

REGULATION ON ENSURING THE SAFETY OF NUCLEAR POWER PLANTS

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Chapter One GENERAL PROVISIONS

Art. 1. (1) The current regulation defines the basic criteria and rules of nuclear safety and radiation protection (safety) of nuclear power plants (NPP), as well as the administrative provisions and the technical requirements for ensuring safety during the stages of site selection, design, construction, commissioning and operation.

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

Art. 2. A nuclear power plant is assumed to be safe when its radiation impact in all operational states is kept at a reasonably achievable low level and is maintained below the regulatory prescribed dose limits for internal and external exposure of the personnel and population, and when in case of any accident, including those of very low frequency of occurrence, the radiation consequences can be mitigated.

Art. 3. The NPP safety shall be ensured through consistently applying the defence in depth concept based on the use of a system of physical barriers to the release pathways of ionising radiation and radioactive substances to the environment, as well as on a system of technical and organizational measures to protect the barriers and retain their effectiveness and to protect the population, the personnel and the environment.

Art. 4. (1) The system of physical barriers of any nuclear unit of a NPP shall include: the fuel matrix, the fuel cladding, the reactor coolant system pressure boundary and the reactor containment (confinement) system.

(2) The system of technical and organizational measures shall include the following levels of defence in depth:

1. first level – prevention of anticipated operational occurrences:
 - a) evaluation and selection of an appropriate site;
 - b) identification of the radiation protection area and the monitored area around the NPP for conducting the control needed for radiation protection purposes and for implementation of planned protective measures;
 - c) development of a NPP design, based on a conservative approach and providing for inherent safety features of the reactor installation;
 - d) ensuring the necessary quality of NPP structures, systems and components (SSCs) and of all activities that are being carried out;
 - e) operation of the NPP in compliance with the respective legislative requirements, the operational limits and conditions, and the operating instructions;
 - f) maintaining the operability of SSCs important to safety through timely failure detection, implementation of preventive measures, replacement of structures and components with spent lifetime, and establishment of an effective system for documentation of the results of activities carried out and of related operational supervision;
 - g) staff selection and providing the NPP personnel with the necessary qualification and safety culture of enabling them to take actions in all operational states and accident conditions;
2. second level – prevention of design basis accidents by the systems for normal operation;
 - a) detection of the deviations from the normal operation and coping with them;
 - b) operation with deviations;
3. third level – prevention of beyond design basis accidents by the safety systems:

- a) prevention of progression of initiating events into design basis accidents and design basis accidents into beyond design basis accidents by the use of safety systems;
 - b) mitigation of consequences from non-prevented accidents by localization of the radioactive substances released;
4. fourth level – management of beyond design basis accidents;
- a) prevention of beyond design basis accidents progression and mitigation of their consequences;
 - b) protection of reactor containment integrity during a beyond design basis accident and maintaining its availability;
 - c) plant recovery in a safe stable state with termination of the fission process, ensuring of long term fuel cooling and confinement of radioactive substances within the established boundaries;
5. fifth level – development and implementation of on-site and off-site emergency plans.
- (3) The defence in depth concept shall be applied at all stages of safety related plant activities. Measures to prevent adverse events at first and second levels of defence have priority over other measures related to ensuring safety.

Art. 5. (1) The operating organization of a NPP shall ensure plant safety, including to implement measures for accident prevention and mitigation of their consequences, for accounting and control of nuclear material, for physical protection of the plant and nuclear material, for radiation monitoring of the environment within the radiation protection area and the monitored area.

(2) The operating organization shall ensure that the NPP is used only for the purposes it was designed and constructed for.

(3) The operating organization bears the full responsibility of ensuring safety, including when other entities implement activities or provide services to the NPP, as well as in relation to the activities of the specialized regulatory authorities in the fields of nuclear energy and ionising radiation.

Art. 6. (1) The entities who implement activities related to site selection, design, construction, commissioning and operation of a NPP shall be well-acquainted with the nature and magnitude of the impact that the activity they perform may cause on nuclear safety and radiation protection, and the possible consequences from violation or not full compliance with the requirements set by the legislative acts and the instructions.

(2) Development of safety culture within the entities under para. 1 entails:

1. for any activity that may impact safety, to conduct an appropriate personnel selection, training and qualification process;

2. to strictly follow the discipline having clear division of personal responsibilities among the management and the employees;

3. to develop and strictly adhere to instructions requirements, and periodically update them considering the own and internationally recognized operating experience.

Art. 7. (1) The operating organization shall develop, implement and maintain a quality assurance system for site selection, design, construction, commissioning and operation of the NPP, including for supervision of the activities of the entities working or providing services for the NPP.

(2) The entities who perform work or provide services related to safety for the NPP shall develop and implement quality assurance programs for the corresponding type of activity in compliance with quality assurance system of the operating organization.

Chapter Two
DESIGN BASIS AND SAFETY ASSESSMENT
Section I
Design Basis

Art. 8. The design basis shall specify the necessary capabilities of the plant to cope with all operational states and design basis accidents within the defined radiological limits for internal and external exposure of personnel and population and the limits for release of radioactive substances into the

environment. The design basis shall include design limits, plant operational states, safety classification of SSCs, important assumptions within the design process, and in individual cases, special methods of analysis.

Art. 9. The design limits shall at least include:

1. radiological and other technical acceptance criteria for all operational states and accident conditions;

2. criteria on protection of fuel cladding, including fuel temperature, departure from nucleate boiling, cladding temperature, fuel rod integrity (tightness) and maximum allowable fuel damage during any operational state and design basis accidents;

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

4. criteria on protection of the containment, including temperatures, pressure and leak rates, considering the necessary margins ensuring its integrity and leak tightness in case of extreme external events, severe accidents and combinations of initiating events.

Art. 10. (1) For all plant operational states, the annual individual effective dose of the population, received from internal and external exposure as a result of liquid and gaseous releases into the environment from all nuclear facilities situated on-site, shall not exceed 0.15 mSv.

(2) The annual individual effective dose of the population from internal and external exposure at the boundary of the radiation protection area and outside it shall not exceed 5 mSv over the first year following a design basis accident.

(3) For severe accidents, the limit of cesium-137 release in the atmosphere is 30 TBq that does not impose long-term restrictions for soil and water use in the monitored area. Combined release of other radionuclides different from cesium isotopes shall not in a long-term perspective, starting three months after the accident, provoke a greater hazard than the one identified for cesium release within the indicated limit.

(4) The frequency of a large radioactive release into the environment that requires undertaking of immediate protective measures for the population shall not exceed $1 \cdot 10^{-6}$ events per NPP per year.

Art. 11. (1) Initiating events for design basis accidents shall be defined in the design to identify the boundary conditions, according to which the SSCs important to safety shall be designed.

(2) Selection of postulated initiating events shall be based on the use of both deterministic and probabilistic methods.

(3) Ruptures of component vessels that are produced and operated in accordance with the most stringent nuclear safety requirements could not be included as postulated initiating events. Furthermore, it shall be proven in the design that the frequency of reactor pressure vessel degradation does not exceed 10^{-7} events per reactor per year.

Art. 12. (1) Postulated internal initiating events shall be grouped into different categories depending on their frequency of occurrence per calendar year. Grouping shall be based on the following four categories of plant states:

1. category 1 – steady and transient states during normal operation;

2. category 2 – anticipated operational occurrences, with frequency above 10^{-2} events per year;

3. category 3 – accidents of low frequency of occurrence, in the range between 10^{-2} and 10^{-4} events per year;

4. category 4 – design basis accident of very low frequency of occurrence, in the range between 10^{-4} and 10^{-6} events per year.

(2) A typical list of postulated initiating events and the categories of plant states to be considered in the safety analysis, are defined in the Attachment to this Regulation.

(3) The design shall consider initiating events resulting from possible human errors and probable combinations of internal and external events and hazards, based on realistic assumptions.

(4) Plant design shall take into consideration specific environmental conditions and loads to SSCs important to safety caused by the following internal events:

1. forces induced by breaks of high-pressure piping, as impingement forces and pipe whipping;
2. internal overflow and flooding due to leaks or breaks of pipes, pumps, valves;
3. internal missiles;
4. load drop;
5. internal explosion;
6. fire.

Art. 13. The NPP design shall take account of the following external events and site specific hazards:

1. extreme weather conditions;
2. earthquakes;
3. external flooding;
4. aircraft crashes;
5. hazards arising from nearby transportation and industrial activities;
6. sabotage;
7. electromagnetic interference.

Art.14. (1) In addition to the design basis, plant performance in beyond design basis accidents shall be assessed. A combination of analytical methods may be used for the evaluations applying best estimate assumptions.

(2) The list of beyond design basis accidents with no severe core degradation shall include the following accidents, if not prevented by the reactor inherent safety features:

1. total loss of on-site and off-site power (station blackout);
2. anticipated transients without scram;
3. multiple SG tube ruptures (for pressurized water reactors);
4. total loss of feedwater;
5. total loss of component cooling water (service water);
6. loss of coolant accident combined with complete loss of an emergency core cooling system (high or low pressure);
7. loss of main heat sink;
8. uncontrolled concentration decrease of the soluble neutron absorber in the coolant (for pressurized water reactors);
9. uncontrolled level drop in the reactor during refuelling or shut down for maintenance (for pressurized water reactors);
10. loss and long term unavailability of safety systems after a postulated initiating event requiring their availability.

(3) If the analysis of severe accident consequences does not confirm the implementation of the criteria under Art. 10, para. 3 and 4, the design shall consider additional technical measures for severe accident management aimed at mitigation of the consequences.

Art. 15. (1) All SSCs (including control system software), important for safety, shall be identified and classified in safety classes. They shall be designed, constructed, mantled, tested, operated and maintained such that their quality, including their reliability, is commensurate with a classification plan.

(2) The classification of SSCs shall be based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment with account taken of the following factors:

1. the safety function performed;
 2. the consequences if they fail to perform the safety function;
 3. the frequency of performance of the safety function;
 4. the required period of performing the safety function.
- 3) The classification plan shall identify for each safety class:
1. the appropriate standards for design, manufacturing, construction and inspection;

2. the degree of redundancy, need for emergency power supply, qualification for operation in specified plant operational states and accident conditions;
3. the availability or unavailability status of SSCs that shall be considered in the deterministic safety analysis;
4. the quality assurance provisions.

Art. 16. (1) Structures, systems and components, important to safety, shall withstand the conditions of postulated initiating events with sufficient margins.

(2) To identify the cases, where application of the principles of diversity, redundancy and independence is needed to achieve the necessary reliability, the potential for common cause failures shall be analysed and considered in the plant design.

(3) The failure of a SSC belonging to one safety class shall not cause a failure of other SSC in a higher safety class. The auxiliary systems, supporting SSCs important to safety, shall be classified in the same safety class.

Art. 17. (1) Design and selection of materials for safety important SSCs, shall consider the effects on their characteristics and performance of plant operational conditions and design basis accidents.

(2) Qualification procedures of safety important SSCs shall be specified in the design to secure fulfilment of assigned functions throughout their operating lifetime, taking into account possible impacts and environmental conditions, which may be expected in all operational states and accident conditions (vibration, temperature, pressure, jet impingement, electromagnetic interference, aging, irradiation, humidity and probable combination thereof).

(3) Working conditions of safety important structures' and systems' components shall be simulated by tests and analysis or by a combination of both methods.

Section II. Safety Assessment

Art. 18. (1) Plant safety shall be analysed using deterministic and probabilistic methods to verify and confirm the established design basis and the effectiveness of defence in depth arrangements.

(2) Computer codes, analytical methods and plant models to be used in the safety analysis shall be verified and validated. uncertainty of the results shall be quantified.

Art. 19. (1) Deterministic safety analysis shall include:

1. confirmation of operational limits and conditions compliance with the design assumptions for normal operation;
2. identification of the postulated initiating events characteristics, including those specific for the selected site;
3. analysis and assessment of postulated initiating events' progression;
4. comparison of analysis results under item 3 against the radiological acceptance criteria and the other design limits;
5. confirmation of the design basis;
6. substantiation of plant capabilities to manage all anticipated operational occurrences and design basis accidents through a combination of safety systems' automatic actions and required actions of the operating personnel.

(2) In the analysis of the postulated initiating events:

1. initial and boundary conditions shall be specified in a conservative way;
2. applicability of analytical methods and assumptions, and degree of conservatism used shall be verified;
3. only safety class SSCs qualified to operate in the relevant conditions shall be used to reach and maintain the reactor in a safe shutdown state;
4. cases of an PIE independent single failure of an active or passive safety systems' component having most adverse effect on the accident progression or a single, independent from the PIE human error

shall be applied to plant states categories 2, 3 and 4. In addition, any latent (undetected) failures challenging the safety limits shall be considered;

5. a single failure of a passive component may not be considered for plant states of category 2;
6. manual action from the main control room shall be assumed to take place, at the earliest, 30 minutes after the first significant information is given to the operator, for plant states of categories 2, 3 and 4;
7. most effective rod shall be considered stuck in upper position for plant states of categories 2, 3 and 4;
8. unavailability of the systems for normal operation shall be assumed in a way, that results in the most adverse effect for the analysed initiating event for categories 3 and 4;
9. any failure, occurring as a consequence of an initiating event shall be included in the analysis;
10. impact of the uncertainties, that significantly influence to results, shall be considered.

Art. 20. Analyses results under Art. 19, para. 2 shall demonstrate:

1. maintaining fuel cladding integrity - in states of categories 1 and 2;
2. maximum fuel cladding temperature shall not exceed 1200 degrees centigrade, the local oxidation of the cladding shall not exceed 17% of the initial thickness, and the reacted amount of zirconium shall not exceed 1% of its mass in the reactor core, in cases of loss of coolant accidents of categories 3 and 4.

Art. 21. (1) Probabilistic safety analysis shall be carried out with the objective to:

1. systematically analyse the compliance with the basic safety criteria;
2. demonstrate a balanced design where each postulated initiating event has a proportional impact upon the overall plant risk and the safety is ensured mainly by the first two levels of defence in depth;
3. provide confidence that any impact of small deviations in operational parameters is prevented, when this could lead to aggravation of plant behaviour;
4. assess the frequencies of severe core damage and large radioactive releases to the environment;
5. evaluate the frequencies and the consequences of the external events specific to the site;
6. identify SSCs that require design improvements or changes in operational procedures, leading to decrease of severe accident frequency or mitigation of their consequences;
7. assess the emergency operating instructions.

(2) The probabilistic safety analyses shall include:

1. all modes of operation, all postulated initiating events, including internal fire and flooding, severe weather conditions and seismic events;
2. all possible important dependencies (functional dependencies, area dependencies and other interactions and impacts, leading to common cause failures);
3. uncertainty analysis or sensitivity analysis of the results;
4. realistic modelling of plant response, taking into account operator actions in accordance with operational and accident instructions;
5. human error analyses, taking into account the factors which can influence the performance of operating personnel in all operational states and accident conditions.

(3) Probabilistic safety analyses shall be performed according to state-of-the-art methodology, documented and maintained according to the quality management program of the operating organization.

(4) Probabilistic safety analyses shall be used to support the deterministic assessments in the decision making for plant design and operation, for assessment of necessary changes of SSCs, operational limits and conditions, operating and emergency operating procedures and training programs of the operating personnel.

Art. 22. (1) Existing plant design and operations shall be periodically reviewed in the light of the operational experience and new safety significant information to identify deviations from current requirements and internationally recognized operational experience. Decisions on design modifications,

improvements, or other measures shall be made depending on the safety significance of identified deviations.

(2) The scope of the periodic safety review shall as a minimum include the following areas of review:

1. site characteristics considered in the design and if necessary, their re-evaluation on the basis of new data obtained or new methods used;
2. plant design as built and actual condition of systems, structures and components taking into account implemented modifications, ageing and other effects that impact safety and plant lifetime;
3. current safety analyses methods and applicable new safety requirements;
4. operating experience during the review period and the effectiveness of the system used for experience feedback;
5. organizational arrangements for operation;
6. safety performance indicators and effectiveness of safety and quality management;
7. staff number, level of training and qualification;
8. emergency preparedness, and
9. radiological impact on the environment.

(3) Periodic safety review shall use systematic and documented methodology, taking into account deterministic and probabilistic methods. Conclusions shall be drawn to justify practically achievable improvement measures considering the interconnections between the identified deviations.

Art. 23. (1) A comprehensive analysis of the risk from fires shall be carried out to substantiate the adequacy of fire protection arrangements. Personnel performing the analyses shall have experience both in technological system analysis and fire protection.

(2) The analysis under para. 1 shall postulate a single fire in an area with combustible materials, consideration of an additional failure (independent of the fire) and their consequences, such as internal explosion or internal flooding. The analysis shall identify the fire spread in each area, the impact on the SSCs located in this area and in any other area, which may be affected by the failures or the environmental conditions as a result of the fire. The consequences of failures or improper operation of the fire detection and extinguishing systems during normal operation shall be analysed as well.

Art. 24. (1) The design basis, the technical and organizational measures providing the implementation of the defence in depth concept and the safety assessment shall be documented in preliminary, intermediate and final safety analysis report associated with the authorization process under the Act on the Safe Use of Nuclear Energy (ASUNE).

(2) The operating organization shall maintain updated the safety analysis report to correspond to the actual status of changed SSCs important to safety, to the new transient and accident analyses and the current safety requirements.

Chapter Three
SITE CHARACTERISTICS
Section I.
General Requirements

Art. 25. Suitable for hosting of a NPP are sites that meet the following conditions:

1. there is full compliance with the legislation on environmental protection and with the requirements on radiation, fire and physical protection;
2. the site is located within the boundaries of a tectonic block, unaffected by capable faults and geodynamic zones;
3. peak ground acceleration at the free field during earthquake determined with annual probability of exceeding 10^{-4} , is less than 1 m/s^2 ;
4. site flooding is impossible.

Art. 26. Unfavourable for hosting of a NPP are the sites located on/in:

1. territories of operating artesian pools and of intensive exchange between underground and surface water;
2. tornado endangered regions;
3. territories with winds of average velocity larger than 12 m/s;
4. territories exposed to effects of tsunami-waves, disastrously high water levels or floods;
5. territories that might be flooded by a wave resulting from dam failures;
6. territories where NPP would be located on the side of the prevailing wind direction with respect to already existing factories and towns;
7. regions where the peak ground acceleration at the free field during earthquake determined with annual probability of exceeding 10^{-4} , is greater than 1 m/s^2 ;
8. territories with registered contemporary differential movements of the earth crust (vertical with speed higher than 10 mm per year, horizontal – over 50 mm per year);
9. linear geodynamic zones within the boundaries of which are recognized differential tectonic movements over the recent one million years with a velocity gradient of the Quaternary movements equal to or greater than 10^{-9} (without revealing of capable faults on the ground surface);
10. zones of tectonic cracks;
11. regions with developing karst (thermal karst);
12. territories with disused mines or other excavation works apart from those that can be used for situating of underground NPP;
13. regions with potentially active or dormant landslides, or other dangerous slope processes;
14. regions with marked thunderstorms foci – within which boundaries the likelihood of intensive thunderstorm activity is considerably higher in comparison to that of the surrounding territory;
15. coastal and estuary floodplains where the displacement speed of the cutting line and the upper plain of the abrasive ledge is more than 1 m per year;
16. slopes with natural angle of steepness 15° or more;
17. sites where groundwater has been found at less than 3 m beneath the surface of the chosen zero level in earth layers of 10 or more meter thickness, having a filtration ratio equal to or more than 10 m per 24 hours;
18. regions abounding in structurally and dynamically unstable earth layers and earth layers having a deformation module below 20 MPa;
19. regions with possible externally induced fires, hazardous for the NPP;
20. territories enclosing facilities within their boundaries, including military facilities, with possibility of release of flammable, toxic and corrosive substances or other impacts on NPP, including blast shock waves and missiles in case of fire and explosion;
21. territories with average population density exceeding 100 people per square km in the monitored area, calculated for the NPP operating lifetime.

Art. 27. Location of a NPP in unfavourable regions and zones characterized by the presence of dangerous processes, phenomena and factors of natural or human induced origin could be accepted if technical and organizational measures to ensure safety have been implemented.

Art. 28. NPP location is not allowed:

1. on territories where the location is forbidden by a legislative act, or on sites that do not meet the legislation on environmental protection, the requirements on radiation, fire and physical protection, or violate any other requirements specified in a legislative act;
2. on sites directly situated on capable faults;
3. on sites where the peak ground acceleration at the free field during earthquake determined with annual probability of exceeding 10^{-4} , equals or exceeds 4 m/s^2 ;
4. in the regions of karst (thermal karst), syphusion and karst-syphusion processes;
5. on sites in zones where avalanches and mudflows appear;
6. sites exposed to impact of tsunami waves;
7. in mine development which stability cannot be ensured over the whole operating lifetime of an underground NPP.

Section II
Investigation of Natural and Human Induced Factors for Site Selection

Art. 29. Engineering surveys and investigations of natural processes, phenomena and factors having potential impact on NPP safety shall be conducted for the region and the site for situating a NPP:

1. the following tectonic characteristics shall be defined:
 - a) location of faults, of potential earthquake foci zones and of geodynamic zones with respect to the NPP site, indicating the orientation and boundaries of potentially dangerous fault zones;
 - b) amplitudes, speed and gradients of the latest and contemporary movements of the earth crust, parameters of potential dislocations;
 - c) characteristics of capable fault areas (geometric schemes, dislocation amplitudes and directions along the faults, data of the latest activity known);
2. within the NPP site boundaries, the following shall be identified:
 - a) characteristics of the input ground motion in earthquakes of intensity of design basis earthquakes (of seismic level 1) with frequency of 10^{-2} events per year and safe shutdown earthquake (of seismic level 2) with frequency of 10^{-4} events per year at the zero level of the site;
 - b) the hazard of landslide displacements of the slopes considering the ground layers conditions and seismic motions with an intensity up to safe shutdown earthquake inclusive and accounting also for the impact of ground-waters, tectonic characteristics, contemporary geodynamic processes;
 - c) the possibility of development of karst (thermal karst), syphusion and karst-syphusion processes and their impact on NPP safety;
 - d) the availability of specific earth layers (biogenic, collapsing, swelling, salted, alluvial, human induced), their thickness and physical-mechanical properties (deformation modules, strength characteristics, etc) and their impact on the non-uniform subsidence beneath the NPP structures, the reactor compartments inclination during earthquakes with an intensity up to safe shutdown earthquake inclusive;
 - e) the zones of water saturated disconnected earth layers yielding to self-liquefaction when exposed to seismic impacts with an intensity up to safe shutdown earthquake inclusive;
 - f) the impact on NPP safety of the groundwater level uplift and flooding the site as a result of spreading of underground water uplift coming from dams, filtration of irrigated lands, water flows, precipitation, snow melting;
 - g) tornado intensity, the peak tangential values of the periphery speed and the speed of the tornado progressive motion; the pressure drop between the tornado periphery and the centre;
3. for a NPP site shall be defined the maximum water level and the duration of possible flooding due to rainfall, intensive snow melting, high water level in water basins, river blocking by ice, avalanche and slide; for the NPP site shall be evaluated the characteristics of possible maximum run off floods from watercourses with a frequency of 10^{-4} events per year combined with a high tides and waves caused by winds;
4. for a 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 evaluated considering the seismic tectonic conditions, the shore configuration, landslides and collapse in the water;
5. for a NPP site, the impact on safety of other processes, phenomena and factors of natural origin shall be determined (hurricane, extreme rainfalls, air and water temperatures, icings, thunderstorms, dust-storms and sand-storms, erosion of river and water basins banks).

Art. 30. (1) The region to locate a NPP and the respective site shall be investigated to identify sources of potential human induced hazards. It is not allowed to neglect sources of human induced hazards with a frequency of occurrence greater than or equal to 10^{-6} events per year.

(2) Sources of human induced hazards shall be characterized by possible accidents causing explosions, fires, and release of explosive, toxic and corrosive substances.

(3) The impact on NPP safety shall be analysed from all stationary and mobile explosives, including all industrial facilities for production, processing, storage or transportation of chemical and

explosive substances and ammunitions warehouses. The impact of the most dangerous explosion shall be determined and NPP safety shall be justified considering the detonation and the follow-up consequences of the explosion in terms of ground layers shaking, missiles and local migration conditions of the gaseous cloud.

(4) Impact on NPP safety shall be analysed of all stationary and mobile sources of accidental release of chemically active substances, including industrial facilities where processing, usage, storage and transportation of toxic and corrosive substances is being performed.

(5) Characteristics of the impact on the NPP and respective probabilities shall be determined for events induced by:

1. explosions and fires, as well as release of explosive, inflammable, toxic and corrosive gases and substances from industrial facilities, ground and water transportation facilities;
2. airplane crashes;
3. floods, including those resulting from failures of dams located upstream of the NPP site, or due to rainfall, avalanches and snow melting;
4. accidents of water transportation facilities and in shore harbour zones occurring together with explosions and fires, chemically dangerous release, if the NPP is situated on the shore;
5. electromagnetic emissions (fields);
6. external fires (forest areas, peateries, burning fluids);
7. deformations and other factors arising on developing underground resources deposits, carrying out excavation works, including tunnel construction, mines and quarries exploitation and their accidental destruction;
8. water level fluctuations in the NPP water supply source.

Art. 31 (1) NPP impact on the population and the environment shall be considered during site selection.

(2) The aerologic, hydrometeorologic, hydrogeologic and geochemical conditions of radionuclide dispersion, migration and accumulation, and also the natural radiation background shall be studied in the monitored area and prediction shall be made of conditions behaviour 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, atmosphere stability, precipitation and fogs in the region of the NPP.

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

1. possible radioactive contamination of drainage and groundwater;
2. radionuclide physical and chemical properties;
3. kinetics of geochemical reactions and possible change in the mineralogical composition of layers;
4. lithological composition and thickness of water-bearing and water-tight layers, the earth layers in the weathering zone and the soil layer;
5. sorption capacity of the embankment sediments, earth layers and soil layer with respect to radionuclides and hazardous chemical substances;
6. direction and speed of contaminated flows towards the release places (drain-pipes, water basins, water output wells, etc.);
7. characteristics and stratification of water-bearing horizons;
8. hydraulic connections between underground and surface waters;
9. characteristics of water basins, hydro facilities, water consumption data, water levels and flow rates, river stream velocity, possible mechanisms of radionuclide transport and deposition.

(5) For NPP site selection, the radiological consequences for all operational states and accident conditions shall be substantiated and technical and organizational measures shall be planned to ensure the safety of the population. The radiological consequences assessment for the operational states shall be conducted by applying probabilistic distribution of the atmospheric dispersion parameters typical for the

region. Assessment of radiological consequences in emergencies shall be conducted for the most adverse weather conditions specific to the region of the NPP.

(6) The design shall justify preventive measures for radiation contamination of water areas with industrial utilization for all operation states and shall provide for protective measures from contamination of these areas in emergencies.

(7) Consequences to the population and the environment in the monitored area shall be determined of possible radiation impact from an accidental radioactive release during NPP operation, considering the following:

1. results of the assessment of the radiation conditions;
2. characteristics of the water output facilities in the monitored area;
3. characteristics of water basins important to fishing, fish population reproduction and other biological resources in the monitored area;
4. data on the existing and planned population distribution in the region of siting of the NPP;
5. forecast of the amount of radionuclides in the farm products.

Chapter Four
SAFETY REQUIREMENTS FOR DESIGN OF NPP AND PLANT SYSTEMS
Section I
General Requirements to NPP

Art. 32. (1) NPP design shall be based on the application of the defence in-depth concept under Art. 3 and aims at preventing as far as practicable:

1. challenges to the integrity of physical barriers;
2. failure of a physical barrier when challenged (under item 1);
3. failure of a physical barrier as a consequence of a failure of another barrier.

(2) For all operational states and accident conditions the NPP unit shall be capable of performing the following fundamental safety functions:

1. control of the reactivity;
2. removal of the heat from the reactor core;
3. confinement of radioactive substances within the established boundaries.

(3) Design technical solutions, technologies and procedures shall be determined and justified in compliance with the achievements of science and technology and with the internationally recognized operational experience.

Art. 33. Safety systems and SSCs important to safety shall perform the following safety functions:

1. to prevent unacceptable reactivity transients;
2. to maintain the reactor in safe sub-critical condition;
3. to shutdown the reactor to prevent anticipated operational occurrences, which lead to design basis accidents, and to mitigate the consequences of design basis accidents;
4. to maintain sufficient reactor coolant inventory to ensure core cooling during and after design basis accidents not involving failure of the reactor coolant pressure boundary;
5. to maintain sufficient reactor coolant inventory to ensure core cooling during and after all postulated initiating events;
6. to remove heat from the core after a failure of the reactor coolant pressure boundary in order to limit fuel damage;
7. to remove residual heat in appropriate operational states and accidents with the reactor coolant pressure boundary intact;
8. to transfer heat from the safety systems to the ultimate heat sink;
9. to ensure necessary services as a support function for the safety systems;
10. to maintain acceptable integrity of the fuel cladding in the reactor core;
11. to maintain the integrity of the reactor coolant pressure boundary;

12. to limit the release of radioactive substances from the reactor containment in accident and post accident conditions;

13. to limit the radiation exposure to the personnel and public in and following design basis accidents and selected severe accidents that release radioactive material from sources outside of the reactor containment;

14. to limit the discharge of liquid and gaseous radioactive substances below prescribed limits in all operational states;

15. to maintain the necessary conditions of the working environment for the operation of safety systems and for habitability for personnel, to allow performance of operations important to safety;

16. to control the radioactive releases from irradiated fuel transported or stored outside the reactor core (but on-site), in all operational states;

17. to remove decay heat from irradiated fuel stored outside the reactor core, but within site boundaries;

18. to maintain sub-criticality of fuel stored outside the reactor core, but on- site;

19. to prevent or limit the consequences of a failure of a SSC, which unavailability may cause degradation of a safety function.

Art. 34. SSCs important to safety shall provide for 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 in case of natural and human induced external events considered in the design.

Art. 35. (1) Design of SSCs important to safety shall preferably consider solutions applying the passive principle of operation, the fail-safe principle and the inherent safety features (self-control, heat inertia and other natural processes).

(2) The design shall provide for technical means aimed at precluding human errors or limiting their consequences, including maintenance of SSCs important to safety.

Art. 36. (1) Design shall, to the extent practicable, consider the diversity principle, the self-checking capability in safety systems and techniques for precluding interference of individual SSCs.

(2) Combining safety functions with normal operation functions shall not lead to violation of the safety requirements and to reduction of the safety systems reliability. The design shall substantiate the multipurpose use of safety systems and their components.

(3) Safety systems shall operate so that any initiated actuation shall lead to complete fulfilment of the safety functions. Recovery of the safety systems in initial state shall require consecutive actions of the operating personnel.

Art. 37. (1) SSCs important to safety, their structure, layout and operational condition shall provide possibility for testing, maintenance, repair, inspection and control, over the plant operating lifetime, without significant reduction in their functional availability. If a SSC important to safety cannot be tested and inspected over its operation to a sufficient extent to detect potential undiscovered failures, their reliability shall be ensured by alternative method, or the design shall take account of higher failure rate.

(2) Prior the beginning of plant commissioning, technical means, programs and methodologies shall be verified, which are needed for:

1. inspection of SSC functional capability (including those inside the reactor) and replacement after their design lifetime;

2. functional tests of systems intended to prove their design characteristics;

3. verification of signal sequences and actuation of systems and components, including of emergency power supply;

4. in-service inspection of the base metal and weldings of structures and pipelines;

5. verification of the metrological characteristics of measuring channels for compliance with design requirements.

Art. 38. (1) SSCs important to safety shall be designed, located and protected so as to ensure, in case of fire, long term fulfilment and maintaining of the safety functions under Art. 32, para. 2 and control over the plant status.

(2) Fire protection measures shall ensure defence in depth through providing prevention of fire starting and expansion, localization of fires propagation and the consequences thereof. To achieve those objectives:

1. building structures shall be conservatively designed as fire resistant with consideration of internal and external fires;

2. internal structures and segregating components shall be composed of non-combustible materials to the maximum practical extent;

3. fire load shall be kept at the practically achievable minimum by using non-combustible materials, where applicable, and in the other cases – of not readily combustible materials;

4. the unit shall be subdivided into fire resistant areas and compartments through fire resistant walls with appropriate fire resistant limit, capable of preventing spread of heat and smoke from fires considered in the design;

5. characteristics of fire-alarms and fire-extinguishing systems (reliability, independence, capacity and qualification) shall be selected with consideration of the fires risk resulting from the analysis, required under Art. 23;

6. arrangements shall be made for successful fire extinguishing, such as external and internal water supply, access and escape routes to the relevant buildings and structures.

Art. 39. For management of severe accidents, plant design shall provide for the use of:

1. instrumentation, qualified for severe accident conditions, to provide the main control room with information allowing timely evaluation of plant status;

2. technical means for containment isolation, overpressure protection, management of combustible gases temperature and concentration;

3. technical means for prevention of high-pressure core melt scenarios and containment melt through.

Art. 40. (1) 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) (amended SG No. 46/2007 and SG No. 53/2008) At least one emergency control centre shall be provided in the design, supplied with devices and systems for communication with the main and supplementary control rooms, as well as with the local municipal authorities and with the authorities of the executive power competent for emergency response – the Nuclear Regulatory Agency and the Ministry of Emergency Situations. The centre shall receive information on unit's status during the phases of accident progression and on the radiological conditions at the NPP site and its surroundings.

(3) The communication systems and devices of the facilities under paragraphs 1 and 2 shall be maintained available, periodically tested and the documentation shall be kept updated.

Section II

Reactor Core – Structure and Characteristics

Art. 41. (1) Reactor core and associated reactor coolant system, reactor control and protection safety systems shall be designed with appropriate safety margins to ensure that the specified design limits for fuel damage are not exceeded at all operational states and design basis accidents with account taken of:

1. design operating modes and their passing;

2. thermal, mechanical and irradiation degradation of the core components;

3. physical-chemical interaction of core materials;

4. limiting values of thermal hydraulic parameters;

5. vibrations and thermal cycles, material fatigue and aging;

6. impact of coolant additives and radioactive fission products on the corrosion of fuel claddings;

7. irradiation and other impacts that deteriorate mechanical characteristics of core materials and fuel cladding integrity.

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

Art. 42. To ensure safe shutdown of the reactor, to maintain the reactor subcritical and to ensure core cooling, the reactor core and associated internal components located within the reactor vessel shall be designed and mounted in such a way as to withstand the static and dynamic loads expected in all operation states, design basis accidents and external events considered in the design.

Art. 43. (1) Reactor core and its elements 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 design basis accidents.

(2) Design shall prove that in all design basis 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 melting is excluded by insertion of the control rods. With respect to beyond design basis accidents, conditions for possible fuel melting or exceeding the specific energy threshold causing fuel damage shall be specified.

(3) For all design basis accidents and for beyond design basis accidents under Art. 14, para. 2, changes in core geometry shall be limited thus ensuring conditions for long-term fuel cooling.

(4) The reactivity coefficients of coolant density, of coolant-moderator and fuel temperature, and of reactor power, shall be negative within the whole range of the reactor coolant system parameters for all operational states and design basis accidents.

(5) Design shall ensure minimization of possibilities for recriticality and reactivity excursions following postulated initiating events.

(6) Design of the reactor core shall reduce demands on the system for control of the neutron flux (distribution, levels and stability within specified limits) in all operational states.

Art. 44. Reactor core and associated coolant, control and protection systems shall be designed to enable adequate inspection and testing throughout the service lifetime of the plant.

Art. 45. The frequency of core damage in a severe accident, determined on the basis of probabilistic safety analysis, shall be sufficiently lower than 10^{-5} events per NPP per year.

Art. 46. The characteristics of nuclear fuel, of reactor structures and of reactor coolant system components (including the coolant clean up system) shall exclude recriticality in severe accidents, considering the operation of the other systems.

Art. 47. Fuel elements and assemblies, taking into account the uncertainties in data, calculations and fabrication, shall be designed to withstand irradiation and reactor core conditions in combination with all degradation processes that can occur in all operation states, such as:

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

Section III Reactor Shutdown Systems

Art. 48. Design shall make provisions for at least two independent reactor shutdown systems based on different (diverse) principles of operation with each alone being capable of rendering and maintaining the reactor core subcritical on the assumption of a single failure or a personnel error, from all operational states and in design basis accidents, with maximum value of the effective multiplication factor.

Art. 49. At least one of the reactor shutdown systems shall have in all operation states and design basis accidents:

1. effectiveness, sufficient for rendering and maintaining the reactor core subcritical, considering possible reactivity increase;

2. fast-acting capabilities, sufficient for rendering the reactor core in subcritical state without violation of the design basis accidents limits for damage of fuel elements (considering the operation of the emergency core cooling system).

Art. 50. (1) At least one of the systems shall be capable of performing the emergency shutdown function. On the assumption of a failure of the most effective rod, the emergency shutdown system components shall have:

1. fast-acting capabilities, sufficient for rendering the core to subcritical state without violation of the safety limits in anticipated operational occurrences;

2. effectiveness, sufficient for rendering and maintaining the core subcritical in anticipated operational occurrences and design basis accidents.

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

(3) The emergency shutdown system shall have at least two independent groups of rods. The 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.

Art. 51. All emergency shutdown system rods shall have intermediate position indicators, end position annunciators and limit switches (end breakers), actuated where practicable directly by the rod. Other reactor shutdown means shall be equipped with position indicators.

Art. 52. When combining reactivity and power control functions with emergency shutdown functions, design shall include developed and justified procedure for system operation, giving a priority to the emergency shutdown functions.

Section IV Reactor Coolant System

Art. 53. (1) Reactor coolant system, its associated auxiliary systems, and control and protection safety systems shall be designed with sufficient margins to ensure that design limits of reactor coolant pressure boundary are not exceeded in all operational states.

(2) Design shall include devices to reduce the pressure in the reactor coolant pressure boundary, the operation of which shall not lead to unacceptable releases of radioactive substances in all operational states and design basis accidents.

Art. 54. (1) Components, pipelines and supporting structures of the reactor coolant system shall withstand all anticipated static and dynamic loads and temperature effects to the components in all postulated initiating events.

(2) Materials to be used for fabrication of the components of the reactor coolant system shall be selected so as to minimize 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-lifetime under the effects of erosion, creep, fatigue and chemical environment.

Art. 55. Reactor pressure vessel and pressure tubes shall be designed and constructed to be of the highest quality with respect to material selection, design standards, capability of inspection and fabrication.

Art. 56. Design of the components contained inside the reactor coolant pressure boundary shall be such as to minimize the likelihood of failure and associated consequential damage to other items of the primary coolant system in all operational states and in design basis accidents.

Art. 57. 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. Implementation of a material surveillance program for the reactor coolant pressure boundary shall control the effects on structural materials of various factors such as irradiation, stress corrosion cracking, embrittlement, and ageing and particularly in locations of high irradiation, and others.

Art. 58. Provisions shall be made in the design to regulate coolant inventory and pressure with adequate capacity for all operational states.

Art. 59. Design shall provide for systems to cleanup reactor coolant from radioactive substances, including activated corrosion products and fission products. Capacity of the necessary systems 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 practical.

Section V Heat Removal System

Art. 60. (1) Plant design shall provide for reliable systems to remove, to an ultimate heat sink, the residual heat from the core and from SSCs important to safety, in all operational states and design basis accidents. All systems that contribute to the heat transfer (by conveying heat, by providing power or by supplying fluids to the heat transport systems) shall be designed according to their contribution to the heat-transfer function.

(2) Reliability of the systems shall be achieved by the use of proven components, redundancy, diversity, physical separation and isolation.

Art. 61. (1) Natural phenomena and human induced events, specific to the NPP site, shall be taken into account in the design of the systems and in the possible choice of diversity in the ultimate heat sinks.

(2) Adequate consideration shall be given to the residual heat removal from the reactor core and cooling of the localization system components in case of a severe accident.

Section VI Control of the Technological Processes

Art. 62. Any NPP unit shall be provided with the following means to control and monitor the systems for normal operation and the safety systems:

1. main control room (MCR);
2. supplementary control room (SCR);
3. control systems for normal operation;
4. control safety systems;

5. independent means for information registration and storage.

Art. 63. (1) The MCR shall provide possibilities for undertaking measures to maintain the plant in a safe state or recover such state if needed in all operational states and design basis accidents.

(2) Design shall demonstrate that arrangements made sufficient to secure MCR personnel health and availability, as well as proper functioning of the MCR, in all operational states and accident conditions.

(3) Layout of instrumentation and control devices and the way of presenting the information shall be such that the operating staff at the MCR be able to clearly and quickly identify plant status and behavior, adherence to the operational limits and conditions, identification and diagnosis of the safety system automatic actuation and operation.

(4) MCR design shall provide for:

1. instrumentation to control the fission process, in all core states, and conditions in normal operation, including in subcritical state during refuelling;

2. position indicators of the reactivity control devices, automatic control of soluble neutron absorber concentration, and status indicators of all means for reactivity control;

3. a system for information support to the operators;

4. a safety parameters display system of the reactor installation.

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

(6) The changes in normal operation conditions, which may affect safety, shall be accompanied by audible and visible indication.

Art. 64. (1) The 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;

4. control of the status of the reactor installation.

(2) 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 all postulated initiating events.

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

Art. 65. (1) Control systems for normal operation shall control and regulate the technological processes, in all operational states, in conformity with the design specified indicators for quality, reliability and metrological characteristics, and shall encompass:

1. means for collecting, treating, documenting and storing of information, which to be sufficient for timely and unambiguous identification of the initiating events for anticipated operational occurrences and accidents, their progression, factual algorithms of operation of the safety systems and the components, which failures are initiating events for design basis and beyond design basis accidents, deviations from the design algorithms and personnel actions;

2. means for automatic control of reactor coolant activity, liquid and gaseous effluents to the environment, and radiation monitoring of plant compartments, and radiation protection and monitored areas, in all operational states and design basis accidents;

3. means for automatic control of the conditions for safe storing of nuclear fuel and radioactive waste, and for signalisation in case these conditions are violated;

4. means and methods for identification of the locations and quantities of coolant leakages;

5. means for reliable group and individual communications between the MCR, SCR and field operators.

(2) Control systems for normal operation shall ensure the most favourable conditions to the operating personnel to take the correct decisions for plant management.

- (3) In the design of computer based control systems for normal operation:
1. special standards and proven practices shall be used in development and verification of the hardware, and especially of the software;
 2. development and verification process shall be conducted in compliance with a quality assurance program;
 3. level of reliability assumed in the safety analysis shall include a specified conservatism to compensate for the inherent complexity of the technology.

Art. 66. (1) Control safety systems shall be designed to:

1. initiate automatically the operation of appropriate systems, including systems for reactor shutdown, in order to ensure that specified design limits are not exceeded as a result of anticipated operational occurrences;
2. detect the symptoms for design basis accidents and automatically actuate other safety systems necessary to limit the consequences within the design basis;
3. lock the switch-off capability of the operating personnel for at least 30 minutes after an automatic actuation;
4. be capable of overriding unsafe actions of the control systems for normal operation.

(2) Design shall provide possibilities for manual remote actuation of the safety systems – and for the isolation components at their location. A failure in automatic actuation circuits shall not impede the remote actuation and the implementation of the safety functions. For remote and manual actuation, the operation of a minimum number of control components shall be sufficient.

(3) Possibilities for erroneous operation of the control safety systems shall be minimized. Design of the remote control of the safety systems shall ensure that at least two logically connected operator actions (two switches, buttons, etc.) are needed for their actuation.

Art. 67. The principles of redundancy, independence and diversity shall be applied in the design of control safety systems. Application of those principles shall result in such conditions that any single failure in control safety system does not affect its functionality and a protection against common cause failures is secured.

Art. 68. (1) Design of control safety systems shall provide for: continuous automatic diagnostics of the systems operability; periodical testing from MCR and SCR of system channels; and diagnosis of the technological components, which failures are initiating events for design basis and beyond design basis accidents.

(2) Any hardware and software failure and control safety system degradation shall lead to indication at the MCR and SCR and result in actions to ensure safety. In case that this is technically not feasible, methods and means shall be provided for periodical testing without reducing the functional availability of other safety systems and technological components, which failures are initiating events for design basis and beyond design basis accidents.

Art. 69. In addition to the requirements set in Art. 65, para. 3, design of computer based control safety systems shall fulfil the following requirements:

1. highest quality requirements and best practices shall be used for selection of the hardware and software;
2. the whole development process, including control, testing and design changes, shall be systematically documented and reviewed;
3. an assessment of the computer based system by an expert organization, independent of designers and suppliers, shall be undertaken;
4. the diversity principle shall be applied in order to ensure the necessary reliability of the system.

Art. 70. Design shall make provisions for self-dependent means that ensure registration and storage of the information necessary for accident investigation. These means shall be protected against

uncontrolled access and shall be operable in accident conditions. Design shall justify the scope of the information to be registered and stored.

Section VII Protection Safety Systems

Art. 71. Design shall provide for reactor protection systems to shutdown and maintaining the reactor subcritical and to remove the heat from reactor core. Redundancy, diversity and independence principles shall be applied to secure reliable performance of the safety functions in all postulated initiating events, on the assumption of a single failure independent of the initiating event.

Art. 72. The effectiveness of the emergency core cooling systems together with the technical means for leak detection of the reactor coolant system, the reactor inherent safety features, and the isolation capabilities, shall be sufficient to:

1. meet the specified in the design criteria for protection of the fuel cladding or of the fuel in design basis accidents;
2. preserve the geometry of the fuel and the reactor internal components in a condition that allows fulfilment of system's functions;
3. ensure long term core cooling.

Art. 73. (1) In case of an actuation and operation of any emergency core cooling system, measures shall be provided to prevent:

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

(2) Actuation of any protection safety system shall not lead to impair or loss of functions of another system.

Art. 74. Design shall include the possibilities for heat removal from the core during severe accidents.

Art. 75. Emergency core cooling system shall be designed in a way that allows conducting of periodic inspections of important components and periodic tests intended to confirm:

1. the integrity of the structure and the tightness of system components;
2. the functional capability and the operational characteristics of system active components, during normal operation;
3. the functional capability of the system under specified operational states.

Section VIII Localization Safety Systems

Art. 76. (1) Reactor installation design shall include localization systems to ensure fulfilment of the established criteria for radioactive releases to the environment. Localization systems shall perform their functions in all postulated initiating events and mitigate the consequences of beyond-design basis accidents including the assumption of a single failure independent of the initiating event.

(2) In establishment of confinement functions, provisions shall include a leaktight structure, systems and means for control of containment parameters, for containment structure isolation, and for reducing the concentration of fission products, hydrogen and other substances that could be released in the containment atmosphere during and after design basis and severe accidents.

Art. 77. (1) The containment structure and its components, including hermetic access doors, penetrations and isolation devices, shall be designed with sufficient safety margins on the basis of potential internal overpressure, underpressure and temperatures, dynamic effects such as missiles impact,

reaction forces, and the effects of other potential energy sources anticipated to arise as a result of design basis accidents. In calculating the necessary strength of the containment structure and its components, natural phenomena and human induced events shall be taken into consideration, as well as a combination of the effects of reactor coolant system break with maximum size and safe shutdown earthquake.

(2) Design shall include means for containment structure surveillance in all operational states and design basis accidents. Design shall make provisions for maintaining the integrity of the containment structure in the event of a severe accident with account taken of the effects of any predicted combustion of flammable gases.

Art. 78. (1) Containment structure and its components shall be designed and constructed to ensure structural integrity testing during commissioning and performing of periodic leaktightness tests over plant lifetime. Design shall specify tests requirements and the respective methods and means. Components located inside the containment shall retain their functional capability after the tests have been conducted.

(2) Design shall ensure possibilities to control containment radioactive leakages in case of a severe accident.

Art. 79. (1) Number of penetrations through the containment structure shall be kept to a practical minimum. All penetrations shall meet containment structure design requirements with account of possible mechanical, thermal, and chemical effects.

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

Art. 80. (1) To prevent radioactive releases outside the containment in case of a design basis accident, any containment penetrating line (part of the reactor coolant pressure boundary or directly connected to containment atmosphere) shall be reliably isolated by at least two isolation valves having independent automatic control, arranged in a series and located outside and inside the containment structure as close to the containment structure as practicable.

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

(3) Design shall include the arrangements to maintain the functionality of isolation devices during a severe accident.

Art. 81. (1) To secure personnel access to containment premises, provisions shall be made of lock and block doors as to secure at least one door in a locked position for all operational states and design basis accidents. The same requirements shall be applied when transporting components through the containment structure.

(2) Design shall include the arrangements to ensure capability of isolation devices to maintain their functionality in the event of a severe accident.

Art. 82. Containment design shall include measures and technical means to ensure sufficiently low pressure difference between the separate internal compartments not to challenge the integrity of pressure bearing structure or of other systems with confinement functions, taking into account the pressure and the possible effects resulting from design basis and severe accidents.

Art. 83. (1) To reduce the pressure and temperature in the containment in case of high-energy pipe breaks during design basis accidents, systems for heat removal shall be provided with the necessary reliability, redundancy and independence of the channels that ensure the required system effectiveness on the assumption of a single failure which is independent of the initial state.

(2) Design shall include the arrangements to ensure the capability of heat (generated as a result of a severe accident) removal from the containment.

Art. 84. (1) Systems shall be installed to control and cleanup containment atmosphere, and as necessary to control the concentration and remove fission products, hydrogen, combustible and other substances, which could to be released in the containment during and after design basis accidents.

(2) Systems for containment atmosphere cleaning up shall have adequate components' reliability and redundancy to ensure the required effectiveness of the system on the assumption of a single failure.

(3) Design shall include the arrangements to ensure control of concentrations of radioactive fission products, hydrogen and other substances that may be generated or released in the event of a severe accident.

Art. 85. Selection of coverings and coatings, and methods of their application on SSCs inside the containment, shall ensure the implementation of their safety functions and to minimize interference with other safety functions in the event of degradation of coverings and coatings.

Section IX Supporting Safety Systems

Art. 86. NPP design shall provide for supporting safety systems fulfilling auxiliary services on supply of safety systems with fluids and energy, and maintaining their operational conditions over a justified period of time in all operational states and design basis accidents.

Art. 87. (1) 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) Functional reliability of supporting systems shall be sufficient enough to meet the required reliability criteria of the respective safety system.

(3) Systems' design shall provide possibility for testing of their functional capability and for failure indication.

Art. 88. Fulfilment of supporting functions shall have priority over supporting systems own protections, if this will not aggravate safety consequences. Design shall specify the non-isolable own protections of the components of the supporting safety systems.

Art. 89. Design shall make provisions for fire alarm and fire-extinguishing systems to prevent fire-induced common cause failures in safety systems and to automatically fulfil the specified functions. Fire-extinguishing systems shall also be able to be manually actuated.

Section X Radioactive Waste Management

Art. 90. (1) 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 design basis accidents.

(2) Systems for management of liquid and gaseous radioactive release 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 personnel and the population, as well as radioactive releases to the environment in case of design basis accidents.

Art. 91. (1) NPP design shall include systems for handling and temporary storage of liquid RAW in a condition suitable for transportation and further treatment.

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

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

Art. 92. (1) NPP design shall ensure maintaining of the volume and activity of 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 the frequency of events that require significant decontamination measures.

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

Section XI Handling and Storage of Nuclear Fuel

Art. 93. SSCs for handling and storage of non-irradiated fuel shall be designed to:

1. prevent criticality by a sufficient margin, even under the most adverse states, by ensuring related 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. minimize the possibility of damage or unauthorized access to nuclear fuel;
5. prevent fuel assembly drop during transportation;
6. prevent the inadvertent dropping of heavy objects upon the fuel assemblies.

Art. 94. (1) SSCs for handling and storage of irradiated fuel shall be designed in compliance with the requirements of Art. 93 and additionally shall have the following:

1. reliable systems for residual heat removal in all operational states and design basis accidents;
2. measures to prevent unacceptable handling stresses on the fuel assemblies;
3. means for safe storage of non-tight 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.

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

1. means to control water chemistry and activity;
2. means to monitor and control the water level in the storage pool and to detect leakages;
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.

(3) Capacity of the structures for storing of irradiated fuel shall be substantiated in the design considering the capability at any time to completely remove the fuel from the reactor core.

Section XII District Heating System

Art. 95. (1) When the NPP is coupled with heat utilization unit for district heating, the design shall include measures to prevent transport of radioactive materials to the supply system coolant of the district-heating unit in all operational states, design basis and severe accidents.

(2) The measures for prevention of radioactive release to the supply system coolant shall be identified considering the following requirements:

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

Art. 96. (1) Means for isolation of the supply system coolant from the intermediate coolant heat exchanger shall be provided to prevent accidental release of radioactive materials to the intermediate coolant.

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

Section XIII Radiation Protection

Art. 97. (1) To ensure radiation protection, 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) The requirements with regard to the classification of zones and compartments, radiation monitoring, the individual protection means and the access control are established by a different regulation under Art. 26, para 2 of the ASUNE.

Art. 98. To keep the exposure of personnel and public as low as reasonably achievable during 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-living radioactive corrosion products;
2. coolant purification from fission and corrosion 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.

Art. 99. (1) The layout of the plant, its buildings and SSCs shall facilitate the operation, inspections, maintenance, repair and replacement of systems and components and shall limit the personnel exposure to ionising radiation.

(2) The buildings, compartments 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 compartments of high contamination level shall be controlled by means of locking devices with interlocks and indication for actuation and unavailability.

Art. 100. (1) Biological protection shall be designed in a conservative way, taking into account the build-up of radionuclides over the plant lifetime, the potential loss of shielding efficiency due to effects of interactions of neutron and gamma rays with the shielding, due to reactions with other materials, decontamination solution, and the expected temperature conditions in design basis accidents.

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

Art. 101. (1) Ventilation systems shall be installed to:

1. prevent spreading of gaseous radioactive substances in plant compartments;
2. reduce and maintain compartments' airborne concentrations below the established limits and as low as reasonably achievable in all operational states and design basis accidents;

3. cleanup the air in premises containing inert or harmful gases.

(2) In designing a ventilation system, the following factors shall be taken into account:

1. mechanisms of thermal and mechanical mixing;
2. limited effectiveness of dilution in reducing airborne contamination;
3. exhausting of the air from areas of potential contamination at points near the source of contamination;
4. ensuring adequate distance between exhaust air discharge point and the intake point;

5. providing a higher pressure in the less contaminated zones in comparison with the zones of higher contamination level;
6. preventing the spread of fire-released smoke products to neighbouring compartments.

Art. 102. (1) Design shall provide for ventilation and air cleaning systems before discharge of gaseous radioactive substances to the environment.

(2) Filters of 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.

Art. 103. (1) Provisions shall be made in the design for an automated system for radiation monitoring at the workplace and at the NPP site, and a system for radiation monitoring at the radiation protection and the monitored areas. These systems shall ensure the collection and processing of information on the radiation conditions, on the effectiveness of protective barriers, on the radionuclide 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 monitoring;
3. radiation monitoring at the workplace 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 at the radiation protection and monitored areas shall ensure measurement of the content of human induced radionuclides in soil, water, deposits, vegetation, water flora and fauna, and agricultural products.

Chapter Five.
CONSTRUCTION, COMMISSIONING AND OPERATION OF NUCLEAR POWER PLANTS
Section I
Operating Organization

Art. 104. (1) The management body of the operating organization shall establish a document defining the safety policy, which gives the highest priority to safety over all other activities, including clear commitment to continuously improve safety and stimulating personnel for critical attitude to the tasks performed, aimed at achieving of best results.

(2) The safety policy shall be brought to the knowledge of plant personnel and entities that perform work at or provide services for the NPP.

Art. 105. (1) To implement the safety policy, the operating organization shall develop a strategy comprising safety objectives, targets and methods that can easily be followed and monitored.

(2) The adequacy and the implementation status of the safety policy shall be evaluated on a regular basis and the results shall be communicated to the personnel.

Art. 106. (1) The operating organization shall establish an organizational structure justified for safe and reliable operation that comprises and clearly defines the responsibilities, authorities and lines of communication of the personnel who perform activities associated with ensuring and supervising safety.

(2) Changes to the organizational structure, which might be significant for safety, shall be justified in advance, systematically planned and assessed after their implementation.

Art. 107. (1) The management body of the operating organization shall ensure the safe operation in accordance with the Act on the Safe Use of Nuclear Energy and the regulations for its implementation.

(2) During operation of a NPP:

1. decisions on safety matters shall be preceded by appropriate investigation and consultation;

2. personnel shall be provided with the necessary resources and conditions to carry out work in a safe manner;
3. activities associated with ensuring the safety shall be continuously monitored;
4. own and international operational experience and the developments of nuclear science and technology shall be systematically analysed and used for continuous improvement of plant activities.

Art. 108. (1) NPP operations shall be conducted by sufficient in number and qualification personnel who know and understand the design basis, the safety analyses, the plant design and operational documentation for all operational states and accident conditions.

(2) The sufficiency of the personnel and their competence shall be analysed and verified in a systematic and documented way. The operating organization shall develop long term staffing plans for implementation of the activities associated with ensuring and supervising safety.

(3) Any changes in the number of staffing, which may be safety significant, shall be justified in advance, planned and evaluated after implementation.

(4) The operating organization shall maintain sufficient number of qualified personnel to assign, manage and control the activities of the entities who perform work at or provide services to the NPP.

Art. 109. (1) The management body of the operating organization shall apply and maintain an effective quality assurance system based on the following quality principles:

1. managers ensure the planning process, directions, resources and commitment in implementing of the assigned goals in a safe manner;

2. the personnel is familiar with their duties and trained to fulfil them in accordance with the established rules;

3. performing of independent audits of the management processes and of the activities implementation, resulting in achievement of high quality and undertaking of corrective measures when necessary.

(2) The quality assurance system of the operating organization shall cover all activities graded according to their safety significance, including for:

1. defining of organizational structure, responsibilities, authority, interrelations and management processes;

2. improving and maintaining the qualification of the personnel performing tasks associated with ensuring and supervising safety;

3. supplies, construction, installation, operation, maintenance, repair, modification of SSCs important to safety;

4. providing of sufficient resources for implementation of the safety requirements.

(3) The documentation of the quality assurance system shall reflect the intentions of the operating organization management in a clear, concise, unambiguous and consistent way, and shall be developed, agreed, approved and used in accordance with proven procedures.

(4) For each safety related technical activity shall be developed:

1. preliminary validated procedures describing the key measures on quality assurance, the specific conditions to be met before the activity implementation, the steps required for conducting the activity and for responding to any identified;

2. procedures on reporting, evaluation and approval of the results, as well as on decision making with respect to further corrective actions.

Section II Construction

Art. 110. (1) The operating organization shall exercise control over the implementation of design, construction and assembling works, and over the quality of used materials, constructions and components by the aid of its own organizational structure and in compliance with the requirements of the quality assurance system.

(2) The construction materials towards which significant requirements have been specified shall pass verification for compliance and shall hold the necessary documentation and identification mark in conformity to the Act on the Technical Requirements to Products.

Art. 111. For technical assistance during the implementation of the detailed design, the operating organization shall ensure supervision by the NPP designer, which shall continue during the NPP commissioning as well.

Art. 112. Documents confirming compliance with significant requirements pursuant to the Act on the Technical Requirements to Products, those proving the execution of construction and assembling work in accordance with the design and changes performed, as well as the results of the acceptance tests of the materials and components and the individual components tests shall be submitted to the operating organization for analysis and storage.

Section III Commissioning

Art. 113. (1) The operating organization shall develop and carry out a commissioning program to confirm that the implementation of the construction and assembling work is in accordance with the design, and that the SSC characteristics and plant technological processes are in compliance with the design requirements.

(2) The NPP commissioning shall be performed at sequential stages, for which separate programs shall be developed. The implementation of each subsequent stage shall be preceded by results' evaluation of the previous one and demonstration that the objectives and design requirements have been met.

(3) The program for each stage shall comprise the objective, description and schedule for implementation of all important activities during the respective stage. The programs shall specify: the sequence, duration and interface between the activities within the stage; the requirements for process preparation and supply with energy and fluids; the acceptance criteria and evaluation of their implementation; the initial and final state for the stage; the work organization and necessary personnel; the conditions for transition to next stage, and a list of specific procedures for implementation of the activities.

Art. 114. (1) Prior the beginning of plant commissioning shall be developed:

1. the organizational documents for commissioning management;
2. the commissioning limits and conditions;
3. the operating instructions and the repair, maintenance, tests and surveillance procedures for SSCs important to safety;
4. the documents of the quality assurance system, including those for configuration control;
5. emergency operating instructions and on-site emergency plan for the NPP.

(2) Prior the plant commissioning:

1. the means and systems for physical protection, fire protection and NPP access control shall be installed, tested and operable;
2. sufficient in number qualification personnel shall be available.

Art. 115. (1) Commissioning activities shall be carried out in compliance with the commissioning program, testing procedures and operating instructions.

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

Art. 116. (1) Before the initial fuel loading of the reactor core: the systems important to safety required for this stage shall be installed, tested and operable; the reactor coolant system characteristics shall be determined by tests; the shielding effectiveness shall be tested; and radiation monitoring shall be performed at the compartments, the site, the radiation protection area and the monitored area.

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

(3) Transition to next power levels shall be made after successful neutron physics tests (experiments) of the reactor installation and completion of all construction and assembling works at the plant.

(4) Trial-testing operation shall be performed as a commissioning stage for plants equipped with new type reactor.

Art. 117. A power plant, which is in a process of commissioning, shall be physically isolated from other operating or under construction plants at the same site.

Section IV Operation

Art. 118. (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) The inefficiency of a physical barrier or unavailability of a level of defence at specified operational states shall be justified in the NPP design.

Art. 119. (1) To maintain the effectiveness of the physical barriers, the NPP operation shall be conducted in compliance with the limits and conditions for operation.

(2) The limits and conditions for operation shall be identified and justified on the basis of design, safety analyses and commissioning tests, and shall be reviewed periodically and as necessary to consider the operational experience, modifications in SSCs important to safety, new safety analyses and developments in science and technology.

Art. 120. (1) The limits and conditions for operation shall cover all operational states including power operation, reactor subcritical and refuelling, and all transitions between these states, and shall include as a minimum:

1. safety limits;
2. limiting safety systems settings;
3. operational limits and conditions (for normal operation);
4. requirements for tests, inspections, surveillance and in-service inspection of SSCs important to safety;
5. minimum number of operating personnel in the operational states, including qualified and authorized MCR staff;
6. actions to be taken in the case of deviations from the limits and conditions for operation.

(2) In case of non-conformity with the limits and conditions for operation, immediate actions shall be undertaken to bring the plant in compliance with them. Such cases shall be analysed and measures taken to prevent their recurrence in future.

Art. 121. The limits and conditions for operation, collected in an individual document (technical specifications), shall be easily accessible to the MCR personnel, who shall be highly knowledgeable of them and their technical basis. The management personnel of the operating organization shall be aware of their significance for safety.

Art. 122. (1) The operating personnel shall operate the NPP in accordance with written operating instructions and procedures, developed on the basis of the design and technical documentation, the limits and conditions for operation and the results of plant commissioning.

(2) The operating instructions and procedures shall comprise responsibilities of the operating personnel, methods of operative interface and specific directions for implementation of the actions in all operational states.

Art. 123. (1) Personnel actions in case of design basis accidents and beyond design basis accidents shall be specified in instructions, developed on the basis of the final safety analysis report, the limits and conditions for operation and additionally performed investigations and analyses of plant behaviour in accident conditions.

(2) The prescribed in the instructions personnel actions shall lead to recovering the plant state to a condition covered by instructions for normal operation, or to achieving a safe extended shutdown under accident conditions.

Art. 124. (1) Operators actions for diagnosis of plant state, for re-establishment or compensation for lost safety functions, and for prevention and mitigation the consequences of a core damage, shall be specified in severe accidents management guidelines and symptom-based emergency operating instructions (SB EOI).

(2) The set of SB EOI shall include:

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

(3) For development of SB EOI their format, structure and contents shall be specified in a way, that:

1. gives precise, clear and sufficient guidance to the personnel to perform the prescribed actions, including when a transition to other procedures, instructions and guidelines is needed;
2. it is easy to distinguish them from the normal operation procedures and easy to use them;
3. they include directions for monitoring of specified technological parameters (symptoms), for following the automated system response, main operator actions for immediate implementation and the anticipated result of them, and alternative operator actions in case of failure of the main ones;
4. the supplementary background information that aids the operators to follow the procedures is clearly separated.

(4) The data used for development of the documents under Para. 1 shall be specific to the corresponding nuclear plant (unit). The effectiveness of the operator actions shall be analytically validated using verified computer codes and plant specific calculation models. The results of the analysis shall be documented and used as a technical basis of the instructions.

(5) Accident procedures shall be verified and validated by an independent team of experts according to established internal rules (programs) in the form, in which they are used. The practical capability for implementation of operators' actions shall be validated using simulator tools.

(6) The up-to-date status of the procedures for accidents shall be periodically verified.

Art. 125. (1) The NPP operational state and the changes therein shall be controlled and managed by authorized and qualified personnel under the terms and procedures of the Act on the Safe Use of Nuclear Energy.

(2) At least two operators shall be available at the MCR during plant operation, who have a formal authorization (certificate) issued by the Chairman of the Nuclear Regulatory Agency.

(3) The responsibilities and authority of the operating personnel and the persons in charge of safety during operation shall be identified in the NPP organizational documents.

Art. 126. (1) The operating organisation shall develop, periodically review and implement programs for testing, maintenance, repair, inspection and control for maintaining the operability and reliability of SSCs important for safety, according to the design conditions over the plant operating

lifetime. The frequency of testing, maintenance, repair, inspection and control shall be determined based on:

1. their safety importance;
2. their reliability and design requirements;
3. operational experience and previous results;
4. possible influence of fulfilled activities to NPP safety.

(2) To conduct the various types of testing, maintenance, repair, inspection and control, written procedures shall be developed in accordance with the quality assurance system.

(3) The status of base and weld metal of SSCs important to safety shall be inspected periodically by non-destructive testing qualified concerning areas, methods, defect detection and effectiveness according to specifically developed procedures.

(4) Data collected from SSC failures and results from testing, maintenance, repair, inspection and control shall be registered, classified, kept and analysed, and also be used for life time management.

Art. 127. Tests or experiments on SSCs important to safety, which are not included in the technical specifications or in the operating instructions, shall be carried out in compliance with special programs and procedures following a positive statement by the Nuclear Regulatory Agency.

Art. 128. (1) The operating organization shall plan, control and implement permanent and/or temporary changes to SSCs important to safety in a way that does not affect plant's ability to be operated safely. The modifications, including those to instructions and procedures, to evaluation methods in the safety study and to other aspects related to the safe operation, shall be classified according to their safety significance.

(2) For management of changes, procedures shall be used, which depending on the classification of the modification include:

1. responsibilities in the management of modifications and the criteria for completion of each phase;
2. reason and justification for modifications;
3. feasibility study;
4. design requirements;
5. comprehensive safety assessment of the modifications leading to changes in the plant configuration or in the limits and conditions for operation;
6. methods for fabrication, installation and testing;
7. commissioning the modification.

(3) The comprehensive safety assessment shall consider all safety aspects, applicable legal requirements and standards and shall be performed by personnel independent of those responsible for the design and implementation of the modification.

(4) Before commissioning of modifications, personnel shall have been trained and all relevant operational documentation shall have been revised or updated.

(5) The management of temporary modifications shall include: their clear identification and marking; informing the personnel about them; assessment of their impact upon the plant operation, limiting their number and period of application, and periodic review of the need for these modifications.

Art. 129. (1) The operating organization shall establish and conduct a program to collect, analyse, document and disseminate their own and the others' operational experience, aiming at identifying the good operational practice and events, precursors and tendencies towards degrading the safety performance or reduction the safety margins, and undertaking of corrective actions.

(2) Safety significant operational events shall be analysed based on procedures, which specify the evaluation methods of the behaviour of SSCs and personnel in view of:

1. establishing the chronological event sequence;
2. identifying the deviations and erroneous actions;
3. direct and root cause analysis;
4. assessment of event safety significance, including possible consequences;

5. identifying corrective actions.

Section V Radiation Protection of Personnel and Public during operation

Art. 130. (1) The operating organization shall develop and agree with the competent state authorities programs for radiation protection of NPP personnel and for radiation monitoring of the environment, and shall periodically review and update them on the basis of the operational experience. The programs shall involve requirements to:

1. classification of areas and access control of personnel and materials;
2. co-operation in establishing operating and maintenance procedures for activities when radiological hazard is anticipated;
3. instrumentation and equipment for monitoring;
4. equipment for collective and individual personnel protection;
5. on-site radiological monitoring and surveys;
6. decontamination structures and systems;
7. environment radiological surveillance and monitoring;
8. monitoring of radioactive liquid and gaseous releases;
9. control to limit the extent of radioactive release outside the NPP site.

(2) The operating organization shall ensure sufficient independence and resources to the organizational structure that implements radiation protection function and control of working conditions.

Art. 131. (1) All site personnel shall be aware of the radiological hazard of the activities and shall have individual responsibility for putting into practice the exposure control measures.

(2) The operating organization shall provide for preliminary and periodic medical examination of the NPP personnel to ascertain their health and psycho-physiological fitness to occupy relevant position.

Art. 132. (1) The generation of radioactive waste shall be kept to the minimum practicable in terms of both activity and volume, by appropriate operating practices.

(2) Treatment and interim storage of radioactive waste shall take account of the requirements for their final disposal.

Art. 133. The operating organization shall perform periodic analysis and assessment of the radioactive discharges to the environment to demonstrate that the radiological impacts and doses to the public do not exceed the annual limit specified under Art. 10, para 1, and are kept as low as reasonably achievable.

Section VI Personnel Training and Qualification

Art. 134. (1) The operating organization shall ensure that the activities related to ensuring and control of safety during NPP operation are performed by personnel possessing the necessary qualification and experience.

(2) The personnel training and qualification shall ensure sufficient knowledge about the characteristics and behaviour of SSCs important to safety, and the NPP as a whole, in all operational states and accident conditions.

Art. 135. (1) NPP operating personnel shall be prepared and trained to occupy gradually ascending operating positions after doubling the corresponding work places over a justified period of time.

(2) The training programs of the operating personnel shall include design basis, final safety analysis report, limits and conditions for operation, on-site emergency plan, operational event analysis, and documentation on modifications of SSCs important to safety.

Art. 136. (1) Control room operators shall pass training at a full-scope simulator at least once a year and the shift crews – periodical emergency training and exercise.

(2) Maintenance personnel shall be trained using scaled models or real components for improvement of their professional skills and reducing the duration of operations before implementation of work under radiation hazard.

(3) Prior carrying out of key operating actions and tests of SSCs important to safety, the personnel involved shall be duly briefed or instructed.

Section VII Management of Safety Related Documentation

Art. 137. (1) The operating organization shall develop and apply procedures for management of documents and records related to safety, such as:

1. design specifications;
2. safety analyses and fire risk assessment;
3. data on equipment and materials;
4. as-built installation drawings;
5. SSC manufacturers' documentation;
6. commissioning data;
7. NPP operational data;
8. reports on events and incidents;
9. records of amounts and movements of fissile, radioactive and other special materials and substances;
10. documents from maintenance, testing, surveillance and inspections;
11. data on modifications;
12. quality assurance documents;
13. data on personnel qualification, positions, medical examinations and training;
14. plant chemistry records;
15. data on occupational exposure;
16. data on radiation surveys at the premises and on the site;
17. effluents discharge records;
18. environmental monitoring data;
19. data on storage and transport of radioactive waste;
20. periodic safety reviews.

(2) The documents under Para. 1, 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.

(3) The on-site emergency plan and the procedures to be used under accident conditions, as well as other important documents may be stored outside the NPP site, as necessary.

Art. 138. The document management system shall ensure that only the latest versions of all documents and programs are being used.

Additional provisions

§ 1. Within the meaning of this regulation:

1. "Accident conditions" are deviations from normal operation more severe than anticipated operational occurrences, including design basis and beyond design basis accidents.

2. "Active component" is a component whose functioning depends on an external input such as actuation, mechanical movement or supply of power.

3. "Capable fault" is a tectonic fault, along which a relative shifting of the adjacent earth crust blocks has been realized at half a meter or more over the last one million years (Quaternary period).

4. "Fail safe" is a failure of a system or component as a result of which the nuclear power plant passes to a safety state with no necessity for any actions to be initiated by control safety systems.

5. "Main control room" is part of a nuclear power unit situated in specially designed premises and intended for centralized control of the technological processes performed by the operating personnel and the instrumentation and controls.

6. "Validation" is the process of determining whether the product (such as computer programs, analytical methods, NPP models, procedures and instructions) is suitable for satisfactory implementation of the intended function.

7. "Verification" is the process of determining whether the product's quality or characteristics (such as computer programs, analytical methods, NPP models, procedures and instructions) are exactly the same as declared, as foreseen, or as required.

8. "Probabilistic safety assessment" is a comprehensive, structured approach to identify failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk.

9. "External event" is an event unconnected with the NPP operation, which could have an effect on the NPP safety.

10. "Internal event" is an event connected with the NPP operation, which could have an effect on the NPP safety.

11. "Inherent safety feature of a reactor installation" is the reactor installation property to ensure safety on the basis of natural feedbacks, processes and characteristics.

12. "Geodynamic zone" is a linear or ring-shaped site of the earth's crust, within which boundaries the identified velocity gradient of the quaternary movements is equal to or larger than 10^{-9} per year.

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

14. "Operational states" are the states defined under normal operation and anticipated operational occurrences.

15. "Operational limits" are values of the parameters and characteristics of the status of systems (components) and of the NPP as a whole, specified in the design for normal operation.

16. "Operational conditions" are 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.

17. "Operation" is all activities performed to achieve the purpose for which the NPP was constructed, including power operation, start up, shutdown, testing, maintenance, repairs, refuelling, in-service inspection and other related activities.

18. "Operating organization" is the organization that is licensee or permit holder under the Act on the Safe Use of Nuclear Energy.

19. "Protection safety system" is a system intended to prevent or limit a damage of the nuclear fuel, of the fuel claddings, and the components that contain radioactive substances.

20. "Qualification" is the process of establishing evidence that the structure, system or component will operate on demand, under specified service conditions to meet the system performance requirements.

21. "Component" is devices, piping, cables, and other articles that ensure the performance of preset functions either on their own or within systems, and conceived as structural units in the reliability and safety analyses.

22. "Conservative approach" is an approach to the design and construction, in the analysis and calculations of which are accepted values and limits for the parameters and characteristics that definitely lead to more unfavourable results.

23. "Reactor coolant system" is the system intended for circulation of the coolant through the reactor core within the operational modes and conditions established in the design.

24. "Ultimate heat sink" is a medium to which the residual heat can always be transferred, even if all other means of removing the heat have been lost or are insufficient.

25. "Safety culture" is personnel qualification and psychological attitude, which establishes that ensuring the NPP safety is an overriding priority and inherent necessity, leading to the conscious of responsibility and self-control in performing all activities that affect safety.

26. "Localization safety system" is a system intended to prevent or limit the dispersion of accidentally released radioactive substances and ionising radiation outside the designed boundaries to the environment.

27. "Beyond design basis accident" is an accident with consequences more severe than the design basis accident and not involving significant core degradation, as well as a severe accident.

28. "Undetected (latent) failure" is a failure of a system (component) which is not developed at the moment of its occurrence during normal operation and is not identifiable with the inspection means according to the programs for maintenance and inspection.

29. "Normal operation" is the operation within specified operational limits and conditions.

30. "Supporting safety system" is a system intended to supply the safety systems with energy and fluids and to maintain appropriate conditions for their operation.

31. "Common cause failure" is a failure of two or more systems or components due to a single specific event or cause.

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

33. "Passive component (passive system)" is a component (system) whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.

34. "Periodic safety review" is 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 plant.

35. "Personnel" means all people working permanently or temporarily on the NPP site.

36. "Postulated initiating event" is a single failure in system (component), external event or human error identified during design as capable of leading to anticipated operational occurrences or accident conditions.

37. "Safety limits" are values of operational parameters, specified in the design, the deviations from which may result in an accident.

38. "Limits and conditions for operation" is 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 a NPP.

39. "Design basis accident" is an accident against which a NPP is designed according to established design limits, including damage to the fuel and the release of radioactive material to the environment.

40. "Design limits" are values of parameters and of characteristic status of SSCs important to safety and of NPP as a whole, specified in the design for all operational states and accident conditions.

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

42. "Region of siting of a NPP" is a territory, incorporating the NPP site, for which the NPP location conditions have to be identified and where the occurrence of phenomena, processes and factors of natural and human induced origin that may affect the plant safety is possible.

43. "Supplementary control room" is part of a nuclear power unit situated in a specially designed premise and intended to reliably bring and maintain in a long term the reactor in cold subcritical state, to actuate the safety systems and to receive information on the reactor state, should there be a loss of functions of the MCR.

44. "Redundancy" is provision of alternative (identical or diverse) SSCs, so that any one can perform the required function regardless of the state of operation or failure of any other.

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

46. “Structures, systems and components for normal operation” are the SSCs intended for normal operation.

47. “Structures, systems and components important to safety” are the safety systems and the SSCs for normal operation, the failures in which either result in deviation of the plant normal operation or prevent the response to deviations of the normal operation and can lead to design basis or beyond design basis accidents.

48. “Structures, systems and components” (SSCs) are all of the elements (items) of NPP, except human factors. Structures are the 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.

49. “NPP states, plant states” are the operational states and accident conditions.

50. “Severe accident” is an accident involving significant core degradation.

51. “Accidents management” is the undertaking of a set of actions during the evolution of a beyond design basis accident to prevent the escalation of the event into a severe accident; to mitigate the consequences of a severe accident; to achieve a long term safe stable state.

52. “Control safety systems” are the systems intended to actuate the safety systems and to control their operation in the process of performing the designated function.

53. “Safety function” is the specific objective to be achieved in order to ensure the safety.

54. “Containment structure” is an assembly of structural and other components packaging the reactor installation space that forms the physical barrier specified in the design and prevents the dispersion of radioactive substances to the environment. The space encircled within the containment structure boundaries constitutes the containment.

§ 2. (1) The site selection, design, construction and commissioning of NPPs, as well as their reconstructions, major repairs and modernizations shall be accomplished observing the technical safety requirements in conformity with the regulation, under the terms of and following the procedure of the Act on Territorial Structure (ATS) and the Act on the Safe Use of Nuclear Energy (ASUNE).

(2) The permit referred to in Art. 33, Para. 1, item. 1 of the ASUNE as to site location (site selection) is a basis (ground) of issuing a permit by the Minister of regional development and public works under Art. 124, paragraphs 2 and 4 of ATS – for the purpose of elaboration of a detailed lay-out plan.

(3) The orders under Art. 33, paragraph 4 of ASUNE concerning the approval of the selected site and basic design are a ground for approval by the Minister of regional development and public works of the detailed lay-out plan and of the technical investment project covered by the ATS.

(4) The construction permit issued under Art. 33, Para. 1, item. 1 of the ASUNE is a ground for issuing a permit under the ATS by the Minister of regional development and public works.

(5) The permit allowing usage under ATS is a ground for issuing a permit for commissioning under Art. 33, Para. 1, item 4 of the ASUNE.

Transitional and Final Provisions

§ 3. (1) The regulation provisions are respectively applied to the implementation of modifications leading to a change in SSCs important to safety of the existing NPPs that had been commissioned prior to the regulation enforcement.

(2) Cases outside Para. 1 regarding the existing NPPs that had been commissioned prior to the regulation enforcement are within the scope of provisions listed in chapter five, sections I and IV – VII of the current regulation, and the following requirements shall be met with respect to these NPPs:

1. the frequency for significant core damage as a result of severe accident determined on the basis of a probabilistic safety analysis shall be sufficiently lower than 10^{-4} events per NPP per year;

2. the annual effective dose for an individual of the public caused by liquid and gaseous effluents to the environment in all operational states shall be lower than 0.25 mSv;

3. the frequency of large radioactive release into the environment that require implementation of urgent protective measures for the public shall be lower than 10^{-5} events per NPP per year;

4. the annual individual effective dose resulting from internal and external exposure of the public at the boundary of the radiation protection area and beyond it shall not exceed 50 mSv for the first year following a design basis accident;

5. the annual individual effective dose resulting from internal and external exposure of the public at the boundary of the area of urgent protective measures shall not exceed 5 mSv for the first year following a beyond design basis accident and less than 1 mSv annually for the subsequent years.

(3) Upon extension of the term of validity of a license for operation of a specific unit of a NPP that have been commissioned prior to the regulation enforcement, the assessment of nuclear safety and radiation protection under Art. 20, Para. 2 of ASUNE shall be carried out complying with the requirements of Art. 22 of this regulation.

(4) Entities who operate NPPs that have been commissioned prior to the regulation enforcement are obliged to bring their activity in compliance with the requirements of Para. 2 above within two years following the enforcement of this regulation.

§ 4. When some of the activities covered by this regulation are assigned by a contract to another entity, the assignor bears the responsibility for ensuring the compliance of the work done with the regulation's requirements.

§ 5. (1) Any violations under this regulation shall be ascertained by statements drawn following the order of the ASUNE and the Law on administrative violations and penalties.

(2) Any failure to comply with the current regulation requirements is penalized as an administrative violation within the meaning of Art. 141, Para. 1 of ASUNE, unless it is liable to a stricter punishment.

§ 6. Guidance and interpretations as to the regulation application are given by the Chairman of the Nuclear Regulation Agency.

§ 7. This Regulation is adopted on the grounds of Art. 26, paragraph 2 of the ASUNE.

Annex to Art.12, paragraph 2

Typical list of postulated initiating events and the categories of plant states to be considered in the safety analysis of a NPP with pressurized water reactors

1. Category 1. Steady and transient states during normal operation:
 - 1.1. Start up
 - 1.2. Power operation
 - 1.3. Hot standby
 - 1.4. Hot shutdown
 - 1.5. Cold shutdown
 - 1.6. Refuelling
 - 1.7. Operation with an inactive loop
 - 1.8. Temperature increase and decrease at a maximum admissible rate
 - 1.9. Step load increase and decrease (by 10 %)
 - 1.10. Load increase and decrease (at a rate of 5 % load/minute) within the range between 15 and 100 % full power
 - 1.11. Switch-over to house load operation from 100 % power with steam dump
 - 1.12. Limiting conditions allowed by the OLCs.
2. Category 2. Anticipated operational occurrences with frequency above 10^{-2} events per year:
 - 2.1. Inadvertent withdrawal of a control rod group with reactor subcritical
 - 2.2. Inadvertent withdrawal of a control rod group with reactor at power
 - 2.3. Static misalignment of control rod or drop of a control rod group
 - 2.4. Inadvertent boric acid dilution, partial loss of core coolant flow
 - 2.5. Inadvertent closure of main steam isolation valve
 - 2.6. Total loss of load and/or turbine trip
 - 2.7. Loss of main feed water flow to steam generators
 - 2.8. Malfunction of the main feed water system of steam generators
 - 2.9. Total loss of off-site power (up to 2 hours)
 - 2.10. Excess increase in turbine load
 - 2.11. Temporary depressurisation of the reactor coolant system (inadvertent actuation of the pressurizer spray)
 - 2.12. Spurious opening of steam-generator safety valve or other secondary side depressurisation caused by a single failure
 - 2.13. Spurious start up of safety injection system
 - 2.14. Malfunction of chemical and volume control system
 - 2.15. Very small loss of reactor coolant (e.g. small instrumentation line break).
3. Category 3. Accidents of low frequency of occurrence in the range between 10^{-2} and 10^{-4} events per year:
 - 3.1. Loss of reactor coolant (small pipe break)
 - 3.2. Small secondary pipe break
 - 3.3. Forced reduction in reactor coolant flow
 - 3.4. Mispositioning of a fuel assembly in the core with consequent operation
 - 3.5. Withdrawal of a single control rod in power operation
 - 3.6. Inadvertent opening and sticking open of a pressurizer safety valve
 - 3.7. Rupture of volume control tank
 - 3.8. Rupture of gaseous radioactive waste hold-up tank
 - 3.9. Failure of liquid radioactive waste effluent tank
 - 3.10. One steam-generator tube break without previous iodine spiking
 - 3.11. Total loss of off-site power (up to 72 hours).
4. Category 4. Design basis accident of very low frequency of occurrence, in the range between 10^{-4} and 10^{-6} events per year.
 - 4.1. Main steam line break

- 4.2. Main feed water line break
- 4.3. Locked rotor of a reactor coolant pump
- 4.4. Ejection of any single control rod
- 4.5. Loss of reactor coolant up to and including double-ended guillotine break of the largest pipe
- 4.6. Fuel handling accidents
- 4.7. One steam generator tube break with previous iodine spiking.

Typical list of postulated initiating events and the categories of plant states to be considered in the safety analysis of a NPP with boiling reactors

- 1. Category 1. Steady and transient states during normal operation:
 - 1.1. Start up
 - 1.2. Power operation
 - 1.3. Hot standby
 - 1.4. Hot shutdown
 - 1.5. Cold shutdown
 - 1.6. Refuelling
 - 1.7. Temperature increase and decrease at a maximum admissible rate
 - 1.8. Step load increase and decrease (by 10 % load).
 - 1.9. Load increase and decrease (at a rate 5 % load/minute) within the range between 15 and 100 % full power
 - 1.10. Switch-over to house load operation from 100 % power with steam dump
 - 1.11. Limiting conditions allowed by the OLCs
- 2. Category 2. Anticipated operational occurrences with frequency above 10^{-2} events per year:
 - 2.1. Inadvertent withdrawal of a control rod with reactor subcritical
 - 2.2. Inadvertent withdrawal of a control rod with reactor at power
 - 2.3. Static misalignment of control rod assembly
 - 2.4. Total loss of load and/or turbine trip
 - 2.5. Loss of main feed water flow
 - 2.6. Total loss of off-site power (up to 2 hours)
 - 2.7. Excess increase in turbine load (at full power)
 - 2.8. Spurious opening and sticking open of the turbine control valve or bypass valve
 - 2.9. Closure of one of the turbine stop valves
 - 2.10. Spurious start up of safety injection system
 - 2.11. Reactor isolation, loss of the main heat sink
 - 2.12. Loss of all internal pumps (if applicable)
 - 2.13. Inadvertent opening of a safety/relief valve
 - 2.14. Stuck open of a safety/relief valve
- 3. Category 3. Accidents of low frequency of occurrence in the range between 10^{-2} and 10^{-4} events per year:
 - 3.1. Loss of reactor coolant (small pipe break inside containment).
 - 3.2. Pipe breaks outside containment (main steam line, feed water line, connection to the pressure suppression pool)
 - 3.3. Mispositioning of a fuel assembly in the core with subsequent operation
 - 3.4. Rupture of gaseous waste hold-up tank
 - 3.5. Total loss of off-site power (up to 72 hours).
- 4. Category 4. Design basis accident of very low frequency of occurrence in the range between 10^{-4} and 10^{-6} events per year.
 - 4.1. Loss of reactor coolant (pipe break inside containment)
 - 4.2. Large Break
 - 4.3. Intermediate break

- 4.4. Break in the bottom of the RPV (instrumentation line)
- 4.5. Break in the component cooling system (internal flooding of safety systems)
- 4.6. Ejection of any single control rod
- 4.7. Fuel handling accident
- 4.8. Control rod accident.