic testing during power operation, provision shall be made in the design for the clear indication of any system bypass that is necessary for the duration of the testing or maintenance activities.

Article 111. (1) When designing computer based instrumentation and control systems for SSCs important to safety, appropriate standards and practices shall be identified and implemented to ensure high quality of development and testing of computer hardware and software throughout the service life of the systems, in particular when developing the software.

(2) The entire development process, including control, testing and introducing of design changes, shall be systematically documented so as to allow traceability and reviewing.

(3) Equipment reliability assessment shall be undertaken by experts who are independent of the designer and the supplier. Where safety functions are essential for achieving and maintaining safe plant conditions, and the necessary high reliability of the hardware and software cannot be demonstrated with a high level of confidence, diverse means of ensuring fulfilment of the safety functions shall be provided.

(4) Common cause failures deriving from software shall be taken into consideration for the assessment of the safety functions performance.

(5) Protection shall be provided against accidental disruption of, or deliberate interference with system operation.

Article 112. (1) Interference between control safety systems and control systems for normal operation shall be prevented by means of separation, by avoiding common connections or by functional independence.

(2) When common signals are used, adequate electrical separation (decoupling) shall be ensured and signals shall be classified as part of the control safety system.

(3) Instrumentation and control systems for the hardware and software intended for multiple failure protection and for containment protection in accidents with fuel melt, shall be separated and independent from the other instrumentation and control systems as far as practicable. Active system components shall be redundant.

Article 113. (1) The MCR shall provide the opportunity to operate safely the nuclear power plant in all operational states, both automatically or manually; it shall be possible to take measures in the MCR to maintain the plant in a safe state or to bring it back to a safe state after anticipated operational occurrences and accident conditions.

(2) The layout of instrumentation and control devices and the way of presenting the information shall be such that the operating crew at the MCR are able to clearly and quickly identify the plant status and behaviour, adherence to the operational limits and conditions, identification and diagnosis of the safety system automatic actuation and functioning, and accident management systems functioning.

(3) The design shall identify those events, both internal and external to the MCR, that could directly challenge its continued operation, and shall provide for reasonably practicable measures to contain the consequences of such events.

(4) The MCR operating personnel shall be protected by provision of barriers against high radiation levels resulting from accident conditions, releases of radioactive substances, fire, or explosive or toxic substances.

Article 114. (1) The supplementary control room (SCR) shall enable the following functions:

1. control of the safety systems;
2. rendering and maintaining the reactor subcritical;
3. heat removal from the reactor coolant system and from the spent fuel pool;
4. control of the status of the reactor installation and of the spent fuel pool.

(2) The SCR shall be physically, electrically and functionally separate from the MCR. Any possibility of parallel actuation of control components from the MCR and the SCR, shall be eliminated by technical means. Appropriate measures shall be taken to eliminate any possibility for failure of the control circuits of both MCR and SCR due to a common cause, in postulated initiating events.

(3) The SCR shall be designed to protect the personnel in all conditions resulting from internal and external events and in accident conditions.

Section V

Reactor Coolant System

Article 115. (1) The reactor coolant system and its associated safety systems shall be designed with sufficient margins to ensure that design limits of the reactor coolant pressure boundary are not exceeded in all operational states.

(2) Provisions shall be made in the design for pressure relief devices to protect the pressure boundary of the reactor coolant systems against overpressure and to prevent unacceptable releases of radioactive substances to the environment in all operational states and accidents without fuel melt.

(3) The design of pipework connected to the pressure boundary of the reactor coolant systems shall provide for isolation devices to limit any leak of primary coolant and to prevent the loss of coolant through interfacing systems.

Article 116. (1) Components, pipelines and supporting structures of the reactor coolant system shall withstand all static and dynamic loads and temperature effects to any of their component in all operational states and accidents without fuel melt.

(2) Materials to be used for fabrication of the components of the reactor coolant system shall be selected so as to minimise their activation and the probability of crack propagation and neutron embrittlement, with account taken of the expected degradation of their characteristics at the end of their lifetime under the effects of erosion, creep, fatigue and chemical impacts.

(3) The design shall provide for at least three independent diverse systems for coolant boundary system early leak detection.

Article 117. The reactor pressure vessel and pressure tubes shall be designed and manufactured to ensure the highest quality with respect to material selection, design standards, capability of inspection and fabrication.

Article 118. The design of the components contained inside the reactor coolant pressure boundary shall be designed such as to minimise the likelihood of failure and associated consequential failure and damage to other items of the primary coolant system in all operational states and in accidents without fuel melt.

Article 119. (1) The components of the reactor coolant pressure boundary shall be designed, manufactured and situated in a way allowing periodical inspections and tests to be carried out throughout the service lifetime of the plant.

(2) The implementation of a material surveillance programme for the reactor coolant pressure boundary shall ensure control of the effects of irradiation, stress corrosion cracking, embrittlement, and ageing of structural materials, particularly in locations of high irradiation, and other factors.

Article 120. The design shall provide for means for controlling the inventory, temperature and pressure of the reactor coolant which have sufficient capacity to ensure that the specified design limits are not exceeded in any operational state, with due account taken of volumetric changes and leakage.

Article 121. (1) The design shall provide for systems to cleanup the reactor coolant from radioactive substances, including activated corrosion products and fission products.

(2) The capacity of the necessary systems under para. 1 shall be based on the fuel design limits on permissible leakage with a conservative margin to ensure that the coolant activity is as low as reasonably practicable.

Section VI

Systems for Core Cooling and Heat Transfer to an Ultimate Heat Sink

Article 122. To perform the fundamental safety function of "heat transfer from the core to an ultimate heat sink" after scheduled shutdown of the reactor, during and after anticipated operational occurrences and in accident conditions, the NPP design shall provide for the following:

1. reactor core residual heat removal (transfer) systems;
2. emergency core cooling systems;
3. systems for heat transfer (removal) to an ultimate heat sink.

Article 123. The systems for reactor core residual heat removal in shutdown state of the reactor plant shall be designed with sufficient capacity to ensure the design limits are not exceeded for the fuel, reactor coolant boundary system and the structures important to safety.

Article 124. (1) The emergency core cooling systems shall ensure restoration and maintaining of the nuclear fuel cooling in accident conditions, including when the coolant boundary system integrity is breached. In order to achieve reliable safety function performance the systems design shall consider loss of off-site power and single failure, and ensure sufficient redundancy, diversity and independence.

(2) The effectiveness of the emergency core cooling systems together with the technical provisions for leak detection of the reactor coolant system, the reactor inherent safety features, and the isolation capabilities, shall be sufficient to fulfil the following requirements:

1. meet the criteria for fuel cladding leak-tightness and fuel integrity;
2. maintain within acceptable limits possible chemical reactions of interaction of the substances in the core;
3. possible changes of the internal core geometry shall allow for moving of the control rods and sufficient coolant flow to maintain adequate core cooling;
4. provisions shall be made to ensure conditions for the required time of core cooling, including sufficient coolant inventory.

(3) The actuation and operation of the emergency core cooling systems shall not lead to impairment or loss of functions of other systems. Therefore, measures shall be designed to prevent:

1. the possibility for reactor criticality;
2. violation of the protection criteria of reactor coolant pressure boundary, specified in the design limits.

(4) The design shall provide for opportunities for long-term heat removal in accidents with core melt.

(5) The emergency core cooling system design shall permit periodical inspections of the components and periodical testing of the systems to confirm:

1. the integrity of the structure and the tightness of system components;
2. the functional capability and the operational characteristics of active system components;
3. the functional capability of the system as a whole under specified operational states.

Article 125. (1) The systems and means for transfer of heat from the reactor core and SSCs important to safety to an ultimate heat sink shall perform their function in all operational states and accident conditions with a very high degree of reliability. The reliability to perform this function shall be ensured by simultaneously applying the principles of redundancy, diversity, physical separation and functional separation under the assumption for loss of the main ultimate heat sink or the access to it.

(2) The main and alternative systems and means for heat transfer shall be capable of performing their function entirely independent from one another in accident conditions.

(3) The choice of main and alternative ultimate heat sink and the design of heat transfer systems shall be made with account taken of the site-specific natural phenomena and man-induced events with the purpose of ensuring the implementation of the function under conditions of extreme external events.

Section VII

Structures and Systems Performing Confinement Safety Function

Article 126. (1) The reactor installation design shall provide for a containment structure and a set of systems and means to ensure the performance of the fundamental safety function of "confinement of releases of radioactive substances to the environment" in accordance with the authorised limits in all operational states and in accident conditions. Besides, the containment shall protect the reactor installation from extreme external events.

(2) In order to perform the confinement function the design shall provide for:
1. leaktight structure (containment);
2. systems and means for control of containment environment parameters;
3. systems and means for containment structure isolation;
4. Systems and means for reducing the concentration of radioactive fission products, hydrogen and other substances that could be released in the containment atmosphere during and after accident conditions, including in accidents with fuel melt.

Article 127. (1) The containment structure and its components, including hermetic access doors, penetrations and isolation devices, shall be designed with sufficient safety margins to withstand static and dynamic loads of internal overpressure and underpressure, temperatures, missiles impact, reaction forces, and of other potential energy sources anticipated to arise as a result of accident conditions, including in accidents with fuel melt. The maximum strength of the containment and its components shall be determined with account being taken of:
1. impacts of extreme external events, including modern passenger aircraft crash;
2. combination of impacts generated by rupture of the reactor coolant boundary pipe with maximum diameter, and a safe shutdown earthquake.

(2) The design shall include means for containment structure surveillance in all operational states and accident conditions.

Article 128. (1) The containment structure and its components shall be designed and constructed to ensure structural integrity testing during commissioning and performing of periodic leaktightness tests over the plant lifetime. The design shall specify requirements to tests and the respective methods and means to conduct the tests. The components located inside the containment shall retain their functional capability after the tests have been conducted.

(2) The design shall ensure possibilities to control containment radioactive leakages in case of accidents with fuel melt.

Article 129. (1) The number of penetrations through the containment structure shall be as low as possible. The penetrations design shall meet the same requirements as applied to the containment structure with account taken of possible mechanical, thermal, and chemical effects.

(2) The elastic components of containment penetrations shall be designed to allow individual leak testing, independent of the containment leak rate detection (integral test).

Article 130. (1) To ensure reliable isolation of the containment during accidents, each line that penetrates the containment (whether part of the reactor coolant pressure boundary or directly connected to the containment atmosphere) shall be reliably isolated by at least two isolation valves having independent automatic control, arranged in a series and located as close as practicable to the containment structure outside and inside.

(2) Each line that penetrates the containment and is neither directly connected to the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall be reliably isolated by at least one containment isolation valve located outside the containment and as close to the containment as is practicable.

(3) Isolation valves shall retain their operability in all accident conditions.

Article 131. (1) To secure personnel access to the containment premises, provisions shall be made of airlocks and doors with interlocks to ensure that at least one of the doors is closed during all operational states and accident conditions. The same requirements shall be applied when transporting components through the containment structure.

(2) The design shall consider the capabilities to ensure the functionality of containment airlocks in all accident conditions.

Article 132. The containment design shall include measures and hardware devices to ensure sufficiently low pressure difference between the separate internal compartments not to challenge the integrity of the pressure bearing structure or of other systems with confinement functions, taking into account the pressure and the possible effects in all accident conditions.

Article 133. (1) The design shall provide for pressure and temperature control systems within the containment for any accidental release of high energy fluids. These systems shall be sufficiently efficient and reliable (including degree of redundancy, independence and diversity) to perform their functions over a long period of time following an accident with fuel melt, taking account of the effects of non-condensable gases.

(2) If a filter venting system is provided for pressure control, the system actuation parameters values and the containment strength shall be dimensioned in a way that the system is not required in the early phase of the accident, taking account of the pressure of the non-condensable gases accumulated in the containment.

(3) When a filter venting system is included in the design, it shall not be designed as the principal means of removing the containment residual heat generated as a result of an accident with fuel melt.

Article 134. (1) The NPP design shall provide for systems to control and cleanup the containment atmosphere, and systems to control the concentration and control of fission radioactive products, oxygen, hydrogen, and other substances, which could be released in the containment during accident conditions.

(2) The systems and provisions for hydrogen concentration management during accident with fuel melt shall have sufficient capacity to prevent the destruction of the containment as a result of an explosion of combustible gases. Loads resulting from instant burning and detonation of combustible gases that could threaten the containment integrity and leaktightness shall be practically eliminated.

(3) The systems for containment atmosphere cleaning up shall have adequate components' reliability and redundancy to ensure the required efficiency of the system on the assumption of a single failure.

Article 135. The selection of coatings and thermal insulation, and methods for their application on SSCs inside the containment, shall ensure the fulfilment of their safety functions and shall minimise interference with other safety functions in the event of degradation of their integrity.

Section VIII

Supporting Safety Systems

Article 136. The NPP design shall provide for supporting safety systems fulfilling auxiliary services of supplying safety systems with fluids and energy, and maintaining their operational conditions over a justified period of time in all operational states and accident conditions.

Article 137. (1) The supporting safety systems shall be designed with adequate components' reliability and redundancy to ensure the necessary effectiveness on the assumption of a single failure, independent of the initial state.

(2) The degree of functional reliability of the supporting systems shall be sufficient to meet the required reliability criteria of the respective safety system.

(3) The systems design shall provide a possibility for testing of their functional capability and for failure indication.

(4) The fulfilment of supporting functions shall have priority over the supporting systems own protections, if this will not aggravate the safety consequences. The design shall specify the uninterruptible own protections of the components of the supporting safety systems.

Article 138. (1) The design shall include emergency power supply systems for the SSCs important to safety, in the event of loss of off-site power, capable of performing their functions in all operational states and accident conditions, with the assumption of a single failure and common cause failure.

(2) The design solutions against common cause failures, including those resulting from external events, shall include an alternative (diverse) emergency source of alternate power supply

(stationary or mobile), qualified to operate under conditions of extreme external events of natural origin.

(3) To ensure the required reliability of the emergency power supply for a long period of time (not less than 72 hours) following accident conditions, the design shall provide for:

1. sufficient capacity of the DC power supply sources or a possibility for their recharging;
2. sufficient stock of fuels and consumables for the emergency and alternative AC power supply sources.

Article 139. (1) The design shall make provisions for fire detection and fire extinguishing systems to prevent fire-induced common cause failures in the safety systems and to automatically fulfil the specified functions. The fire extinguishing systems shall be capable of automatic actuation.

(2) The fire safety measures shall ensure defence in depth by preventing fire, fast detection and extinguishing of any fire, ensuring the structure stability in the event of fire, limiting the fire and smoke propagation and the fire consequences, creating conditions for occupants evacuation and for safety of the rescue teams. To achieve those objectives:

1. the building structures shall be conservatively designed as fire resistant with consideration of internal and external fires;
2. the internal structures and components shall be of fire response class A1 or A2;
3. the fire load shall be kept at the practically achievable minimum;
4. the power unit shall be subdivided into fire protection sections through fire protection barriers with adequate fire resistance, capable of preventing spread of smoke and heat from fires considered in the design;
5. the characteristics of the fire detection and fire extinguishing systems (reliability, independence, capacity and qualification) shall be selected with consideration of the fire hazard resulting from the analysis, required under Article 77;
6. the required protected areas, safe areas, evacuation routes and fire evacuation exits shall be provided;
7. conditions for urgent fire extinguishing shall be ensured: external and internal water supply for fire extinguishing, fire routes and access of the rescue teams.

(3) When air ducts cross fire protection barriers, fire dampers with fire resistance corresponding to the fire resistance of the barrier being crossed shall be provided at the crossing points or the entire air ducts shall be provided with fire resistance classified inside-out and outside-in corresponding to the fire resistance of the barrier being crossed in accordance with the fire hazard analyses under Article 77. The insulation of the air ducts shall be provided with products with a class of reaction to fire not lower than A2.

Section IX

Other SSCs important to safety

Article 140. (1) The design of the steam generators and steam lines system (steam supply system), the steam generators feedwater system and the turbine and generators system for the nuclear power plant shall be such as to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded in all operational states or in accident conditions.

(2) The design of the steam supply system shall provide for appropriately rated and qualified

steam isolation valves capable of closing under the specified conditions in operational states and in accident conditions.

(3) The steam supply system and the steam generators feedwater system shall be of sufficient capacity and shall be designed to prevent anticipated operational occurrences from escalating to accident conditions.

(4) The design of the turbine and generators system shall be provided with appropriate protection such as overspeed protection and vibration protection, and measures shall be taken against possible impacts of missiles generated by component destruction on SSCs important to safety.

Article 141. Overhead transport and lifting equipment that is used for the SSCs important to safety, or for transport of heavy loads in close proximity to such SSCs shall be designed so as to:

1. prevent the movement of loads exceeding the design load capacity;
2. prevent, using conservative design measures, any unintentional dropping of loads that could affect SSCs important to safety;
3. be used only in specified operational states by means of safety interlocks;
4. be seismically qualified.

Article 142. The design basis for any compressed air system that serves an item important to safety shall specify the quality, flow rate and cleanness of the air to be provided.

Article 143. Adequate protection against lightnings of all buildings and working areas on the NPP site shall be ensured in all operational states and accident conditions.

Section X

District Heating System

Article 144. (1) When the NPP is coupled with heat utilisation unit for district heating, the design shall include measures for practical elimination of transport of radionuclides from the NPP systems to the supply system coolant of the district-heating unit in all operational states and accident conditions.

(2) The measures for preventing radioactive contamination of the supply system coolant shall be identified considering the following requirements:

1. The plant heat shall be transferred to the intermediate coolant through leak tight heat-exchangers;
2. The heat-up of the supply system coolant by the intermediate coolant shall be performed by means of heat-exchangers;
3. The intermediate coolant pressure shall be lower than the supply system coolant pressure.

Article 145. (1) Means for isolation of the supply system coolant from the intermediate coolant heat exchanger shall be provided to prevent accident-related ingress of radioactive substances in the intermediate coolant.

(2) The heat exchangers used for heating of the supply system coolant shall be located on the NPP site.

Section XI

Radioactive Waste Management

Article 146. (1) The radioactive waste (RAW) management systems shall be designed based on analysis and assessment of the composition and quantities of solid and liquid RAW and the gaseous radioactive substances generated in all operational states and accident conditions.

(2) The systems for management of liquid and gaseous radioactive discharges to the environment shall be designed so that their quantities and concentrations are kept as low as reasonably achievable in all operational states and within the specified dose limits for the occupational exposure of the personnel and the dose limits for the population.

Article 147. (1) The NPP design shall include systems for preliminary treatment and temporary storage of liquid RAW in a condition suitable for transportation and further treatment.

(2) The NPP design shall include facilities for temporary storage of solid RAW, equipped with automatic means for handling.

(3) The RAW storage compartments shall be watertight and provided with systems for ventilation, decontamination, fire detection and fire extinguishing.

(4) The design shall specify the way for management of large quantities of liquid RAW generated in accident conditions.

Article 148. (1) The NPP design shall ensure maintaining of the volume and activity of the generated liquid RAW as low as reasonably achievable through the use of effective cleanup systems and multiple use of radioactive fluids, leakage prevention in systems containing radioactive fluids, and reduction of the frequency of events that require significant decontamination measures.

(2) The plant RAW management systems shall be designed with account taken of the requirements to the subsequent stages of their safe management.

Section XII

Handling and Storage of Nuclear Fuel

Article 149. The NPP design shall include SSCs for handling and storage of non-irradiated fuel designed to:

1. prevent criticality by a sufficient margin, even under the most adverse conditions, by ensuring appropriate physical means or processes, such as geometrically safe configurations, and characteristics of the components and medium;
2. permit appropriate fuel acceptance test, maintenance, periodic inspection and testing of components important to safety;
3. ensure control of the storage conditions;
4. minimise the possibility of damage or unauthorised access to the nuclear fuel;
5. prevent fuel assembly drop during transportation;
6. prevent inadvertent dropping of heavy objects upon the fuel assemblies.

Article 150. (1) The design of SSCs used for spent fuel storage and handling shall be aimed at practical elimination of accidents with large or early radioactive release, including accidents with fuel melt.

(2) In addition to the requirements applicable to the fresh fuel storage, the spent fuel storage and handling SSCs design shall take into consideration the following additional measures and means:

1. Availability of sufficiently reliable and, if needed, diverse, residual heat removal systems and means in all operational states and accident conditions.
2. Measures to prevent unacceptable handling stresses on the fuel assemblies;
3. Means for safe storage of untight or damaged fuel assemblies or fuel elements;
4. Systems for local ventilation and other means for radiation protection;
5. Means for identification of the fuel assemblies.

(3) For reactors using a water pool system for storage of irradiated fuel, the design shall provide for the following:

1. Means for monitoring and controlling the water parameters in the pool (including level, chemistry and activity), as well as means for leak detection;
2. Means and measures for maintaining the integrity of the pool's civil structure, including against extreme external events;
3. Measures to prevent emptying the pool as a result of syphon effect in the event of a pipe break;
4. Means to control the concentration of the soluble neutron absorber.

(4) The design and the safety assessment shall confirm capacity of the structures for storing of irradiated fuel and consider the capability at any time to completely remove the fuel from the reactor core.

Section XIII

Radiation Protection

Article 151. (1) To ensure radiation protection, the NPP design shall identify all real and potential sources of ionising radiation and shall provide measures for ensuring the necessary technical and administrative control over their use.

(2) (amended - SG No. 37/year 2018) The requirements with regard to the classification of zones and premises, radiation monitoring, the personal protective equipment and the access control are established by a different regulation under Article 26, para 2 of the SUNEA.

Article 152. To keep the exposure of personnel and public as low as reasonably achievable during the plant operation, the design of the reactor coolant system shall arrange for:

1. Use of structural materials with minimum content of chemical elements with high activation cross-section and producing long-lived radioactive corrosion products;
2. Coolant purification from fission and corrosion radioactive products;
3. Water chemistry control;
4. Minimum length of the pipelines with a minimum number of isolation valves and connections;
5. Leak-tightness testing of operating components;
6. Decontamination of SSCs outer and inner surfaces;
7. Prevention of uncontrolled radioactive leaks in the NPP premises.

Article 153. (1) Shielding shall be designed in a conservative way, taking into account the build-up of radionuclides over the plant lifetime, and the potential loss of shielding efficiency due to the effects of interactions of neutron and gamma rays with the shielding.

(2) The buildings, premises and components, which may be contaminated with radioactive substances, shall be designed in a way that allows easy decontamination by chemical or mechanical means.

(3) The personnel access to premises of high contamination level shall be controlled by means of locking devices with interlocks and indication for actuation and for inoperability.

Article 154. (1) Shielding shall be designed in a conservative way, taking into account the build-up of radionuclides over the plant lifetime, the potential loss of shielding effectiveness due to the effects of interactions of neutron and gamma rays with the shielding, due to reactions with other materials, decontamination solutions, and the expected temperature effects.

(2) The choice of shielding materials shall be made on the basis of the nature of the ionizing radiation, shielding, mechanical and other properties of materials and spatial limitations.

Article 155. (1) The plant design shall provide ventilation systems to:
1. prevent dispersion of gaseous radioactive substances in the plant premises;
2. reduce and maintain airborne concentrations in plant premises below the authorised limits and as low as reasonably achievable in all operational states and design basis accidents;
3. ventilate the air in premises containing inert or noxious gases.

(2) The ventilation system design shall take into account the following factors:
1. Mechanisms of thermal and mechanical mixing;
2. Limited efficiency of dilution in reducing airborne contamination;
3. Extracting the air from areas of potential contamination at points near the source of contamination;
4. Ensuring adequate distance between exhaust air discharge points and the air intake points;
5. Providing a higher pressure in the less contaminated zones in comparison with the zones of higher contamination level;
6. Preventing the spread of flue gases and combustion products to neighbouring premises.

Article 156. (1) The design shall provide for ventilation and air cleaning systems for discharge of gaseous radioactive substances to the environment.

(2) The filters of the air cleaning systems shall be sufficiently reliable to perform their function with the necessary decontamination factor in all operational modes. The design shall provide means to test their efficiency.

Article 157. (1) Provisions shall be made in the design for an automated system for radiation monitoring in the rooms and at the NPP site, and a system for radiation monitoring in the emergency planning zones and the surveillance zone. These systems shall ensure the collection and processing of information on the radiation conditions, on the effectiveness of protective barriers, on the radionuclides activity, and information necessary to predict changes in the radiation conditions in all operational states and accident conditions.

(2) The equipment of the automated system for radiation monitoring shall enable the implementation of:
1. Process radiation monitoring;
2. Individual dosimetry monitoring;
3. Radiation monitoring in the work rooms and at the NPP site;

4. Area monitoring for limiting the spread of radioactive contamination.

(3) The laboratory methods and technical means of the system for radiation monitoring shall ensure measurement of the human induced radionuclides content in soil, water, deposits, vegetation, water flora and fauna, and agricultural products.

Chapter Seven
CONSTRUCTION AND COMMISSIONING
Section I
General Requirements

Article 158. The operating organisation shall exercise control over the implementation of the design, construction and installation works, and over the quality of used materials, structures and components assisted by its own organisational structure and in compliance with the requirements of the regulatory framework.

Article 159. During the construction and commissioning, the NPP site shall be monitored, guarded and maintained so as to protect SSCs, to support the testing stage and to maintain consistency with the technical design and the safety analyses.

Article 160. The SSCs supplied to the plant shall be manufactured under quality assurance programmes that include inspections of the technological cycle, cleanliness, calibration and verification of operability.

Article 161. (1) The operating organisation shall ensure designer supervision by the NPP designer to provide technical assistance.

(2) The fabrication of the structures and components at the NPP site, construction methods, installation works, single tests and inspections during the construction stage shall be concurred with the NPP designer.

Article 162. (1) The construction and installation works and the single tests of SSCs important to safety shall be carried out in accordance with written procedures that contain the measures to ensure compliance with the safety requirements, requirements for quality assurance, and requirements for the industrial safety of the personnel.

(2) The activities of the suppliers shall be carried out in accordance with procedures, specifications and drawings with specific requirements for quality assurance.

Article 163. (1) During the construction and commissioning the operating personnel shall profit from the opportunity to receive on-the-job training and thus shall be introduced to the construction and installation solutions, the testings of SSCs and the NPP as a whole and the results thereof, validation of the operational and organisational procedures on maintenance, surveillance and inspection, access to work and management of non-compliances.

(2) All site personnel, including suppliers and contractors participating in the construction and installation works and the commissioning shall be trained to develop safety culture in

compliance with Articles 24 and 25.

Article 164. (1) The SSCs constructed, supplied and installed shall be compared to their design characteristics during construction and commissioning.

(2) Maintenance shall be performed on structures and components that could experience degradation of their characteristics during construction. The maintenance during commissioning shall comply with the same requirements as during operation.

(3) Initial control shall be exerted on SSCs that will have subsequent metal control in the future.

(4) Adequate actions shall be taken to prevent foreign materials and dirt from entering into exposed and unprotected components.

Article 165. When handing over the management and control of the completed works during construction, installation and commissioning from one organisation to another, it shall be ensured that the documentation related to the SSCs is verified for its completeness and accuracy, that all the non-compliances or unfinished works can be identified and resolved in a way that will not affect safety.

Article 166. The following shall be developed, introduced and provided before the commissioning:

1. Organisational documents for commissioning management;
2. Commissioning limits and conditions;
3. Operating procedures and maintenance, tests and surveillance instructions for SSCs important to safety;
4. Documents of the management system, including those for change control;
5. Emergency operating procedures and on-site emergency plan for the NPP;
6. Physical protection system and fire safety system;
7. Sufficient personnel with the required qualification and certification.

Section II

Commissioning Programme

Article 167. The operating organisation shall develop and implement a commissioning programme covering all the operational states. The results of the programme implementation shall demonstrate compliance of the characteristics of SSCs important to safety, and the NPP process parameters with the design requirements and the provisions of the commissioning permit issued by the Chair of the Nuclear Regulatory Agency.

Article 168. (1) The commissioning programme shall ensure that:

1. all the necessary tests to confirm the compliance of the constructed nuclear power plant with the design requirements are completed;
2. tests were performed in their logical sequence;
3. the "hold points" were identified in the commissioning process;
4. the operating personnel was trained and the instructions validated;

(2) The tests conducted under the commissioning programme shall not lead to operational states and accident conditions that have not been analysed in the interim safety analysis report.

Article 169. (2) The NPP commissioning shall be performed in sequential stages, for which separate programmes shall be developed. The implementation of each stage shall be preceded by an evaluation of the results from the previous stage and a confirmation that the objectives set and design requirements have been met.

(2) The programme for each stage shall describe:

1. The sequence, timing and logical connections between the activities at the stage;
2. The initial and final status at the respective stage;
3. The organisation for implementation and the required personnel;
4. The preconditions for implementation of the tests;
5. The requirements on the technological preparation and provision of power sources and operating fluids;
6. The criteria for acceptance and an assessment of their fulfilment;
7. The conditions for transition to the next stage.

(3) The programmes for each stage shall contain a time schedule and a list of procedures to be followed during the tests.

Article 170. (1) The tests shall be carried out in accordance with written procedures that shall have as a minimum requirements on:

1. Introduction and withdrawal of temporary modifications necessary to conduct the testing;
2. Verification that the prerequisites and the preconditions to conduct the tests are fulfilled;
3. Measurement equipment used and its calibration;
4. Limits and conditions to conduct the testing;
5. Test results recording means;
6. Acceptance criteria for the results and their allowable ranges;
7. Clear and unambiguous instructions on the test performance;
8. Clear rules to follow when the acceptance criteria are not met;
9. Plant recovery to normal state following the test performance.

(2) The developing, approving, modification, distribution and storage of the test procedures and the reporting documents containing the results thereof shall be in compliance with the management system.

Article 171. The test results approval shall be arranged in a way that the following objectives are met:

1. The NPP behaviour has been compared with the design requirements;
2. Sufficient data on the revaluation of the design bases are ensured, in case the unit behaviour deviates from the expectations;
3. Demonstrate that the NPP as tested allows to proceed with the next stage of commissioning or with the next test.

Article 172. (1) The commissioning activities shall be carried out in compliance with the commissioning programme, testing procedures, operation, maintenance, surveillance and inspections.

(2) The applicability and quality of the operating procedures shall be confirmed (validation and verification) during the commissioning process.

Article 173. (1) Before the initial core loading with nuclear fuel: SSCs important to safety and required at this stage shall be tested and their availability shall be confirmed; tests to determine the characteristics of the reactor coolant pressure boundary shall be carried out; biological shielding effectiveness shall be tested; radiation monitoring shall be carried out at the premises, site and the environment.

(2) Before reaching initial criticality of the reactor installation, functional tests of SSCs important to safety shall be carried out to confirm the fulfilment of the design functions and the compliance with the design characteristics.

(3) The transition from one power level to another shall be performed after successful neutron physics test (experiments) of the reactor installation and completion of all construction and assembly works.

(4) Trial-testing operation shall be performed as a commissioning stage for evolutionary NPPs.

Article 174. A nuclear power plant unit, which is in a process of commissioning, shall be physically isolated from other units that are in operation or under construction at the same site.

Chapter Eight

OPERATIONS

Section I

Operational Safety Management

Article 175. (1) The management body of the operating organisation shall establish a document defining the safety policy which prioritises safety in all activities and shall demonstrate clear commitment to continuously improve safety and stimulate personnel's critical attitude to the work performed in order to achieve best results.

(2) The personnel performing safety related activities shall be familiar with the Safety Policy to an extent ensuring clear understanding and application. Contractors shall be acquainted with the key aspects of the Safety Policy in order to understand and observe the requirements of the operating organisation.

Article 176. (1) The Safety Policy shall provide for the issue of procedures on its application and monitoring of activities with impact on safety.

(2) The Safety Policy shall define clear safety objectives and intentions which can be easily controlled and tracked by the management personnel.

(3) The Safety Policy shall stipulate continuous improvement of nuclear safety by means of:

1. Continuous process of reassessment of the nuclear power plant safety taking into account the operating experience, safety assessments and analyses, and research and technological developments;
2. Timely implementation of feasible improvements;
3. Timely application of important new information which may be related to the nuclear power plant safety.

(4) The adequacy and status of implementation of the Safety Policy shall be assessed more frequently than the Periodic Safety Review.

Article 177. (1) The nuclear power plant shall be operated safely, providing that:

1. Safety related decisions shall be made in a timely manner and preceded by the relevant research and consultations in order to consider all safety aspects;

2. The safety problems shall be subjected to safety analyses conducted by qualified personnel not involved in the operations;

3. Personnel shall be provided with the required technical means and work conditions in order to perform their activities in a safe manner;

4. Work performance shall be continuously monitored based on the relevant management processes in order to ensure sustaining the required safety level as well as its enhancement, if necessary;

5. Applicable operating experience, development of international safety standards, and new knowledge gained through research and development projects shall be continually analysed and used to improve the NPP as well as the activities involved in its operation;

6. The processes shall be managed using a documented management system covering all the activities, including contractors' ones, that may impact the safe operation.

(2) Compliance with safety regulations as well as physical protection regulations shall meet the safety objectives as well as the physical protection objectives. Possible conflicts shall be solved based on cooperation between the safety officers and physical protection officers.

(3) Occupational safety provisions shall be integrated with the nuclear safety and radiation protection programmes in such a manner as to keep the safety risk as low as reasonably achievable.

Article 178. (1) As part of the operation of a nuclear power plant, a system for continuous monitoring of safety and performance of activities shall be developed and implemented. Systematic self-assessment at all levels of the operating organisation shall be an integral part of this system.

(2) Monitoring and self-assessment shall determine the level of safety performance achieved as well as any indications of a degraded safety.

(3) The monitoring shall cover personnel behaviour and attitude towards safety as well as any breaches of the operational limits and conditions, operating procedures, regulatory requirements, and provisions of the operating licences. For the purpose of monitoring the condition of the NPP, implementation of activities, and personnel behaviour, the plant managers shall conduct systematic walkdowns.

(4) Relevant indicators of safety performance shall be developed and implemented enabling plant managers to detect and correct any weaknesses and non-conformances in safety management.

(5) Based on the monitoring and review of safety performance, corrective actions shall be identified and implemented, controlled and assessed.

Article 179. (1) Periodic Safety Reviews shall assess the consequences of the cumulative ageing effects, modifications and requalification of SSCs, operating experience, current safety standards and research and development achievements, changed features of the NPP site, and organisational and managerial problems. Periodic reviews shall focus on ensuring high level of safety throughout the operating life of the NPP.

(2) The review scope shall cover the safety factors under Article 89 which shall be assessed

using deterministic methods. Within the meaning of Article 80 (1), probabilistic analyses may be used in the summarised safety assessment mostly to determine the contribution of the detected facts to the common safety level.

(3) Based on the results, all the corrective actions and feasible modifications shall be implemented to achieve the safety level prescribed in the current safety standards.

Article 180. (1) During normal operation, all physical barriers shall be effective and all levels of defence shall be available. In case of a failure of a physical barrier or unavailability of a level of defence, the reactor shall be brought to a safe shutdown condition.

(2) A failure of a physical barrier or unavailability of a level of defence at all operational states shall be justified in the design as well as in the safety assessment of the NPP.

Article 181. (1) To maintain the availability of the levels of defence, the NPP operation shall comply with the operational limits and conditions.

(2) The operational limits and conditions shall be defined and justified on the basis of design, safety analyses and commissioning tests, and shall be reviewed periodically and as necessary to reflect the operating experience, modifications of SSCs important to safety, latest safety analyses, and research and technological developments.

(3) Changes of the operational limits and conditions shall be justified based on analyses of the safety margins and independent review of those analyses.

Article 182. (1) The operational limits and conditions shall cover all the normal operation states, including power operation, subcritical reactor, core refuelling, and all the transients between those states, operational states, or temporary conditions resulting from maintenance works and testing, and shall include as a minimum:

1. Safety margins;
2. Safety system actuation parameters;
3. Operational limits and conditions;
4. Tests, inspections, surveillance, and on-line monitoring of SSCs important to safety;
5. Minimum number of shift personnel for the operational states, including the certified and qualified MCR operators;
6. Actions to be taken in case of deviations from the operating limits and conditions.

(2) In order to prevent any undesirable frequent actuation of the safety system, adequate margins shall be ensured between the operating limits and the safety system actuation parameters.

(3) A conservative approach shall be used to define the safety limits, taking into account the uncertainties of the safety analyses.

(4) The operating limits and conditions shall cover operating parameter limits, minimum number of operable SSCs (in service or standby) for each normal operation state, the actions to be taken in case of a deviation from the operating limits, and time for execution of those actions.

(5) If the conditions for operability of the SSCs important to safety can not be met, the actions and time to bring the power unit to a safe and stable state shall be identified.

Article 183. (1) In case the operators are unable to verify that the power unit is operated within the operating limits, or the unit behaviour is unexpected, immediate actions shall be taken to bring it to a safe and stable state.

(2) The power unit shall not be restarted after an unplanned shutdown unless it is demonstrated that this can be executed safely.

Article 184. (1) The surveillance programme shall cover monitoring of compliance with the operating limits and conditions, including the trends within the established margins, in order to detect any deviations from the design targets.

(2) In case of noncompliance with the operating limits and conditions, immediate actions shall be taken to restore compliance with the requirements. In order to prevent recurrence of such noncompliances, corrective actions shall be implemented.

(3) The cases when the actions taken to correct deviations from the operating limits and conditions differ from the prescribed ones, including the cases when the established times for implementation of those actions are exceeded, shall be considered noncompliances with the operating limits and conditions.

Article 185. The operating limits and conditions collected in a single document (Technical Specifications for Operation) shall be easily accessible to the MCR personnel, who shall be knowledgeable of them and their technical basis. The management of the operating organisation shall be well aware of their significance for safety.

Article 186. (1) Activities with an impact on safety shall be analysed and monitored to ensure that the radiological risk remains as low as reasonably achievable when those activities are performed.

(2) All operators' actions shall be assessed against their potential risk. The level of assessment and monitoring of the activities shall depend on their significance for safety.

Testing of SSCs important to safety, special operational states, or experiments which are not included in the Technical Specifications for operation or in the operating procedures shall be assessed against their impact on safety and carried out in compliance with special programmes and procedures following a positive statement issued by the Bulgarian Nuclear Regulatory Agency.

Article 187. (1) The nuclear power plant shall be managed by competent managers who are familiar with the safety assurance principles, specifics of the technological process, and related risks. All the activities with an impact on safety shall be performed by appropriately qualified and experienced personnel.

(2) The NPP operational state and the changes therein shall be monitored and controlled by certified and qualified operators as stipulated in the Safe Use of Nuclear Energy Act.

(3) At least two operators shall be available at the MCR during plant operation and their certificates of competency shall be issued by the Chair of the Nuclear Regulatory Agency.

(4) The duties and authority of the operators and safety officers shall be defined in job specifications and job descriptions for each job position.

Article 188. (1) The shift personnel shall operate the NPP in compliance with written operating procedures and instructions. Strict observance of procedures and instructions shall be an essential element of the NPP Safety Policy.

(2) The level of detail of procedures and instructions shall be based on their intended purpose. The guidance shall be clear and concise, verified, and validated.

(3) Procedures, instructions, and aids shall be clearly identified, discernible in respect of intended function, and readily accessible at the MCR and other control rooms, if necessary.

Article 189. (1) The operating procedures and instructions for normal operation shall be prepared based on the design and engineering documentation, operating limits and conditions, and results of plant commissioning.

(2) Operators' emergency response actions for all the operational states shall be stipulated in Emergency Procedures and Severe Accident Management Guidelines (SAMGs).

Article 190. (1) The Emergency Procedures shall cover the design basis accidents and scenarios where a significant fuel damage at the core or at the spent fuel pool can be prevented. The Emergency Procedures shall be symptom-based (SB EOPs) and compatible with the Alarm Instructions and SAMGs.

(2) The Design Basis Accident Emergency Procedures shall prescribe how to bring the plant to a stable safe condition, while the Emergency Procedures for the scenarios where a significant nuclear fuel damage can be prevented shall prescribe how to restore or compensate for lost safety functions as well as how to prevent nuclear fuel damage at the core or at the spent fuel pool (SFP).

(3) The set of SB EOPs shall include:

1. State diagnostic procedures;
2. Procedures for optimal recovery in the event of transients and design basis accidents;
3. Condition monitoring procedures and procedures for restoration of safety functions, such as subcriticality, core cooling, residual heat removal, coolant inventory, integrity of the reactor coolant pressure boundary and containment integrity;
4. Procedures for transition to severe accident management.

(4) For the preparation of SB EOPs, their format, structure, and contents shall be specified in a way that:

1. Gives precise, clear, and adequate guidance for the personnel to perform the prescribed actions, including when a transition to other procedures, instructions, and guidelines is required, considering only the operation of the equipment and instruments qualified for the relevant work environment;
2. It is easy to distinguish them from the Normal Operation Procedures and it is convenient to use them;
3. They include instructions for monitoring of specific process parameters (symptoms), for monitoring of automated system responses, fundamental operators' actions for immediate execution and the anticipated result of them, as well as alternative operators' actions in case of failure of the fundamental ones;
4. The supplementary background information that aids the operators in observing the procedures is clearly discriminated.

Article 191. (1) The Severe Accident Management Guidelines shall result in mitigation of consequences of severe accidents when the personnel actions, including the actions prescribed in the SB EOPs, have not been successful in preventing core damage or fuel damage at the SFP.

(2) The Severe Accident Management Guidelines and SB EOPs shall:

1. Provide for the management of accidents impacting both the reactor and spent fuel pool and consider the possible interaction between the reactor and SFP;

2. Consider the capabilities for one of one powerunit to support another power unit on the NPP site without compromising its own safety;

3. Ensure their implementation, even when all the on-site nuclear facilities are in accident conditions, and take into account the dependencies among the systems and common resources;

4. Consider the expected conditions on the NPP site, including the radiological situation resulting from the accidents which they are intended for as well as the initiating event or external hazard that may have caused it.

Article 192. (1) Severe Accident Management Guidelines shall be based on strategies for management of the scenarios resulting from the analysis of the weaknesses and capabilities of the nuclear power unit in the event of a severe accident, and the possible measures for management including for containment protection. The SAMGs shall consider as a priority the operation of the equipment and instruments qualified for the relevant conditions.

(2) For the preparation of SAMGs and SB EOPs, unit-specific data shall be used. The effectiveness of the operators' actions shall be analytically validated using verified computer codes and unit-specific calculation models. The results of the analysis shall be documented and used as a technical basis for the procedures.

(3) The emergency procedures and guidelines shall be verified and validated by an independent team of experts according to established internal rules (programmes) in the form they are used. The practical capability to execute the operators' actions shall be validated using simulator tools.

(4) The up-to-dateness of the emergency procedures and guidelines shall be verified by the operating organisation periodically.

Article 193. (1) Throughout the operation of the NPP, a configuration management process shall be developed and implemented to ensure correspondence between the design requirements, physical configuration of SSCs important to safety, and operating documents.

(2) The organisational measures in respect of configuration management shall ensure that the modifications of the NPP and SSCs important to safety are identified, designed, assessed, implemented, and documented.

(3) The appropriate organisational measures shall be planned for the NPP configuration changes resulting from maintenance, tests, repairs, and modernisation of the NPP.

Article 194. Temporary and permanent modifications of SSCs important to safety shall be planned, controlled, and implemented so as not to compromise the safe operation of the plant. Changes, including the ones related to the organisational structure, operating limits and conditions, procedures and instructions, methods and computer codes for safety assessment, shall be categorised based on a graded approach with appropriate criteria in respect of their safety significance.

Article 195. (1) When modifying SSCs important to safety, the following sequence of steps shall be performed:

1. Identification of the causes and justification of the need for modification;
2. Identification of the requirements for design and safety assessment scope and methods;
3. design and justification of safety, manufacturing, procurement, installation, and testing;
4. Documents changes and personnel training;
5. Implementation of the modification.

(2) The initial safety assessment shall assess and justify the category of the modification proposed. A detailed and comprehensive safety assessment shall be conducted unless the initial assessment has demonstrated that the scope of this assessment can be reduced in order to demonstrate that all safety aspects have been considered and the applicable safety requirements met. The scope, safety level conclusions, and consequences of the modifications proposed shall be assessed by the personnel who have not taken part initially in their designing or implementation.

(3) The implementation and tests of SSCs modifications, their impact on procedures and personnel training, including at the full scope simulator, shall be conducted in compliance with the Management System processes and documents.

(4) Prior to commissioning of modifications, the personnel shall be trained on their servicing and operation, and all the relevant operating procedures shall be modified or updated.

Article 196. (1) Temporary modifications of safety related SSCs shall be managed in accordance with specific instructions. All temporary modifications shall be tagged in the field and marked on the alarm panels and controls. The operators shall be aware of those modifications and their effect on the operation of the plant.

(2) The number and duration of the temporary modifications shall be limited to a minimum. The need for the temporary modifications to remain effective shall be assessed periodically.

Article 197. (1) Throughout the operation, a programme shall be drawn up and systematically applied to collect, analyse, and document internal and external operating experience as well as the operating events occurring at the plant.

(2) Based on the assessment of the plant operating experience, the latent weaknesses in respect of safety, potential preconditions and possible trends towards poor performance of activities impacting safety or reducing the safety margins shall be identified. Significant findings and trends shall be reported to the plant management.

(3) For implementation of the programme specified in paragraph 1, for dissemination of the results important to safety, and for identification of recommendations for improvements, personnel who are adequately trained, provided with resources, and supported by the plant management shall be assigned.

(4) To prevent recurrence and counteract events compromising safety, the recommendations for improvement shall be implemented, timely and adequate corrective actions shall be taken, and the best practices shall be considered.

(5) The information related to the operating experience and safety shall be organised, documented, and stored in a manner facilitating its accessibility, screening, and assessment by the personnel assigned for that purpose.

Article 198. (1) Safety significant operating events shall be reported in accordance with established procedures and criteria. Plant personnel shall report deviations from normal operation and shall be encouraged to report safety significant near misses.

(2) In the event of safety significant failures or deviations from normal operation, consultation shall be ensured on the required corrective actions from the manufacturer of the SSCs, from the designer, or from the plant R&D manager.

Article 199. Information based on the operating experience shall be disseminated to the relevant personnel, shared with the interested national and international organisations, and used in the training of the personnel performing activities with impact on safety.

Article 200. Periodic reviews of the operating experience feedback efficiency, based on specific indicators or criteria, shall be conducted by the operating organisation as part of the self-assessment process or by an independent team.

Article 201. (1) Throughout the operation of the NPP, specifications and procedures shall be prepared for procurement, verification, receipt, accountability and control, loading, use, refuelling, and testing of nuclear fuel and core components.

(2) Only nuclear fuel manufactured in accordance with the established specifications and design criteria for the nuclear fuel and its enrichment shall be loaded into the core.

(3) The use of a new type of nuclear fuel shall be deemed to be a modification significantly changing the plant configuration and shall be preceded by a detailed and comprehensive safety assessment.

Article 202. (1) The decisions, planning, assessment, execution, and supervision of all operations or modifications involving nuclear fuel and having a potential impact on the reactivity control shall be undertaken in compliance with the reactivity control programme, approved procedures, and operating limits and conditions for the nuclear fuel.

(2) Regarding core monitoring, a programme shall be drawn up and implemented to ensure that:

1. The core parameters have been measured, trends have been analysed and assessed in order to detect defects in the nuclear fuel;
2. The actual state of the core complies with the design requirements;
3. The values of the basic operating parameters are recorded and stored in a logical, consistent, and easy to use manner.

Article 203. (1) Operations influencing reactivity shall be executed with consideration and care to ensure the reactor remains within the operating limits and conditions, reaching the expected response of the reactor installation.

(2) The operating procedures concerning reactor start-up, power operation, shutdown, and core refuelling shall include safeguards and controls necessary to ensure nuclear fuel integrity and compliance with the operating limits and conditions throughout the entire period of nuclear fuel operation.

(3) Procedures governing the operations involving nuclear fuel and core components shall ensure controlled handling of fresh and irradiated nuclear fuel, adequate storage, transfer or transport preparation. A person having the required experience, knowledge, and qualification shall be assigned in charge of the supervision and conduct of the operations involving nuclear fuel.

Article 204. (1) Outage planning shall be a continuous process, focused on improvements, accounting for completed, forthcoming, and future major maintenance activities.

(2) The process of planning and performing outage activities shall consider and prioritise the safety factors. Special attention shall be paid to the reactivity control, residual heat removal, nuclear fuel handling, and ensuring reactor pressure boundary integrity and containment integrity.

Plant configuration management in this operational state shall comply with the operating limits and conditions.

(3) Programmes and procedures shall be issued to enable outage management and provide the required resources for ensuring safety.

(4) Tasks, responsibilities, authority, and lines of communication of the teams and persons taking part in the preparation, implementation, and assessment of the outage schedules and activities shall be stipulated in administrative procedures and observed by the plant and contractors' personnel.

(5) Outage programmes and procedures shall account for the optimisation of the radiation protection and industrial safety of the personnel, minimisation of the generated radioactive waste, and control of chemical hazards.

(6) Upon completion of each outage, a detailed review of the activities shall be conducted to document the lessons learned.

Article 205. During plant operation, fire safety measures identified in the fire hazard analysis shall be taken. Those measures include requirements for management of the activities impacting fire safety – maintenance, combustible material management, personnel training, testing and emergency exercises, changes in location and configuration of the fire suppression systems, fire detection, ventilation systems, power supply systems, fire safety control systems and process control systems.

Article 206. (1) In order to prevent internal fires, procedures shall be established for the management and minimisation of combustible materials and potential ignition sources that may affect SSCs important to safety. Those procedures shall ensure the operability of fire fighting technical means through inspections, maintenance, and testing of the fire barriers, fire suppression systems, fire detection and alarm systems, and portable fire fighting equipment.

(2) Responsibilities and actions of the personnel in case of fire shall be stipulated in a fire fighting strategy and emergency procedures that are to be learnt in the course of emergency exercises. This strategy shall cover each area where an internal fire may affect SSCs important to safety as well as the radioactive material protection.

Article 207. (1) When national or local fire brigades are involved in the fire fighting, coordination shall be established between the plant personnel and those bodies who shall be familiarised with the risks at the NPP.

(2) When plant personnel take part in fire fighting activities, their organisation and number, competence certification and training requirements shall be documented and verified by an officer competent in fire protection.

(3) Joint emergency exercises shall be conducted periodically to assess fire fighting performance.

Article 208. In case the operating organisation intends to operate the plant beyond the established design life, a large-scale long term operation programme shall be drawn up and implemented, covering the following:

1. Preliminary conditions, including licensing basis, implemented measures for enhancement and verification of the safety level, and existing operating programmes;

2. Identification of SSCs to be covered by the Programme;

3. Categorisation of SSCs in respect of ageing and degradation processes and selection of a strategy for extension of their design life if necessary;

4. Conducting of new safety analyses based on time-limited assumptions and initial conditions;

5. Preparatory plan for long term operation.

Section II
Conduct of Operations

Article 209. The administrative unit in charge of the plant operations shall perform the following tasks, functions, and duties:

1. Planning of all the activities related to the plant operations while preparing an integrated operations programme together with the other administrative units;

2. Planning of human resources and operators' career development;

3. Direct operation of the plant based on monitoring and control of the systems in compliance with the established rules, operating and administrative procedures, and operating limits and conditions;

4. Supervision and control over the operators' actions by the shift supervisor;

5. Organisation of surveillance of the core refuelling activity and outage;

6. Development of operating instructions and procedures and coordination of their preparation to ensure safe and reliable operation of the SSCs;

7. Coordination of the preparation and implementation of programmes and policies for safe operation;

8. Participation in the preparation of programmes for surveillance of SSCs important to safety and coordination of their implementation;

9. Development and performance of the operations management processes which provide operators with information about the activities being carried out and ensure maintaining of the plant configuration;

10. Configuration management based on exact implementation of the changes in the plant status required for the conduct of maintenance, modifications, and tests;

11. Detection of failures, defects, and shortcomings of SSCs in order to plan and effectively perform the maintenance activities;

12. Support of outages by assigning operators for the preparation of the schedule of activities, testing, identification and monitoring of the operating condition of components and systems, and return of the systems to service;

13. Preparation and implementation of procedures for prevention of unauthorised access to or interference with SSCs important to safety;

14. Identification of training needs and participation in the development of training programmes, supervision of training sessions, and assessment of training programmes;

15. Verification of the good housekeeping of rooms, SSCs, and the material condition of the plant areas the organisational unit is in charge of;

16. Identification of operating objectives and intentions which shall be in line with the common plant objectives and intentions;

17. Reporting and participation in the investigation of all operating events and deviations, including near-misses and low-level events, as well as in the identification of measures to reduce

the potential of their recurrence;

18. Registration, assessment, and documentation of the safety significant internal and external operating experience and dissemination of information about it to the operators.

Article 210. (1) During all operational states and accident conditions and at all times, the required number of qualified operators shall be ensured.

(2) The number and composition of operators' shifts shall reflect the periodic training, assignments, and level of automation of the technological processes as well as the provision of diversity and competence redundancy within the shift.

(3) Interactions between the operators during the shift, conduct of operations, channels of communication and execution of commands, shift turnover, and relations with the other administrative units shall be stipulated in a procedure for the operational interactions.

Article 211. (1) Habitability and good working conditions (lighting, noise, radiation level, temperature, means of communication) shall be maintained in the MCR and ECR during all operational states and accident conditions.

(2) The Emergency Response Centre, ECR, and all local control panels shall be kept operable, easily accessible and free of excess materials, provided with the required documents and means of communication, and periodically inspected.

(3) The MCR and ECR ventilation systems along with their filter modules shall be subject to periodic performance tests to ensure their compliance with the design characteristics and purification factors.

Article 212. (1) Throughout the operation of an NPP, alarm procedures shall be prepared and implemented to stipulate operators' actions in the event of alarm signals at the control panels and displays of the MCR information systems. Unexpected alarm signals shall be announced and documented. All alarm signals shall be treated as correct and valid unless they have been verified to be spurious based on assessment against other indications.

(2) The information about the status of the alarm signals and their causes shall be available to the MCR operators.

(3) Unavailability of alarm signals, both due to failure or deliberately taking out of service, shall be documented and the number of such cases shall be minimised.

(4) To determine whether the affected systems and components comply with the operating limits and conditions as well as to monitor the process parameters, the MCR operators shall be provided with alternative means.

(5) In the event of a transient, abnormal operation, or other situation involving multiple alarm signals, a detailed analysis shall be conducted to determine the unexpected or irrelevant alarm signals.

(6) When executing an emergency procedure, the MCR operators shall place a higher priority on the assessment of safety functions performance than on the assessment of the alarm signal status.

Article 213. (1) To maintain water chemistry within the established margins, limit the ingress of chemical contaminants, and limit the radiological factors during the commissioning and operation of the plant, a programme for water chemistry and radiochemistry control shall be implemented.

(2) This programme shall cover monitoring and data processing systems which, along with the laboratory analyses, shall provide for the measurement and recording of chemical and radiochemical parameters.

(3) The operators shall be able to correctly understand the chemical parameters, identify states deviating from the operating limits and conditions, detect adverse trends related to inadequate water chemistry, and take timely corrective actions when necessary.

(4) The data from the radiochemical analyses, which are representative of the integrity of the fuel rod cladding, shall be systematically monitored and analysed in order to assess the trends during all operational states.

(5) The use of chemical reagents shall be strictly monitored, including reagents brought to the plant by contractors. Appropriate engineering and organisational measures shall be in place when using chemical substances and reagents to prevent harmful effects or degradation of SSCs important to safety.

Article 214. (1) During the operation of the power plant, good working conditions shall be maintained at all the working areas. The rooms, systems, and components shall be well maintained, lighted, clean of lubricants, chemicals, and waste, and easily accessible for work. The components where fluid leaks, corrosion spots, loose parts, or damaged thermal insulation have been found, shall be reported and repaired in a timely manner.

(2) The means of radiation protection, industrial safety, emergency medical services, and fire protection shall be adequately situated, well marked, and available during all operational states. Evacuation routes shall be well lighted, clearly marked, and free of all kinds of materials and objects.

(3) Operators shall be encouraged to detect and report non-conformances and shortcomings in the condition of the systems, components, and rooms.

Article 215. (1) For identification of the SSCs, marking rules shall be established, applied, and constantly maintained. These rules shall be well known to the personnel. Those rules shall enable clear identification of each individual component and room.

(2) Marking plates shall be suitable for the environment in which they are installed. Their form and location shall allow for clear and easy identification of the components and prevent easy removal or wrong installation.

(3) Marking rules shall enable the personnel to identify missing or required plates and shall ensure timely implementation of corrective actions.

Article 216. (1) The operating condition of all active components of the systems important to safety shall be documented, periodically inspected during operation, and monitored before returning to service of a power unit, scheduled annual outages, and planned changes to the reactor installation state.

(2) Specific measures shall be developed and maintained to prevent unauthorised access to or interference with the SSCs important to safety. Those measures shall include controlled access to specific rooms and zones as well as an effective control panel locking system. The measures shall not hinder the operators from effectively checking the safety system operational readiness as

well as from executing quick and timely switchovers during normal operation and accident conditions.

Section III
Maintenance, Tests, Surveillance, and Inspections. Ageing Management

Article 217. (1) During commissioning and operation, maintenance, test, surveillance, and inspection programmes shall be prepared and implemented to ensure compliance of the operability, reliability, and functionality of SSCs important to safety with the design criteria throughout the entire period of plant operation. Those programmes shall consider the operating limits and conditions and shall be revised in order to reflect operating experience.

(2) The programmes for predictive, preventive, and corrective maintenance shall cover activities for control of degradation processes, prevention of failures, and restoration of the operability and reliability of SSCs important to safety.

(3) Maintenance programmes shall consider the results of the ageing management programme and cover replacement of obsolete SSCs or SSCs whose design life has expired, requalification of SSCs important to safety, and use of new maintenance technologies.

(4) The programmes for periodic inspections, surveillance, and testing shall confirm either that the SSCs important to safety meet the requirements for extended safe operation, or that restoration measures are required.

Article 218. (1) The scope and frequency of the maintenance, tests, surveillance, and inspections of SSCs shall be determined using a systematic approach based on:

1. Their importance to safety;
2. Their inherent reliability;
3. Their susceptibility to degradation (based on operating experience, scientific studies, and manufacturers' recommendations);
4. Operating as well as any applicable experience and the results from the SSCs condition monitoring.

(2) Throughout the entire operating life of the NPP, the SSCs intended for prevention of practically eliminated accident sequences shall be provided with the same reliability as the one used in the design justifications.

(3) In-service inspection shall be performed at intervals determined on the basis of detection of each degradation of the most loaded component before it results in a failure.

Article 219. The data from maintenance, tests, surveillance, and inspections of SSCs shall be documented, stored, and analysed. Those data shall be used to detect emerging and recurrent failures as well as to revise the preventive maintenance programme.

Article 220. (1) The maintenance programmes shall be revised periodically (at periods of less than 10 years) to reflect the operating experience. All the changes proposed shall be assessed against their effect on the SSCs' operability, and their compliance with the relevant requirements. The maintenance total potential impact on plant safety shall also be assessed.

(2) When using the PSA methods to manage maintenance, a comprehensive and structured approach shall be adopted to identify the failure scenarios.

Article 221. For the conduct of maintenance, tests, surveillance, and inspections jobs, working instructions shall be prepared, validated, and approved. Those instructions shall also stipulate the actions to be performed in case the obtained results deviate from the acceptance criteria.

Article 222. (1) During commissioning and operation, a comprehensive planning and work management system shall be applied to ensure the maintenance, tests, surveillance, and inspection activities are duly authorised, safely performed, and documented in compliance with the work procedures.

(2) The works on SSCs important to shall be planned in a manner ensuring plant safety throughout the entire period of plant operation and complying with the operating limits and conditions. When planning works in the protected area, the principle of optimisation of the personnel radiation protection shall be applied. Operating personnel shall take an active role in the process of work planning and prioritisation.

(3) Authorisation of the works performance shall be granted upon a proper and adequate assessment of the personnel health hazards and plant safety resulting from the specific activity and taking into account the following specific aspects:

1. Compliance with the operating limits and conditions;
2. Isolation of the workplace, control devices, mechanical and electrical parts of the SSCs, and tagging;
3. Radiation protection, fire safety, and industrial safety measures;
4. Draining, filling, and venting of the technological systems;
5. Draining and venting means at the workplaces;
6. Monitoring of the implemented modifications.

Article 223. Taking safety related SSCs out of service for maintenance, tests, surveillance, and inspections shall only be performed upon an authorisation by the operators in charge of the relevant SSCs. Returning SSCs to service shall be authorised by the operators upon a written confirmation that the new configuration complies with the operating limits and conditions as well as upon completion of performance tests, if required.

Article 224. (1) Corrective maintenance of SSCs shall be planned, authorised, and executed as soon as possible and in compliance with the operating limits and conditions. Activity prioritisation depends on the defective SSCs relative importance to safety.

(2) After each operating event that has affected safety functions or the functional integrity of a component or system, a verification of the safety functions and the relevant remedial actions shall be performed, including inspections, tests, maintenance and repair, if necessary.

Article 225. (1) The reactor coolant pressure boundary system shall be leak-tested prior to unit restart following a refuelling outage.

(2) A reactor coolant pressure boundary pressure test shall be performed at the end of each inspection period, as well as in the cases identified in the nuclear power plant design.

Article 226. Surveillance measures to verify containment integrity shall cover:

1. Containment leak rate tests;

2. Individual tests of leak-tight penetrations and sealing devices, such as airlocks, doors, and isolation valves which are part of the containment, to determine their leak-tightness and, if necessary, their operability;

3. Integrity inspections of containment metal liner and prestressing tendons.

Article 227. (1) The instruments and equipment intended for tests and studies shall be qualified and metrologically checked before use. Record keeping and validity of metrological assurance shall comply with the management system.

(2) The methods and procedures, instruments and personnel to be used for the reactor coolant system in-service inspection shall be qualified.

(3) In case of detecting unacceptable defects in the metal or new indications of defects in certain sections, additional tests shall be conducted in the problematic section and other analogous sections. The scope for further tests shall depend on the defects' nature and extent to which they influence the assessments of nuclear safety and potential consequences.

Article 228. (1) The operating organisation shall prepare, implement, assess, and improve an ageing management programme to identify all the mechanisms for ageing of SSCs important to safety, their possible consequences as well as the required actions to maintain the SSCs' operability and reliability. The ageing management shall start at the design stage.

(2) All SSCs important to safety shall be assessed in terms of performance of intended safety functions throughout their design life taking into account the ageing and wear mechanisms as well as possible degradation resulting from the time of operation and load cycles.

(3) Environmental conditions, process parameters, maintenance and test schedules, and strategy for replacement of SSCs important to safety shall be considered factors in ageing management.

(4) Monitoring, testing, sampling, and inspections shall be conducted to evaluate ageing effects and early detection of unanticipated behaviour or degradation.

(5) In case of specific conditions resulting from a delay in the construction of a nuclear power plant or a prolonged shutdown, the operating organization shall implement measures to manage the potential impact on the ageing of the SSC in the scope of the ageing management program.

Article 229. The ageing management programme shall be assessed and updated at least during the periodic safety review to include new information and emerging problems, use advanced methods and tools, and assess the maintenance practice used.

Art. 229a. The operating organization shall proactively identify SSCs for which reliability and availability may be at risk due to technological obsolescence and shall develop a long-term strategy to ensure the management of moral obsolescence.

Article 230. (1) Ageing management of the reactor pressure vessel and its welding seams shall consider all relevant factors, including radiation embrittlement, thermal ageing and fatigue, and shall compare their evolution throughout the plant design life with the design limits.

(2) Surveillance of major structures and components shall detect in a timely manner the initial phases of ageing processes in order to reduce preventive and remedial actions.

Section IV

Radiation protection during operations

Article 231. Before commissioning of a nuclear power plant, a Radiation Protection Programme and Environmental Radiological Monitoring Programme shall be prepared and approved by the national competent authorities. Throughout the operation of an NPP, the programmes shall be periodically reviewed and updated based on the operating experience.

Article 232. (1) The Radiation Protection Programme for the personnel shall be aimed at limiting the personnel dose exposure below the authorised limits and as low as reasonably achievable during all operational states of the plant.

(2) Compliance with the Radiation Protection Programme as well as the accomplishment of its objectives shall be verified based on surveillance, inspections, and audits.

Article 233. The Radiation Protection Programme for the personnel shall be based on a preliminary analysis of the radiological risk which shall consider the zones and proportions of the radiological hazards and shall cover:

1. Classification of radiation zones and access control;
2. Internal regulations and activity surveillance at the radiation zones;
3. Personnel and workplace radiation monitoring;
4. Planning of activities and work authorisations at the radiation zones;
5. Collective and personal protective equipment;
6. Facilities, shielding materials, and radiation protection equipment;
7. Personnel medical surveillance;
8. Application of the protection optimisation principle;
9. Reduction of ionising radiation sources;
10. Radiation protection training;
11. Emergency response actions.

Article 234. (1) The limits on liquid and gaseous radioactive discharges stipulated in the Technical Specifications shall be based on an assessment of their radiological effect on representative members of the public using modelling of the anticipated exposure doses in relation to the distance from the NPP, food and drinking water consumption, sources of food and drinking water, and all local customs and practices which may cause exposure doses higher than the average.

(2) The limits on liquid and gaseous radioactive discharges shall be monitored at the sources of those discharges and verified through environmental measurements. Three types of measurements can be used for source monitoring:

1. Online discharge monitoring;
2. Continuous sampling and laboratory measurements of sample activity;
3. Periodical sampling and laboratory measurements of sample activity.

(3) The type of measurement to be chosen for source monitoring shall depend on the source term, sensitivity of the measuring system, anticipated discharge variations in time, and potential for unplanned discharges requiring timely detection and notification.

Article 235. (1) The Environmental Radiological Monitoring Programme under Article 231 shall be implemented at least two years prior to commissioning of an NPP. The programme shall cover measurement of the radiation background in the vicinity of the plant and its seasonal fluctuations, sampling and analysing the radionuclide content of samples of vegetables, air, milk, water, sediments, fish, soil, and plants from several spots identified off the plant site.

(2) The Environmental Radiological Monitoring Programme shall provide information for the purpose of:

1. Verifying the adequacy of the radioactive discharges monitoring;
2. Comparing environmental radiological monitoring data with radioactive discharge source monitoring data;
3. Checking the validity of the environmental models used to determine the authorised limits of radioactive discharges;
4. Strengthening public trust;
5. Evaluating the trends in the concentrations of radionuclides in the environment.

Article 236. (1) Throughout the operation of an NPP, generation of radioactive waste shall be kept as low as possible in terms of both activity and scope based on the appropriate operating practice.

(2) The structures, systems, and components used to treat and store radioactive waste throughout the operation of the plant shall be maintained in an operable condition and monitored through the maintenance, surveillance, test, and inspection programmes.

(3) Accumulation of large quantities of untreated radioactive waste as well as their prolonged temporary storage outside the design facilities shall be avoided.

(4) Radioactive wastes shall be characterised at all stages of their management throughout the operation of the plant. The process of characterisation shall cover the measurement of physical and chemical parameters, source term, and specific activity.

(5) Variations in the radioactive waste characteristics during their storage shall be monitored through inspections and periodic analyses.

(6) RAW treatment and storage shall provide for their removal from the facilities at the end of their storage period.

(7) Throughout the operation of an NPP, the relevant margins shall be ensured in the RAW storage facilities in case of unanticipated circumstances. The facilities’ fill-up level shall be monitored and, if necessary, actions shall be taken to recover the margins.

(8) The On-Site Emergency Plan of an NPP shall provide for the storage of large quantities of liquid radioactive waste in the event of a core melt accident or damage of the spent nuclear fuel stored at the plant.

Article 237. (1) The administrative unit assigned to control the radiation protection during the operation of an NPP shall have sufficient independence and required resources to be able to efficiently apply the optimisation principle, propose and monitor the internal radiation exposure limits, and take actions to comply with the radiation protection regulations at all levels of the operating organisation.

(2) The plant management shall identify mechanisms for including the personnel into the development of methods for maintaining the personnel and public radiation exposure doses as low as reasonably achievable and shall provide feedback on the efficiency of the radiation protection measures.

(3) Plant and contractors’ personnel shall be aware of the radiological hazards of the activities being performed and shall bear personal responsibility for putting into practice the personal protective measures.

(4) Plant personnel shall be recruited in advance and subjected to periodical medical examinations in order to verify their physical and psychophysiological fitness for the relevant job positions.

Section V

Personnel Qualification and Training

Article 238. (1) The operating organisation shall establish qualification and training requirements to ensure that when performing safety related activities the personnel are capable of carrying their duties irrespective of the plant operational states or accident conditions.

(2) Plant management shall adopt a common training policy and implement a detailed training plan based on the long-term needs for qualified personnel as well as identify the training objectives substantiating the critical role of safety in all aspects of operation and supporting safety culture.

Article 239. (1) To train the personnel performing safety related activities, initial and continuing training programmes shall be prepared and implemented for each major personnel unit. Each training programme contents shall be based on the systematic approach to training.

(2) The training programmes, lecture materials, technical training aids, computer models, and full-scope simulators shall reflect the actual condition of the plant.

(3) The personnel training programmes shall cover analyses of internal and external operating experience, operating events, root causes identified, and relevant corrective actions. Changes in regulations, international safety standards, and nuclear technology shall be considered in the management training programmes.

(4) Plant technical personnel and contractors shall complete a basic training course in nuclear safety, radiation protection, fire protection, and actions from the On-Site Emergency Plan.

Article 240. (1) The NPP operators shall be prepared and trained to gradually occupy ascending operating positions upon shadowing the corresponding work places over a justified period of time.

(2) Operator training programmes shall cover the NPP design basis, final safety analysis report, operating limits and conditions, on-site emergency plan, implemented modifications of SSCs important to safety, and safety culture attributes.

Article 241. (1) The main control room operators shall be trained at a full-scope simulator at least once a year, and shift crews shall complete periodic emergency drills.

(2) Maintenance personnel shall be trained on mock-ups or real components to improve their professional skills and reduce the duration of the operations executed before the actual performance of radiologically hazardous maintenance activities.

(3) Before and after key operations and tests on SSCs important to safety, the personnel involved shall receive briefings.

Section VI

Management of Safety Related Documentation

Article 242. (1) As part of the management system, the documentation management process throughout the operation of an NPP shall ensure the availability, control, use, and storage of safety related documents and data, such as:

1. Design specifications;
2. Safety analyses and fire risk assessments;
3. Equipment and material data;
4. As-built system drawings;
5. SSCs' manufacturer documentation;
6. Commissioning data;
7. Operational data;
8. Operating event reports;
9. Records of amounts and movements of nuclear, radioactive and other special fissile materials and substances;
10. Maintenance, test, surveillance and inspection reporting documents;
11. Designs, as-built drawings, and information about the implemented modifications on SSCs important to safety;
12. Management System documentation;
13. Data on personnel qualification, occupied positions, medical examinations, and training;
14. Water chemistry data;
15. Personnel occupational exposure data;
16. Data on radiation monitoring at the premises and on the site;
17. Data on radioactive liquid and gaseous discharges;
18. Environmental monitoring data;
19. Data on the storage and transport of radioactive waste;
20. Periodic Safety Review reporting documents.

(2) The documentation management process shall ensure reliable completion of the configuration management process throughout the operation of an NPP.

Article 243. (1) The documents under Article 242, items 1, 2, 3, 4, 8, 11 and 15 shall be kept in two copies in two physically separated premises, protected against fire and flooding.

(2) The on-site emergency plan, procedures that may be used in accident conditions, and other important documents may be stored outside the plant site, if necessary, providing adequate measures against unauthorised access have been taken.

Section VII

Preparation for Decommissioning

Article 244. (1) The operating organisation shall prepare and support throughout the operation of an NPP a decommissioning plan which shall demonstrate safe decommissioning and reaching the final state of the plant site in compliance with the Regulation on Safety during Decommissioning of Nuclear Facilities.

(2) The decommissioning plan shall be updated to reflect changes in the regulatory

requirements, plant modifications, technological developments, changes of the need for decommissioning, and the Strategy under Article 74 of the Safe Use of Nuclear Energy Act.

Article 245. To ensure the necessary motivated and qualified personnel for safe operation throughout the entire design life, for safe preparation of the decommissioning, and for safe decommissioning, a human resource management programme shall be prepared.

Article 246. (1) Throughout the preparatory period for decommissioning until removal of the nuclear fuel from the reactor and spent fuel pool, a high level of safety shall be maintained.

(2) For sites with more than one power unit, measures shall be taken to ensure availability of the common SSCs for maintaining the safety and support the needs of the power units in operation.

Article 247. To facilitate the planning of decommissioning, the data on contaminated or irradiated SSCs shall be stored and documented in the course of the plant design life. All the analysed data shall be collected and submitted to the holder of the nuclear facility decommissioning licence.

SUPPLEMENTARY PROVISIONS

§ 1. As used in this regulation:

1. “Accident conditions” shall mean deviations from normal operation that are more severe than anticipated operational occurrences, including accidents with or without fuel melt.

2. “Active component” shall mean a component whose functioning depends on an external input such as actuation, mechanical movement or supply of power.

3. “Capable fault” shall mean a tectonic fault that has a significant potential for displacement of adjoining earth blocks at or near the ground surface.

4. “Safe failure” shall mean a failure of a system or component as a result of which the nuclear power plant transitions to a safe state with no necessity for any actions to be initiated by safety control systems.

5. “Safe state” shall mean a plant state, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and maintained stable for a long time.

6. “Main control room” (MCR) shall mean that part of a nuclear power unit situated in specially designed premises and intended for centralised control of the technological processes performed by the operating personnel and the instrumentation and controls.

7. “Validation” shall mean the process of determining whether a product (such as computer programmes, analytical methods, NPP models, procedures and instructions) is adequate to perform its intended function satisfactorily.

8. “Verification” shall mean the process of determining whether the quality or performance of a product (such as computer programmes, analytical methods, NPP models, procedures and instructions) is as stated, as intended or as required.

9. “Probabilistic safety assessment” shall mean a comprehensive, structured approach to identifying failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk.

10. “External event” shall mean an event unconnected with the operation of a nuclear power plant (NPP) that could have an effect on the safety of the NPP.

11. “Internal event” shall mean an event connected with the operation of an NPP that could

have an effect on the NPP safety.

12. “Inherent safety features of a reactor installation” shall mean the reactor installation property to ensure safety on the basis of natural feedbacks, processes and characteristics.

13. “Geodynamic zone” shall mean a linear or ring-shaped part of the earth’s crust within the boundaries of which the identified velocity gradient of the quaternary movements is equal to or greater than 10^{-9} a year.

14. “Large releases” shall mean releases of radioactive material to the environment, which necessitate off-site protective actions to be implemented for protecting people and their application cannot be limited in terms of times and areas.

15. “Single failure” shall mean a failure which results in the loss of capability of a system or a component to perform its intended safety function(s), and any consequential failure(s) which result(s) from it.

16. “Operational states” shall mean the states defined under normal operation and anticipated operational occurrences.

17. “Operational limits” shall mean the values of parameters and characteristics of the condition of systems (components) and of the nuclear power plant as a whole, specified in the design for normal operation.

18. “Operational conditions” shall mean the conditions specified in the design with regard to amount, characteristics, functional capability and maintenance of the systems (components) necessary for operation without violation of the operational limits.

19. “Operation” shall mean all activities performed to achieve the purpose for which the nuclear power plant was constructed, including power operation, start-up, shutdown, testing, maintenance, repairs, refuelling, in-service inspection and other related activities.

20. “Operating organisation” shall mean a person or organisation applying for a licence/an authorisation or a licensee/authorised organisation pursuant to the provisions of the SUNEA (Safe Use of Nuclear Energy Act).

21. “Qualification” shall mean the process of generating evidence that the structure, system or component will operate on demand, under specified service conditions to meet the system performance requirements.

22. “Components” shall mean instruments, pipes, cables, and other articles that ensure the performance of pre-set functions either on their own or within systems, and conceived as structural units in the reliability and safety analyses.

23. “Conservative approach” shall mean an approach in design and construction in the analysis and calculations of which the accepted values and limits for parameters and characteristics definitely lead to more unfavourable results.

24. “Structures, systems and components”, (SSCs), shall mean all of the elements (items) of a nuclear power plant, except human factors. Structures are passive elements, such as buildings, vessels and shielding. A system comprises several components assembled in such a way as to perform a specific (active) function.

25. “Structures, systems and components important to safety” shall mean the safety systems, SSCs for normal operation whose malfunction or failure could lead to unacceptable radiation exposure of the site personnel or members of the public, and the engineered features provided to mitigate the consequences of malfunction or failure of structures, systems and components.

26. “Structures, systems and components for normal operation” shall mean the SSCs intended for normal operation.

27. “Controlled state” shall mean the plant state, following an anticipated operational occurrence or accident conditions, in which fulfilment of the fundamental safety functions can be

ensured and which can be maintained for a time sufficient to implement provisions to reach a safe state.

28. “Reactor coolant system” shall mean the system intended for the coolant circulation through the reactor core in the states and conditions for operation considered in the design.

29. “Ultimate heat sink” shall mean a medium into which the transferred residual heat can always be accepted, even if all other means of removing the heat have been lost or are insufficient.

30. “Safety culture” shall mean the personnel qualification and psychological preparation which establishes that ensuring the nuclear power plant safety is an overriding priority and inherent necessity, leading to responsibility awareness and self-control in performing all activities that influence safety.

30a. “Leadership” is the ability of a person to give direction to and motivate individuals and groups and to influence their commitment to shared goals, values and behaviour.

31. “Licensing basis” shall mean a set of regulatory requirements applicable to the safety of a nuclear power plant, including agreements and commitments made between the regulatory body and the licensee in the form of letters exchanged or of statements, opinions and minutes made at bilateral technical meetings.

32. “Normal operation” shall mean operation of the nuclear power plant within specified operational limits and conditions, and this includes start-up, power operation, reactor installation planned and unplanned shutdown, core refuelling, maintenance and testing of structures, systems and components.

33. “Common cause failure” shall mean a failure of two or more systems or components due to a single specific event or cause.

34. “Anticipated operational occurrence” shall mean a deviation of an operational process from normal operation that is expected to occur at least once during the operating lifetime of a nuclear power plant but which, in view of appropriate design provisions, does not cause any significant damage to SSCs important to safety or lead to accident conditions.

35. “Passive component (passive system)” shall mean a component (system) whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.

36. “Periodic safety review” shall mean a systematic reassessment of the safety of an existing NPP carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience and technical developments, and aimed at ensuring a high level of safety throughout the service life of the nuclear power plant.

37. “Personnel” shall mean all persons working permanently or temporarily on the NPP site.

38. “Postulated initiating event” shall mean a single failure in a system (component), external event or operator error identified in design as capable of leading to anticipated operational occurrences or accident conditions.

39. “Near miss” shall mean a potential significant event that could have occurred as the consequence of actual occurrences but did not occur owing to the conditions prevailing at the nuclear power plant.

40. “Cliff edge effect” shall mean an instance of disproportionately severe abnormal consequences from an event following a relatively small deviation in a parameter or a state of SSCs.

41. “Practically eliminated states” shall mean states, events, sequences of emergencies, scenarios, processes, phenomena, conditions and loads that would be physically impossible to arise or regarding which a combination of deterministic and probabilistic safety analyses have

proven the efficiency of the practical measures in place and the extremely low probability of occurrence determined with a high level of confidence. The criterion of physical impossibility is applied with priority in demonstrating practically eliminated states.

42. "Safety limits" shall mean values of design-specified parameters of a technological process, the deviations from which may result in an accident.

43. "Operational limits and conditions" shall mean a set of rules setting forth parameter limits, the functional capability and the performance levels of SSCs and personnel approved following a specified procedure for safe operation of an NPP.

44. "Design basis" shall mean the range of conditions and events taken explicitly into account in the design and the improvements of a nuclear power plant, according to established criteria, so as not to exceed authorized limits through the intended activity of SSCs that perform safety functions.

45. "Design limits" shall mean values of parameters and of characteristics of the condition of SSCs important to safety and of an NPP as a whole, specified in the design for all operational states and accident conditions.

46. "Diversity" shall mean the presence of two or more redundant components or systems to perform an identified function, where the different systems or components have different attributes so as to reduce the possibility of a common cause failure.

47. "NPP siting area" shall mean the territory, incorporating the NPP site, for which the conditions for NPP placement have to be identified and where occurrence is possible of phenomena, processes and factors of natural and human induced origin that may affect the plant safety.

48. "Early releases" shall mean radioactive releases to the environment that would require off-site emergency measures for protection of the public, which is rendered impossible due to insufficient time to implement them.

49. "Supplementary control room" shall mean part of a nuclear power unit situated in premises envisaged in the design and intended - in case of a loss of functions of the MCR - to reliably bring the reactor in cold subcritical state and maintain it thus for indefinite time, to actuate the safety systems and to receive information on the reactor state.

50. "Redundancy" shall mean provision of alternative (identical or diverse) structures, systems and components, so that any single one of them can perform the required function regardless of the state of operation or failure of the other structures, systems and components.

51. "Safety system" shall mean a system important to safety provided to ensure the safe shutdown of the reactor or the residual heat removal (transfer) from the reactor core, or to limit the consequences of anticipated operational occurrences and accidents caused by a single initiating event. Safety systems are protection systems, confinement systems, safety actuation systems and the safety support systems.

52. "Plant states" shall mean the operational states and accident conditions.

53. "Severe accident" shall mean an accident involving significant core degradation.

54. "Accident management" shall mean taking of a set of actions in order to: prevent evolution of an event into a severe accident; mitigate the consequences of a severe accident; achieve a stable safe state for a long time.

55. "Functional isolation" shall mean prevention of the influence from the mode of operation or failure of one circuit or system on another. Functional isolation is applied to interconnected circuits and systems to prevent propagation of defects and spurious signals from one to another, and may include isolation of electrical circuits, information flow, etc.

56. "Safety function" shall mean the specific objective that must be accomplished in order to

ensure safety.

57. “Containment structure” shall mean an assembly of components of civil and other structures that encloses the reactor installation surrounding space, forms the physical barrier provided in the design, and prevents the dispersion of radioactive substances to the environment. The space closed within the containment structure boundaries constitutes the containment.

§ 2. This Regulation introduces the provisions of the Council Directive 2014/87/Euratom, of 8 July, 2014, (OB, L 219, 25/07/2014), amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations (OB, L 172, 02/07/2009).

TRANSITIONAL AND FINAL PROVISIONS:

§ 3. This Regulation supersedes Regulation on Ensuring the Safety of Nuclear Power Plants, approved by Decree of the Council of Ministers No 172 of 2004 (promulgated in SG issue 66 of 2004; amended in issue 46 of 2007, issue 53 of 2008 and issue 5 of 2010).

§ 4. (1) The provisions of this Regulation shall apply to existing nuclear power plants that have been commissioned prior to the entry into force of this Regulation, insofar as this is practicable.

(2) Persons operating nuclear power plants under para. 1, within one year from the entry into force of this Regulation shall prepare and submit to the Chairperson of the Nuclear Regulatory Agency an assessment of the applicability of the new safety requirements to these plants and a programme containing:

1. all practicable measures to bring the respective nuclear power plants into compliance with the safety requirements of this Regulation;

2. corrective or compensatory measures to mitigate inconsistencies - with regard to safety requirements that cannot be met in full, determined on the basis of in-depth analysis.

§ 5. The Chairperson of the Nuclear Regulatory Agency shall provide instructions for implementation of the Regulation and, if necessary, print guides on applying of its provisions.

§ 6. This Regulation is adopted on the grounds of Article 26, paragraph 2 of the SUNEA.

Appendix to Article 50

Indicative list of enveloping scenarios and events to be considered in the NPP design

Events with “PWR” indication are applicable to pressurized water reactors, and those with “BWR” indication are applicable to boiling water reactors.

Events without indication are applicable to all light-water reactors.

1. Postulated initiating events:

1.1. Pipe rupture with small, medium and large loss of coolant (including the bounding case of the largest diameter pipe rupture in the reactor coolant pressure boundary).

1.2. Pipe rupture in the secondary circuit with leaks of water and steam.

1.3. Forced decrease of reactor coolant flow.

1.4. Forced increase of coolant flow (BWR).

1.5. Forced increase or decrease of the main feedwater flow.

1.6. Forced increase or decrease of the main steam flow.

1.7. Inadvertent opening of a pressurizer valve (PWR).

1.8. Inadvertent operation of the emergency core cooling system.

1.9. Inadvertent opening of a steam generator safety valve (PWR).

1.10. Inadvertent opening of the main steam relief/safety valves (BWR).

1.11. Inadvertent closure of main steam isolation valves at a steam line.

1.12. Steam generator tube rupture (PWR).

1.13. Inadvertent trip of turbine generator.

1.14. Uncontrolled movement of a control rod group.

1.15. Uncontrolled ejection of a control rod.

1.16. Inadvertent decrease of the boric acid concentration in the reactor coolant and in the spent nuclear fuel pool (PWR).

1.17. Core instability (BWR).

1.17. Malfunctioning of the chemical and volume control system (PWR).

1.19. Pipe rupture or heat exchanger tube leaks in systems connected to the reactor coolant system and installed partially outside the containment structure.

1.20. Loss of off-site power

1.21. Fuel handling accidents

1.22. Load drop due to a lifting device failure.

2. Internal and external events and hazards.

2.1. Internal fire.

2.2. Internal explosion.

2.3. Internal floods.

2.4. Flying objects (missiles) as a result of component destruction.

2.5. Leaks of fluids (including oils) from damaged systems.

2.6. Vibration.

2.7. Pipe whip effects, reactive forces and other force impacts induced by high-energy line breaks.

2.8. Initiating events resulting from an earthquake, external flood or other natural phenomena and hazards.

2.9. Initiating events due to aircraft crash or other events of human-induced origin.

2.10. Initiating events caused by transport or industrial activities in close proximity to the plant site, or by other conditions that could cause external fires and explosions or otherwise endanger plant safety.

3. Multiple failure events.

3.1. Prolonged total loss of power (up to several days, with or without considering the additional stationary sources of AC power).

3.2. Short-term or prolonged (up to several days) loss of the main ultimate heat sink.

3.3. Anticipated transients without scram (ATWS).

3.4. Uncontrolled concentration decrease of the soluble neutron absorber in the coolant (PWR).

3.5. Accident with high reactivity insertion.

3.6. Total loss of feedwater.

3.7. Loss of coolant accident and total loss of an emergency core cooling system (e.g. high pressure system and hydro accumulators).

3.8. Rupture of main components under high pressure (not covered by postulated initiating events).

3.9. Uncontrolled decrease of reactor coolant level during refuelling or shutdown for outage (PWR).

3.10. Total loss of service water.

3.11. Loss of core cooling in residual heat removal plant state.

3.12. Loss of the active cooling system for a spent fuel pool.

3.13. Rupture of a large number of steam generator heat exchanging tubes (PWR).

3.14. Long-term loss of safety systems after postulated events that require them to operate properly.

3.15. Initiating events due to an earthquake, external flood and other natural phenomena and hazards more severe than design basis events.

3.16. Initiating events due to aircraft crash or other events of human-induced origin more severe than design basis events.

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1 The time period required to restore safety functions and/or receive assistance from outside is considered.