RULES

ON RADIATION AND NUCLEAR SAFETY FACTORS (JV5)

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Pursuant to the Protection against Ionising Radiation and Nuclear Safety Act (Official Gazette RS, No. 102/04 – official consolidated text and 70/08-ZVO-1B) Article 7a paragraph four, Article 63 paragraph six, Article 71 paragraph five, Article 78 paragraph five and Article 80 paragraph five, the Minister of the Environment and Spatial Planning issues the following

RULES

on Radiation and Nuclear Safety Factors

1. GENERAL PROVISIONS

(1) These Rules lay down:

Article 1 (contents)

1. the design bases for radiation and nuclear facilities;

2. the content of the applications and documents accompanying applications for consents and permits for radiation and nuclear facilities, and less important radiation facilities;

3. the content of safety-analysis reports and other documents required to demonstrate and guarantee the safety of radiation and nuclear facilities;

4. detailed requirements concerning the organisational set-up of a radiation or nuclear facility, and concerning the content and form of the management system and its implementation in radiation and nuclear facilities;

5. detailed requirements concerning nature, scope, method of protection and preservation of documents of operator of a radiation or nuclear facility.

(2) With these Rules in the Slovenian legal system transfer:

1. the Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations (OJ L No. 172 of 7 June 2009, p. 18 ), last amended by Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations (OJ L No. 219 of 25 July 2014, p. 42 );

2. Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste (OJ L No. 199 of 2 August 2011, p. 48 ) and

3. Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom,

97/43/Euratom and 2003/122/Euratom (OJ L No. 13 of 17 January 2014, p. 1) last amended by Corrigendum to Council Directive 2013/59/Euratom of 5 December 2013

laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom,

90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom (OJ L No. 72

of 17 March 2016, p. 69).

Article 2 (definition of terms)

The terms appearing in these Rules shall have the following meanings:

1. The deterministic safety analysis shall mean a safety analysis carried out by modeling, identifying and calculating safety-relevant parameters and processes in a radiation or nuclear facility, which arise after the postulated initial events. The main purpose is to verify that the authorized values of basic safety parameters of the object are not exceeded;

2. An event shall be a human error or action caused by incorrect written procedure or instruction, equipment failure, operational error, natural event or design deficiency that may challenge radiation or nuclear safety;

3. Electromagnetic compatibility shall mean the capacity of electric and electronic systems, equipment and devices to function in an electromagnetic environment in which they shall be operable, without any deterioration of their functionality due to electromagnetic disturbances or mutual interference;

4. Single failure shall mean a failure that results in the loss of the capability of a component to perform its intended safety function(s), and any consequent failure(s) resulting from this;

5. Safe enclosure mode shall mean the period between the operation phase and the commencement of its decommissioning of a radiation or nuclear facility and between operation and post closure phase or re-operation of disposal facility. Its duration may vary and its primary purpose is to decrease radiation loads during the subsequent decommissioning by allowing for the decay of short-lived isotopes. The purpose of safe enclosure mode of disposal facility is to optimise the operation of disposal facility;

6. Extreme weather conditions shall mean extreme weather conditions in the area in question, determined based on an analysis of historical weather data for the area;

7. Classification of structures, systems and components (hereinafter refered as SSCs) shall mean their classification in safety categories according to their relevance to the risks established in a probabilistic safety assessment;

8. Control room shall mean a room in nuclear power plant or research reactor, in which information is gathered on the facility operation and from which all processes important for safety can be controlled;

9. Competencies shall consist of educational level, training, skills and experiences to execute tasks;

10. Qualification shall mean a process that proves that a specific SSC can operate on demand in predicted environmental conditions until the end of its qualified lifetime and in accordance with required criteria;

11. Site shall mean a geographical area where a radiation or nuclear facility is located and where its intended practices are carried out;

12. Facility configuration management shall mean the management of the design documents of a radiation or nuclear facility to ensure conformance of the actual facility condition with the design specifications, the availability of all the required data and information in the documents, and the conformance of documents with the actual condition of the facility;

13. Abnormal operation shall mean operation outside normal ranges of expected operating parameters, which happens at least once during the operating lifetime of the facility, but which, in view of appropriate design provisions, does not cause any significant damage to SSCs important to safety or lead to an accident;

14. Accident shall mean deviation from normal operation, which is less frequent and has more severe consequences than an abnormal operation. In the case of an accident severe damage of a nuclear or radiological facility or reduced efficency of safety barriers may occur;

15. Beyond-design-basis accident shall mean an accident which may occur, but it is not considered in the design basis of a nuclear facility, because of its very low probability of occurrence. It includes design extension conditions and severe accidents;

16. Incident shall mean an undesirable condition with consequences, which are not negligible from radiation protection and nuclear safety. Inncident may cause by inappropriate human action or a system or component failure. Incident require identification of the fault and its remedying or corrective action;

17. Site region of a radiation or nuclear facility shall mean the site and the area surrounding it that is relevant to the facility safety evaluation, due to the characteristics of natural and human-induced events;

18. Defence-in-depth shall mean the design principle according to which a protection purpose (e.g. the prevention of radioactive material releases) should be achieved by combining, in the design and operation phases, several safety measures to ensure the achievement of the purpose even in the event of a failure of one of the safety measures;

19. Operational conditions and limits shall mean the set of rules, set out in the safety- analysis report, that lay down limits for parameters, capabilities and performance of equipment and of personnel actions to ensure safe operation of a radiation or nuclear facility;

20. Operation shall mean all the activities carried out to realise the intended purpose of a radiation or nuclear facility, including maintenance, refuelling (in the case of a nuclear power plant or a research reactor), in-service inspection, keeping, storage, disposal of mining or hydrometallurgical tailings, radioactive waste or spent fuel and other associated activities;

21. Irradiation rig shall mean a device containing a source of ionising radiation intended for use in irradiation;

22. Common cause failure shall mean the failure of two or more SSC as a result of the same cause or event (e.g. design deficiency, deficiency in the implementation or maintenance of the SSC, external events or their combinations);

23. Environmental conditions shall mean the conditions in which a selected SSC, equipment or instrument shall be operable and include, inter alia, temperature, pressure, radiation, relative humidity, chemical environment, exposure to flood, earthquake, etc.;

24. Operability shall mean the condition of a SSC in which its capability to operate or perform its tasks in conformance with operational conditions and limits is maintained. Furthermore, operability shall also be maintained for those SSCs that ensure, through ancillary functions (e.g. power supply, cooling, lubrication, etc.), the operability of the SSC in question in conformance with operational conditions and limits;

25. Operator shall mean a qualified person with a licence to control or provide instructions for control of a radiation or nuclear facility:

26. Licensed personnel shall mean workers that have successfully completed professional training and tests of proficiency in accordance with the regulation on conditions to be fulfilled by workers performing safety-significant tasks at nuclear or radiation facilities;

27. Counterfeit items shall mean items that do not comply with applicable standards. These items include:

- inadequate items made by unknown or unapproved producers, which do not comply with applicable standards, specifications or technical requirements referred in purchase documents;

- counterfeit items that are intentionally designed or modified so that it would look like real products;

- fraudulent items whose material, properties or characteristics deliberately shown differently than they really are;

- suspicious itmes for which after visual examination, test, or other preliminary information, there is a suspicion that they do not comply with the applicable standards, specifications or technical requirements referred in purchase documents;

28. Procedure shall mean a method laid down for the execution of an activity or process.

Operational procedures are detailed written procedures for the operation of a radiation or nuclear facility;

29. Emergency operating procedures shall mean procedures provided for management of design basis events and include procedures to restore the facility state to a safe condition. For nuclear facilities these procedures also include management of design extension conditions category A, which are not included in the design basis of the facility;

30. Core damage shall mean core uncovery and overheating up to a level where increased oxidation and severe damage of a major part of the core can be expected;

31. Fire cell shall mean an area, separate from other cells, containing safety-related equipment. A fire cell need not necessarily be fully enclosed by fire-protection barriers; accordingly, the propagation of fire between fire cells may be avoided by limiting the volume of flammable materials, controlling distances between objects, provision of fire- extinguishing systems and passive fire-protection (e.g. fire shields and envelopes). A fire cell shall be constructed so as to limit the propagation of fire originating in the cell to the outside for a given time period, which may differ from that specified for a fire compartment;

32. Fire compartment shall mean a building or part of a building fully enclosed with fire barriers that can withstand even the total anticipated fire load within the compartment or outside the compartment. Fire barriers include doors, walls, floors and ceilings. A fire compartment shall be constructed so as to limit the propagation of fire originating in the compartment to the neighboring compartments for a given time period;

33. Treatment shall mean any of the following procedures for radioactive waste or spent- fuel conditioning prior to storage, transport or disposal:

- preconditioning to prepare radioactive waste or spent fuel for conditioning;

- conditioning to alter radioactive waste or spent fuel characteristics for technical, economic or safety reasons, and

- preparation to render packaged and unpackaged radioactive waste or spent fuel into a form meeting requirements for transport, storage or disposal;

34. Postulated initiating event shall mean an event that is recognized as a part of design bases, and that may trigger an anticipated operational occurrence or an accident;

35. Transient shall mean a set of developments through which a system transfers from one stable state to another;

36. Testing shall mean an activity planned in advance to check the operability of a SSC, and shall be carried out following maintenance operations on or modifications of a SSC. Periodic testing laid down in operating conditions and limits is surveillance testing;

37. Tested components shall mean components properly tested and qualified or equal to other components tested under the same conditions;

38. Anticipated operational occurrence shall mean an event that is expected to occur at least once within the operating lifetime of a radiation or nuclear facility, but does not cause significant damage to SSCs important to safety and does not lead to an accident;

39. Process shall mean a set of interlinked activities or activities mutually affecting each other, with the intention of reaching a given goal;

40. Design limit shall mean an extreme (limit) value of a parameter, specified in the design phase, which may not be exceeded during the operation of a facility, nor, in the case of a disposal facility, after its closure;

41. Design bases accident shall mean accident caused by design bases event. A radiation or nuclear facility shall be designed to assure that releases of radioactive materials at the design basis accident are below the authorised limits;

42. Design bases of a SSC consist of information specifying the purpose of the SSC and specific values or ranges of values to be achieved by the SSC. These values mean limits established based on generally accepted engineering practice concerning fulfilment of functional requirements, or requirements based on analysis (applying

either calculation or experimental methods) of the consequences of a postulated initiating event in which a SSC must perform its function;

43. Design basis event shall mean an event that leads to design basis accident for which the facility is designed in accordance with established design criteria and conservative methodology;

44. Research reactor shall mean a nuclear reactor that is applied primarily to generate and apply neutron and ionising radiation for the purposes of research, production of radionuclides and similar. A research reactor consists of a reactor core, experimental rig and all other plant associated with the operation of the reactor and its experimental rig;

45. Design extension conditions shall mean an accident caused by design extension conditions event. It comprises of design extension conditions category A and category B;

46. Design extension conditions event shall mean an event or combination of events with extermely low probability and more severe consequences than design bases events. It can also include multiple failures of SSC in contrast to single failure postulated in the design basis of the nuclear facility. There are two categories of design extension conditions events:

- Design extension conditions categoriy A, for which prevention of severe fuel damage in the reactor or spent fuel storage can be achieved;

- Design extension conditions category B with postulated severe fuel damage, exceeding the design basis fuel damage;

47. Reference documents shall mean the documents referred to in the contents of the safety-analysis report or that have been applied in any other way as a basis in the process of granting the consent to construction or granting an operating licence, a permit to terminate operation, or a permit to decommission a radiation or nuclear facility, and in the case of a disposal facility also for closure;

48. Mining works shall mean works for the purposes of exploration and exploitation of mineral resources or suspending exploitation, and are, according to the given nature and purpose of works, classified as exploratory mining operations, mining and mining rehabilitation work;

49. Normal evolution of a disposal facility shall mean the anticipated long-term degradation of the facility condition following its closure due to natural processes or human interventions, and is compiled by extrapolating the present conditions into the future;

50. Alternative evolution of a disposal facility shall refer to accounting for undesired events and development in conditions following closure, due to natural causes, or human-, animal- or plant-induced, leading to accelerated long-term degradation of the disposal facility, migration of radioactive materials and increased radiation levels (e.g. intentional human ingress, water or mineral bores, consequences of greenhouse effects, activation of geological faults, global icing, failure of facility sealing, migration with generated gases);

51. Simulator shall mean a device which responds to the operator’s action in the same way as the real system. A nuclear power plant simulator usually includes the control room in the scale equal to the scale of the real control room, while the simulator' software must include normal operation, abnormal operation and accidents;

52. The spent fuel storage is a place or facility where spent nuclear fuel is temporarily stored;

53. Severe accident management guidelines shall mean written procedures including guidelines for operators on managing the consequences of such accidents;

54. Modification of a radiation or nuclear facility shall mean any proposed modification in relation to the facility or to the method of its control or operation, including maintenance works, inspections, testing, or implementation of any technical, organisational or other change in relation to such works;

55. SSC shall be an abbreviation for a set of structures, systems and components.

Structures mean passive elements such as buildings and shields. A system means a set of components combined so as to perform an (active) function. The term SSC

includes instrumentation and control software. In the case of radioactive waste storage or disposal facility, the term SSC includes the radioactive waste package;

56. SSC important to safety shall ensure that the anticipated operational occurrences and design basis events do not lead to the exceedance of the limits specified in the design bases, and whose failure or malfunction can lead to undue radiation exposure or contamination of humans or the environment;

57. Facility state shall mean the operational states of a radiation or nuclear facility or accident conditions. The operational states are divided into normal operation of a radiation or nuclear facility, when there are no equipment failures or violation of operating procedures and abnormal operation when there is a failure or violation of procedures, but the nuclear and radiation safety are not threatened. During the accident conditions the nucear and radiation safety are threatened;

58. Safe shutdown state of a reactor shall mean a state in which the reactor is subcritical and residual-heat removal is provided;

59. Graded approach shall mean the processes for ensureing that the level of analysis, documentation and actions used to comply with a requirement in this part are commensurate with:

- the relative importance to safety, safeguards and security;

- the magnitude of any hazard involved;

- the life cycle stage of a facility;

- the programmatic mission of a facility;

- particular characteristic of a facility;

- the relative importance of radiological and nonradiological hazards and any other relevant factor;

60. Technical support centre shall mean rooms and corresponding support equipment on site or near to the facility site available to personnel to provide technical support to operators and professional personnel and to allow the management of emergencies within the facility site;

61. Severe accident shall mean an accident in a nuclear power plant, research reactor or spent fuel storage, the consequences of which exceed a design basis accident cathegory A and lead to core meltdown or spent fuel and dangers to the environment or may cause irradiation or contamination of people. Severe accidents can occur due to multiple failure, such as loss of all trains of safety systems, or due to the extremely unlikely event, for which the plant is not designed;

62. The pressure boundary shall mean a physical barrier separating two technological systems operating under different operating pressures. Usually it consists of pressure vessels, pipes, valves, piping and instrumentation connections;

63. Training shall mean the systematic acquiring of required knowledge and skills to supplement the appropriate education level for certain work posts;

64. Validation shall mean demonstration, through material evidence, of compliance with the requirements for the intended use;

65. Safe state shall mean any state of the facility (eg. shutdown state, operation or safe enclosure mode), in which all safety functions are provided;

66. Nuclear-criticality safety shall mean a state in which any risk of self-sustaining nuclear chain reaction is prevented;

67. Safety function shall mean a purpose that must be achieved or a action that must be performed in order to ensure radiation or nuclear safety. Safety functions for a nuclear reactor are the following:

- control of reactivity of the nuclear fuel;

- heat removal from the reactor core and the spent fuel storage;

- confinement of radioactive materials and prevention of their uncontrolled release into the environment;

68. Safety classification shall mean classification of SSCs according to their required safety functions in order to ensure radiation or nuclear safety and the classification of safety- related SSCs into safety classes according to their importance for nuclear safety;

69. Safety margin shall mean the difference between the limit value of a parameter that leads to a SSC failure, and the value of the parameter authorised by the Slovenian Nuclear Safety Administration (hereinafter: Administration)as part of the procedure for issuing consents and permits to radiation and nuclear facilities;

70. Safety limits shall mean the limits of parameters determined on the most unfavorable values of parameters, including the safety margin, within which the safe state of the facility is still ensured;

71. Safety system shall mean a system necessary to perform a safety function, including support systems;

72. Safety-analysis report shall mean a document or set of documents providing key information on a radiation or nuclear facility, its operating conditions and limits, its impacts on the environment, the description of the design, analysis of possible accidents and measures necessary to prevent or mitigate threats to the environment, population and facility personnel;

73. Protection system shall mean a system in the radiation or nuclear facility that monitors the status of safety-related parameters and automatically initiates protection measures in the event of the parameters exceeding their set limit values;

74. Large early release shall mean a rapid, unchecked release of fission products from the containment into the atmosphere, which takes place prior to the implementation of effective measures to mitigate the consequences of an emergency and poses a potential threat to environment and human health;

75. Probabilistic safety analyses shall mean safety analyses of the reliability of a radiation or nuclear facility systems, which, applying probabilistic methods, identify and assess the extent of possible influences on radiation or nuclear safety, such as component failures, component loss of availability, human errors, undesired impacts from the environment, fires, floods, and earthquakes. Probabilistic safety analysis are classified into three levels:

- probabilistic safety analyses of the first level determine the sequences of events that may lead to reactor core damages, estimate the anticipated frequency of core damage of this kind and identify weaknesses and advantages of safety systems and procedures to avoid such damage;

- probabilistic safety analyses of the second level determine the ways in which radioactive releases from a radiation or nuclear facility may reach the environment, evaluate the extent of such releases and their anticipated frequency, and assess the relative importance of measures to prevent or mitigate such releases;

- probabilistic safety analyses of the third level identify and evaluate the consequences of radioactive releases for the environment and human health;

76. Management shall mean an individual or a group of individuals authorised to manage the overall radiation or nuclear facility or an organisational unit thereof;

77. Impact area shall mean a three-dimensional space adjacent to a radiation or nuclear facility, below and above the facility, determined by the environmental factors that may influence the facility, and to the boundaries of which the facility may influence the environment;

78. Commissioning shall mean the process in which a constructed facility complete with all its SSCs is rendered ready for operation and conformance with design and design bases i.e. conformance with operating conditions is verified;

2. DESIGN BASES

Article 3 (design principles)

(1) The design of a radiation or nuclear facility shall adhere to the following principles:

1. defence-in-depth principle;

2. single-failure principle;

3. independence principle;

4. diversity principle;

5. redundancy principle;

6. fail-safe principle;

7. proven-components principle;

8. graded-approach principle.

(2) The defence-in-depth principle shall mean the application, in design and operation, of several safety measures for a particular protection purpose (e.g. the prevention of radioactive-material releases) to ensure the achievement of the protection purpose even in the event of a failure of one of the safety measures. Safety measures shall be envisaged at different levels of defense-in-depth as far as reasonably achievable. The levels of defense-in-depth are determined by the following objectives:

- prevention of abnormal operation and failures;

- control of abnormal operation and failures;

- control of the accident in order to limit the radiological releases and prevent severe core damage;

- control of the accident with a severe core damage in order to limit the off-site radiological releases;

- mitigation of consequences of significant radiological releases.

(3) The single-failure principle shall mean the postulation in the safety analyses, along with the analysed event, of the additional single failure most unfavourable for radiation or nuclear safety.

(4) The independence principle shall mean the observance of functional and physical separation of safety systems in their design. In the design, the independence principle shall include:

- independence between redundant components;

- independence of system components from the effects of the postulated initiating events so as to prevent any loss of functionality of the safety system or of a safety function required to mitigate the consequences of the event;

- appropriate independence of systems and components of different safety classes in nuclear power plants, classified according to Item 2.1 in Annex 1, which is a constituent part of these Rules, and, in other facilities, according to their design bases;

- independence between safety-relevant and non-safety-relevant systems and components.

(5) The diversity principle shall mean the achievement of any individual safety function through different means. This minimises the risk of common-cause failures and maximises reliability. Diversity shall be observed in the design of safety-related systems and components differing in characteristics and intended for the achievement of the same safety function. Such characteristics may involve application of different operational methods, different physical phenomena, different operational conditions, different equipment manufacturers, etc.

(6) The redundancy principle shall mean the designing of a system in such a way as to provide the achievement of a certain safety function through several equivalent subsystems or components in the required manner. Any failure or unavailability of one subsystem or component shall not prevent the system from achieving the required safety function.

(7) The fail-safe principle shall mean the automatic transition of a safety-related system or component into a state safe for the facility following its failure.

(8) The proven-components principle shall mean the ensuring of system reliability through the application of proven components. Proven components shall mean components that have demonstrated their adequacy in similar operational conditions or are appropriately tested and qualified.

(9) The graded-approach principle shall mean the design of a radiation or nuclear facility in such a way as to apply stricter criteria in the design and safety functions and barriers to prevent releases of radioactivity into the environment to more important facilities i.e. facilities in which an accident is likely to cause more severe consequences to the environment and people than in the case of facilities where accidents are likely to cause less severe consequences.

Article 4 (general design bases)

(1) In the design bases, an investor planning to construct a radiation or nuclear facility, or an operator planning to decommission such a facility, shall:

1. from the list of all postulated initiating events, select, in accordance with the second paragraph of Article 11 of these Rules, those anticipated operational occurrences and design basis events that may influence the safety of the radiation or nuclear facility and of which the probability of occurrence is not negligibly low;

2. ensure compliance with the safety provisions of the safety analysis report, with due consideration given to all the facility lifecycle phases: design, construction, trial operation, operation, deactivation period, safe enclosure mode, decommissioning, closure of a disposal facility or completion of any mining works, in the case of long- term surveillance of a disposal facility the compliance shall be assured by long-term surveillance contractor;

3. demonstrate that the standards and materials applied in the design of the facility are appropriate to ensure safe operation for the foreseen service life of the facility, or, in the case of a disposal facility, also for the period beyond its closure, for as long as the disposal facility shall perform its isolating function;

4. take into account the ageing of SSCs and ensure their achievement of safety functions for the foreseen service life of the facility, or, in the case of a disposal facilit, also for the period beyond its closure, as well as measures set out for in-service maintenance, testing and inspection;

5. ensure the prevention, or, in the event that prevention is impossible, mitigation of excessive exposure to ionising radiation due to design-basis accidents and design extension conditions category A, so there is no need for protective measures, such as iodine prophylaxis, sheltering or evacuation;

6. by design provisions, in all facility states (including decomissioning), ensure minimum quantity and activity of radioactive waste;

7. ensure that doses to population and workers (individual and collective) and impacts on the environment do not exceed the regulatory limits and are as low as reasonably achievable in all the facility states (including decomissioning);

8. ensure protection against radiological impacts of the facility adequate to prevent any risk for health or life of any member of the population significantly higher than if the facility were not present;

9. taking into account the principle of defense-in-depth ensure several levels of defense, including a series of physical barriers to prevent, or in case prevention is not successful, to mitigate uncontrolled releases of radioactive materials into the environment, as well as a combination of safety features that contribute to the effectiveness of the barriers;

10. ensure the prevention of a compromise of the integrity of individual physical barriers;

11. ensure the prevention of a failure of any physical barrier during its performance of its function;

12. ensure the prevention of a collapse of any physical barrier as a consequence of the collapse of another physical barrier;

13. ensure conformance of monitoring and warning systems with the operating requirements of a facility and their suitability in terms of providing clear messages and enabling effective response during design-basis events and accidents;

14. ensure the inclusion, in the design bases, of fire-safety requirements compiled based on an analysis of fire risks and considering the defence-in-depth principle;

15. ensure that the nuclear power plant operator or the operator of a research reactor has a period of 30 minutes from reception of the first characteristic information on an event to the time when the first action to prevent or mitigate the consequences of the event is required. In the meantime, the activations and control of the safety functions shall be automated or accomplished by passive means;

16. in the case of a radioactive-waste disposal facility, ensure that the burden of radioactive materials and radiation on the environment does not exceed the statutory limits according to the normal evolution of a disposal facility.

(2) Without prejudice to the provision of item 14 of the previous paragraph, in the case of the Krško nuclear power plant, the design bases shall ensure that the operator has a period of 15 minutes from reception of the first characteristic information on an event to the time when the first action to prevent or mitigate the consequences of the event is required. Any action by the operator required, according to the design, within the first 30 minutes of an event, shall be substantiated, warranted by written procedures and regularly trained for on a simulator.

Article 5

(design extension conditions of a nuclear power plant)

(1) As part of defence in depth, analysis of Design Extension Conditions shall be undertaken with the purpose of further improving the safety of the nuclear power plant by:

1. enhancing the plant’s capability to withstand more challenging events or conditions than those considered in the design basis;

2. minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.

(2) The operator of a nuclear power plant shall be identify postulated severe accident and with analyses referred in the previous paragraph define measures for their prevention or mitigation of their consequences.

(3) The first and second paragraph of this Article shall also apply to the spent fuel storage, for which all possible measures have to be taken with the goal that a severe accident in such storage becomes extremely unlikely to occur.

(4) Detailed requirements for the Design Extension Conditions are set out in Item 4 of Annex

1 of these Rules.

Article 6

(passive and active safety functions)

(1) In designing a radiation or nuclear facility, the use of passive safety functions is preffered to reduce the dependence on active safety functions, monitoring and human intervention in ensuring safety.

(2) In designing a radioactive waste or a spent fuel disposal the safety after closure and during the period of long-term survaillance shall be ensured by passive means only.

Article 7

(basic safety functions)

(1) In designing a radiation or nuclear facility, the following basic safety functions shall be fulfilled during the operational states, design basis accident and design extension conditions category A:

- sub-criticality;

- heat removal;

- confining radioactive material.

(2) In designing a nuclear facility confinement of radioactive materials shall be ensured for design extension conditions category B. For this purpose, the heat removal from the damaged fuel shall be ensured.

(3) In the case of a disposal or a storage facility, the third item of the first paragraph shall relate to radioactive materials in solid or liquid form, while in the case of a disposal facility, it shall also relate to the normal evolution of a disposal facility.

(4) In designing a radiation or nuclear facility, nuclear-criticality safety shall be ensured through design solutions. When those are not available, the administrative measures can also be used.

Article 8

(SSC design bases)

(1) SSCs shall be designed to prevent the occurrence of design-basis events or to mitigate accident consequences.

(2) Safety-related SSCs shall be identified and classified as important for safety. They shall be designed, manufactured, installed and tested in accordance with quality standards, with due consideration being given to a graded approach in relation to the importance of their safety functions.

(3) In specifying independence, redundancy and diversity requirements for SSCs, the possibility of common cause failures shall be taken into account.

(4) Safety-related SSCs shall be designed taking into account limit conditions imposed by design-basis events.

(5) Safety-related SSCs shall be designed to withstand influences from the environment and to be compatible with such influences and with natural conditions prevailing in the site region of a radiation or nuclear facility in all the facility states and during accidents.

(6) Safety-related SSCs shall be designed and installed in the facility so as to retain their capacity of accomplishing their safety functions even during fires or exposure to explosions.

(7) Following any design-basis event, safety-related SSCs shall prevent any release of radioactive materials into the environment exceeding the regulatory limits. They shall continue to function, despite the postulated initiating events and their consequences.

(8) Safety-related SSCs shall be provided with the means for maintenance, testing, repairing and inspection, as well as periodic inspection of integrity and functionality throughout the operating life of the facility, and in the case of a disposal facility, also in the period beyond its closure, without exposing workers to undue risks or reducing SSC availability. If this is impracticable, proven alternative methods or indirect methods and appropriate measures shall be laid down to compensate for any undetected failures and their consequences.

(9) Safety-related SSCs coming into contact with liquids shall be made of materials of appropriate anti-corrosion properties, and resistant to abrasion and chemical reactions in all facility states and during accidents throughout the service life.

(10) During the decommissioning of the facility the clasification of safety–related SSCs can change depending on the importance for safety. Changes in the classification of SSCs shall follow the procedure laid down for approval of changes in accordance with the law governing radiation protection and nuclear safety.

Article 9

(site characteristics)

(1) The design and siting of a radiation or nuclear facility shall take due account of the characteristics of the site, including the influences on the facility from the site impact area.

(2) The design shall take due account of special environmental loads and conditions to which SSCs may be exposed due to external and internal events, including natural events characteristic for the site region and events associated with human activities including malevolent acts.

Article 10

(normal operation, events and accidents)

(1) The design of a radiation or nuclear facility shall take due account of the conditions of normal operation, anticipated operational occurrences, accidents, and in the case of a disposal facility, of the normal evolution of a disposal facility and of the alternative evolution of a disposal facility.

(2) The design of a nuclear facility shall ensure that the frequency of anticipated operational occurrences is as low as reasonably achievable.

(3) The design of a nuclear facility shall reduce the possibility of escalation of anticipated operational occurrences into accidents.

(4) The boundary conditions for the design of safety-related SSCs of a radiation or nuclear facility shall be specified based on anticipated operational occurrences and design-basis events selected in accordance with item 1 of the first paragraph of Article 0 of these Rules.

Article 11 (postulated initiating events)

(1) The design bases shall include those postulated initiating events, either external or internal, induced by human activity or natural events, the probability of which is not negligibly low or the consequences of which to the environment, population or to personnel are not negligible.

(2) The anticipated operational occurrences and design-basis events shall be selected from all the postulated initiating events according to deterministic or probabilistic methods or combination of both or based on engineering judgment, so as to encompass all the boundary conditions that may be imposed by the postulated initiating events.

(3) Design bases events shall be selected taking into account the characteristics of the facility and experience and analyses of other comparable facilities.

(4) In addition to individual postulated initiating events from the first and second paragraph of this article, design bases shall also take into account probable combinations of internal and external events that may lead to anticipated operatinal occurrences or design-basis events. The selection of such combinations may be based on deterministic or probabilistic methods or engineering judgement.

Article 12

(capability for decommissioning)

(1) A radiation or nuclear facility shall be designed so as to allow its decommissioning after the termination of its operation with minimum radiation loads on personnel and population, and to avoid undue contamination of the environment in the course of decommissioning.

(2) The design of a radiation or nuclear facility shall ensure the keeping of all detailed information on the facility that is required for its decommissioning, which are generated in all the phases of the facility's lifetime, including its siting, design, construction, trial operation, operation and termination of operation, and at the minimum, information on the use of the facility, events and accidents, radionuclide inventory, dose rates and contamination levels.

(3) The information referred to in the previous paragraph shall ensure incorporation of the design and modifications of a radiation or nuclear facility and its operational history into the facility-decommissioning programme.

Article 13 (physical protection)

(1) The design of a radiation or nuclear facility, which holds nuclear or radioactive materials shall ensure conditions relevant for the physical protection, so as to effectively prevent any criminal offences involving threats to safe operation and use of radioactive or nuclear materials.

(2) The provisions of the previous paragraph refer to facilities containing nuclear or radioactive materials in quantities or activities greater than the criteria, which are for nuclear materials laid down in the regulation governing the physical protection of nuclear materials, while, in the case of radioactive materials, are determined by the Government of the Republic of Slovenia on the basis of the law governing radiation protection and nuclear safety.

(3) Physical protection measures of nuclear and radiation safety shall be designed and implemented in a coordinated and integrated manner, decisions and actions related to them, shall ensure that physical protection measures do not adversely affect the nuclear or radiation safety, or that nuclear or radiation safety measures do not adversely affect the physical protection.

Article 14 (cyber security)

Unauthorized access to computer systems of nuclear or radiation facilities and cyber attacks shall be prevented by physical, technical and administrative security measures. Detailed requirements for cyber security are set out in Annex 8 of these Rules.

Article 15 (facility states)

(1) The design bases for a radiation or nuclear facility shall specify the facility states.

(2) According to their probability of occurrence, the postulated initiating events shall be grouped in a limited number of categories. The categories shall include: anticipated operational occurrences, design-basis accidents, and, in the case of nuclear power plants, design extension conditions.

(3) The criteria for classification referred to in the previous paragraph shall be specified, for each category, to ensure that frequent postulated initiating events have minor or no radiological consequences, while the frequency of events with severe radiological consequences shall be very low.

Article 16 (safety analyses)

(1) The accomplishment of the design bases for a radiation or nuclear facility shall be verified through safety analyses. The radioactive waste disposal safety analysis shall include the period of operation of the facility and the period following its closure.

(2) Safety analyses shall consider:

1. the most unfavourable single failure of safety-related equipment, where cases of the failure of passive components need not be taken into account if it can be proven that such a failure is highly improbable and that the event under analysis does not compromise the intended safety function of the component;

2. a period of 30 minutes available to the operator from reception of the first characteristic information on an event to the time when the first action to prevent or mitigate the consequences of the event is required;

3. initial and boundary conditions for the analysed scenarios selected in a conservative manner;

4. operability of non-safety-related systems, including off-site power supply, only in the conditions where such systems further aggravate the event consequences;

5. assumption of the least favourable performance of safety systems for the postulated initiating event;

6. any and all potential failures that may develop due to the postulated initiating event;

7. any uncertainties that may affect the results;

8. the evaluation of the performance and robustness of the facility, system and its components, in the case of radioactive waste disposal facility;

9. an inadvertent human intrusion in the case of radioactive waste disposal facility, with a focus on reducing the likelihood of such an event and the possible consequences. Measures to prevent this incident should not affect the operational and post‐closure safety.

(3) Any exemption from the assumptions referred to in the previous paragraph shall be substantiated.

(4) Safety analyses shall:

- rely on methods, assumptions or arguments which are justified and conservative;

- provide assurance that uncertainties and their impact have been given adequate consideration. This assurance may take the form of conservative assumptions, safety factors or uncertainty and sensitivity analysis;

- give evidence that adequate margins have been included when defining the design basis to ensure that all the design basis events are covered;

- be auditable and reproducible.

(5) The operator of a nuclear power plant or a research reactor shall carry out a probabilistic safety analysis for the facility. In the case of a nuclear power plant, this shall include all three levels of analysis.

(6) The operator of a radiation or nuclear facility other than a nuclear power plant or a research reactor may carry out a probabilistic safety analysis for the facility if the design bases suggest the need for such an analysis.

(7) Without prejudice to the provision of item 2 of the second paragraph of this article, in the case of the Krško nuclear power plant, a period of 15 minutes shall be allowed to the operator from reception of the first characteristic information on an event to the time when the first action to prevent or mitigate the consequences of the event is required.

(8) Without prejudice to the provision of fifth paragraph of this article the Krško nuclear power plant shall develope probabilistic safety analysis for at least the first and second level.

(9) Provision of fifth paragraph of this article shall not be used for a research reactor TRIGA Mark II.

Article 17 (emergency preparedness)

(1) The design bases for a radiation or nuclear facility shall include a protection and rescue plan for emergencies.

(2) Based on an analysis of beyond-design-bases events, the protection and rescue plan referred to in the previous paragraph shall provide special measures to assist the planning of emergency measures. Adequate evacuation routes shall be provided, with emergency illumination, ventilation and other equipment to allow their safe use. The plan shall observe radiological zones, fire-protection arrangements, requirements of safety at work and requirements of physical protection of the facility.

(3) The protection and rescue plan referred to in the first paragraph shall provide warning systems and arrangements to inform site personnel in emergencies. The informing arrangements shall be available in the control room and in the supplementary control room, if provided, with due account being taken of their diversity.

Article 18 (documenting)

The design bases of a radiation or nuclear facility shall be specified in a clear and systematic manner, documented and updated, as appropriate, during the construction, during the period of operation, during the safe enclosure mode, if any, and during decommissioning. In the case of a disposal facility, this provision also applies to the period of long-term surveillance, to reflect the actual condition of the facility.

Article 19

(review of design bases)

(1) The operator of a radiation or nuclear facility shall regularly, and not only as a part of each periodic safety inspection, review the facility design bases, while this provision also applies, in a meaningful manner, to the contractor of a long-term surveillance of a closed disposal facility.

(2) Such a review of design bases shall also be carried out following any operational occurrence relevant for radiation or nuclear safety or in cases where new information relevant for radiation or nuclear safety has arisen (e.g. evaluation of site characteristics, safety analysis and the development of safety standards or practices).

(3) In the review of design bases referred to in the first and second paragraphs of this article, deterministic and probabilistic safety analyses or engineering judgement may be employed to identify the needs and opportunities for improvements, as well as comparison of the design solutions with statutory requirements and good practice.

(4) Depending on the safety relevance of the findings of the review referred to in the first and second paragraphs of this article, the operator shall update SSCs as appropriate or undertake any other measures to ensure radiation or nuclear safety.

Article 20 (special design bases)

In addition to the design bases provided in articles 0 to 07 of these Rules, the following design bases shall apply to specific types of facilities:

- for a nuclear power plant, design bases in Annex 1 to these Rules;

- for a research reactor, design bases in Annex 2, which forms a constituent part of these Rules;

- for a low and intermediate level radioactive waste storage facility, design bases in

Annex 3, which forms a constituent part of these Rules;

- for a spent-fuel storage facility or high level radioactive waste storage facility, design bases in Annex 4, which forms a constituent part of these Rules;

- for a radioactive waste or spent-fuel disposal facility, design bases in Annex 5, which forms a constituent part of these Rules;

- for a disposal facility for mining or hydrometallurgical tailings, design bases in Annex 6, which forms a constituent part of these Rules;

- for an irradiation rig or particle accelerator other than that for medical or veterinary application, design bases in Annex 7, which forms a constituent part of these Rules.

3. GRANTING OF CONSENTS AND PERMITS

3.1. CONSENT FOR FACILITY CONSTRUCTION

Article 21

(contents of an application for consent for construction in an area of limited use of space due to a nuclear facility)

The following shall be enclosed with an application for consent for construction in an area of limited use of space due to a nuclear facility:

1. documentation required by regulations governing building construction for different facilities compiled in accordance with Article 40 of these Rules;

2. evidence demonstrating the compliance of the construction project with the criteria laid down in regulations on areas of limited use of space due to a nuclear facility and the conditions of facility construction in these areas.

Article 22

(contents of an application for consent for construction on a radiation or nuclear facility site)

An application for consent for construction on a radiation or nuclear facility site shall be accompanied with a construction-permit design file compiled in accordance with Article 40 of these Rules. In the case of construction of an non-complex facility, the application shall be accompanied by documentation required by the regulations governing the building construction.

Article 23

(contents of an application for consent for construction of a less important radiation facility)

(1) The following shall be enclosed with an application for consent for construction of a less important radiation facility:

1. an expert report on radiation safety drawn up in accordance with Article 40 of these Rules, demonstrating compliance of the design with the regulations governing protection against ionising radiation;

2. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in privious item;

3. a construction-permit design file compiled in accordance with Article 40 of these

Rules.

(2) The Administration shall approve the documents referred to in item 1 of the previous paragraph as a part of the procedure of granting consent for construction. In the case of a less important radiation facility, intended for medical or veterinary radiation applications, the conformance of the documents shall be subject to approval by the Slovenian Radiation Protection Administration.

Article 24

(contents of an application for consent for construction or mining works on a radiation or nuclear facility)

(1) The following shall be enclosed with an application for consent for construction or mining works on a radiation or nuclear facility:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. a decommissioning programme as referred to in article 50 of these Rules;

3. a proposal for the scope and duration of the pre-operation monitoring of radioactivity according to the provisions of the rules governing radioactivity monitoring;

4. management system documents as referred to in article 55 of these Rules;

5. a radioactive waste or spent-fuel management programme according to the provisions of the rules governing radioactive waste or spent-fuel management;

6. a operational experience program according to the provisions governing the operational safety of radiation and nuclear facilities;

7. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in items 1 to 6 of this paragraph;

8. a construction-permit design file or, in a case of mining works, a mining-works design file, compiled in accordance with Article 40 of these Rules;

9. documentary evidence demonstrating that any subcontractors will comply, during mining works, with radiation or nuclear safety standards equivalent to those undertaken to be complied with by the facility investor or operator;

10. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety, as a separate and confidential document in accordance with the regulations governing data confidentiality.

(2) The Administration shall approve the documents referred to in items 1 to 3 of the previous paragraph as a part of the procedure of granting consent for construction or mining works.

Article 25

(contents of an application for consent for construction of a radioactive waste or spent-fuel disposal facility)

(1) In the case of the facility being a radioactive waste or spent-fuel disposal facility, the following shall be enclosed with the application for consent for construction:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. a safety-analysis report for the disposal facilities covering the period following its closure, the contents of which shall be specified by the Administration as part of the procedure of specification of conditions for environmental-protection consent. this safety-analysis report may be a part of the safety-analysis report referred to in the previous item;

3. a plan for long-term surveillance and maintenance of the disposal facility after its post closure;

4. a decommissioning programme, as referred to in Article 50 of these Rules;

5. a proposal for the scope and duration of the pre-operation monitoring of radioactivity according to the provisions of the rules governing radioactivity monitoring;

6. management system documents as referred to in Article 55 of these Rules;

7. a radioactive waste or spent-fuel management programme according to the provisions of the rules governing radioactive waste or spent-fuel management;

8. a operational experience program according to the provisions governing the operational safety of radiation and nuclear facilities;

9. an opinion by an approved radiation and nuclear safety expert concerning the radiation and nuclear safety of the facility, drawn up based on the documents referred to in items 1 to 8 of this paragraph;

10. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety, as a separate and confidential document in accordance with the regulations governing data confidentiality;

11. evidence that the subcontractors shall, during the construction of the facility, take into account the same standards for radiation or nuclear safety as the operator of the facility.

(2) The Administration shall approve the documents referred to in items 1 to 5 of the previous paragraph of this article as a part of the procedure for granting consent for construction.

3.2. CONSENT FOR THE FACILITY TRIAL OPERATION

Article 26

(contents of the application for consent for trial operation of a radiation or nuclear facility)

(1) The following shall be enclosed with an application for consent for trial operation of a radiation or nuclear facility:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. a trial-operation programme;

3. a radioactive-waste or spent-fuel management programme according to the provisions of the regulation governing radioactive-waste or spent-fuel management;

4. management system documents as referred to in Article 55 of these Rules;

5. a decommissioning programme as referred to in Article 50 of these Rules;

6. a programme of monitoring operational experiences according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

7. a programme of monitoring operational indicators according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

8. a programme of monitoring ageing according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

9. in the case of a nuclear power plant or research reactor, programmes of SSC

qualifications referred to in Item 2.5 of Annex 1 to these Rules;

10. an analysis of fire risks as referred to in Item 3.4 of Annex 1 to these Rules;

11. programmes of SSC maintenance, testing and inspection according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

12. a description of the results of pre-operation monitoring of radioactivity according to the provisions of the regulation governing radioactivity monitoring;

13. an organisational procedure for the modification management according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

14. an opinion by an approved radiation and nuclear safety expert the radiation and nuclear safety of the facility, drawn up based on the safety-analysis report and other documents referred to in items 1 to 13 of this paragraph;

15. documentary evidence of the financial resources, their amount and financial securities to cover any subsidiary action by the state in accordance with the provisions of the act governing protection against ionising radiation and nuclear safety;

16. a list of written procedures covering the use of the facility according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

17. a report on successful completion of pre-operational testing, including: testing in accordance with the technical documents for installation, testing of fluid systems and ventilation systems, cold and hot pressure testing of systems and components, functional and other testing specified in the technical documents;

18. documentary evidence of the quality of installed equipment and materials according to quality-assurance programmes, standards, technical regulations and product- and service-quality regulations;

19. documentary evidence demonstrating that any subcontractors have complied, during the facility construction, with radiation or nuclear safety standards equivalent to those complied with by the facility operator;

20. a declaration by the radiation or nuclear facility operator stating:

- the readiness of the facility for every trial-operation phase in accordance with the programmes and instructions foreseen in the project;

- the readiness, alignment and approval of documents governing the activities in each of the trial-operation phases;

- the availability of qualified personnel, with completed training and an appropriate educational level, to the operator;

- compliance of the completed works with the facility design bases;

21. a protection and rescue plan;

22. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety and amendments having entered into force during the facility construction, as a separate and confidential document in accordance with the regulations governing data confidentiality.

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with the application for consent for trial operation any documents or data that have been submitted in the previous phases and have not changed since.

(3) The Administration shall approve the documents referred to in items 1 to 2 of the first paragraph of this article, or, in the case of a disposal facility, in items 1 to 3 of the first paragraph of this article, as a part of the procedure for granting consent for trial operation.

(4) In the case of a disposal facility for radioactive waste or spend fuel the consent for the facility's trial operation from the first paragraph of this Article shall be construed as permit for disposal of radioactive waste or spent fuel while the possibility to remove waste from the disposal facility and to recover the facility's original state has to be ensured.

3.3. OPERATING LICENCE AND PERMIT FOR TERMINATION OF OPERATION

Article 27

(contents of an application for an operating licence for a radiation or nuclear facility)

(1) The following shall be enclosed with an application for an operating licence for a radiation or nuclear facility:

1. a report on the trial operation;

2. a safety-analysis report made in accordance with Chapter 4.1 of these Rules and amended according to any modifications that have arisen during the trial operation;

3. a radioactive-waste or spent-fuel management programme according to the provisions of the regulation governing radioactive-waste or spent-fuel management;

4. management system documents as referred to in Article 55 of these Rules;

5. a decommissioning programme as referred to in Article 50 of these Rules;

6. a programme for monitoring operational experiences according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

7. a programme of monitoring operational indicators according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

8. a programme of monitoring ageing according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

9. in the case of a nuclear power plant or a research reactor, programmes of SSC

qualifications as referred to in Item 2.5 of Annex 1 to these Rules;

10. an analysis of fire risks as referred to in Item 3.4 of Annex 1 to these Rules;

11. programmes of SSC maintenance, testing and inspection according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

12. an organisational procedure for modification management according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

13. the results of the pre-operation monitoring of radioactivity;

14. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in items 1 to 10 of this paragraph;

15. the design file of the facility as built, which shall reflect the actual condition of the facility and provide information on any modifications or amendments during the facility's trial operation, drawn up in accordance with Article 40 of these Rules;

16. documentary evidence of the financial resources, their amount and financial securities to cover any subsidiary action by the state in accordance with the the act governing protection against ionising radiation and nuclear safety;

17. information concerning the granted operating permit;

18. documentary evidence of the quality of installed equipment and materials according to quality-assurance programmes, standards, technical regulations and product- and service-quality regulations;

19. a list of written procedures covering the use of the facility and in conformance with the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

20. a protection and rescue plan;

21. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety and amendments that have entered into force during the facility's trial operation, as a separate and confidential document in accordance with the regulations governing data confidentiality.

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with the application for an operating licence any documents or data that have been submitted with the application for consent for trial operation and have not changed since.

(3) In the case of a radioactive-waste, spent-fuel, mining or hydrometallurgical-tailings disposal facility, the granting of the operating licence referred to in the first paragraph of this article shall be deemed granting of a permit to deposit radioactive waste, spent fuel, mining or hydrometallurgical tailings generated in the extraction of nuclear mineral materials.

(4) For the facilities referred to in the previous paragraph, the application referred to in the first paragraph of this article shall be accompanied with a programme of closure, which shall provide the measures and procedures for closure and propose phases and deadlines for these works.

(5) The Administration shall approve the documents referred to in items 1 to 2 of the first paragraph of this Article, or, in a case of a disposal facility, in items 1, 2 and 3 of the first paragraph and the previous paragraph, as a part of the procedure of granting an operating licence.

Article 28

(contents of an application for a licence for fresh fuel storage

at the construction site of a nuclear power plant or a research reactor)

(1) The following shall be enclosed with an application for a licence for fresh fuel storage at the construction site of a nuclear power plant or a research reactor:

1. parts of the safety-analysis report of a nuclear power plant or a research reactor under construction, made in accordance with Chapter 4.1 of these Rules, which

shows the nuclear safety of the facility, where the fresh fuel is kept, and which shows that the facility for fresh fuel storage is safe even if the rest of the nuclear power plant or research reactors is not constructed;

2. management program of radioactive waste or spent fuel in accordance with the regulation governing the management of radioactive waste and spent fuel, in part related to the facility for the fresh fuel storage;

3. an analysis of fire risks for a fresh fuel storage as referred to in Item 3.4 of Annex 1 to these Rules;

4. an opinion by an approved radiation and nuclear safety expert concerning the radiation and nuclear safety of the facility, drawn up based on the documents referred to in items 1 to 3 of this paragraph;

5. operating licence for the for the fresh fuel storage issued in accordance with the regulations governing building construction;

6. the as built design of the fresh fuel storage, which shall provide information on any modifications or amendments of the project, drawn up in accordance with Article

40 of these Rules;

7. documentary evidence of the quality of installed equipment and materials according to quality-assurance programmes, standards, technical regulations and product- and service-quality regulations;

8. a list of written procedures covering the use of the facility according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

9. documentary evidence demonstrating that subcontractors, during the facility construction, complied with radiation or nuclear safety standards equivalent to those undertaken to be complied with the facility investor;

10. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety, as a separate and confidential document in accordance with the regulations governing data confidentiality,

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with the application for consent for fresh fuel storage any documents or data that have been submitted in the previous phases and have not changed since.

(3) The Administration shall approve the documents referred to in items 1 of the first paragraph, as a part of the procedure of granting a licence for a fresh fuel storage at the construction site of a nuclear power plant or a research reactor.

Article 29

(SSC categorisation in nuclear power plants)

The operator of a nuclear power plant intending to apply SCC categorisation according to Item

2.2 of Annex 1 to these Rules shall enclose the documents referred to in Item 2.6 of Annex 1 to these Rules to the application provided in articles 26 and 27 of these Rules.

Article 30

(contents of an application for a permit for termination of operation of a radiation or nuclear facility)

(1) The following shall be enclosed with an application for a permit for termination of operation of a radiation or nuclear facility:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. a decommissioning programme as referred to in Article 50 of these Rules;

3. a radioactive-waste or spent-fuel management programme according to the provisions of the regulation governing radioactive-waste or spent-fuel management;

4. management system documents as referred to in Article 55 of these Rules;

5. a programme of monitoring operational experiences according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

6. programmes of SSC maintenance, testing and inspection in relation to the termination of operation according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

7. a proposal for the scope and duration of the post-operation monitoring of radioactivity according to the regulation governing radioactivity monitoring;

8. the results of the operational monitoring of radioactivity;

9. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in items 1 to 8 of this paragraph;

10. a list of written procedures covering the termination of operation of the facility, in conformance with the regulation governing the ensuring of operational safety of radiation and nuclear facilities following the commencement of operation;

11. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety and amendments that have entered into force during the facility's operation, as a separate and confidential document in accordance with the regulations governing data confidentiality.

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with an application for consent for termination of operation any documents or data that have been submitted in the previous phases and have not changed since.

(3) The application referred to in the first paragraph of this article shall be submitted by the facility operator at least two years prior to the planned termination of operation.

(4) In the case of a disposal facility, the application referred to in the first paragraph of this article applies to processing and storage facilities that are to be decommissioned.

(5) The Administration shall approve the documents referred to in items 1 to 3 of the first paragraph of this article as a part of the procedure for granting a permit for termination of operation.

3.4. PERMIT FOR COMMENCEMENT AND COMPLETION OF THE DECOMMISSIONING OF A FACILITY

Article 31

(contents of an application for consent for decommissioning or for mining works on a radiation or nuclear facility)

The following shall be enclosed with an application for consent for decommissioning or for a permit for mining works on a radiation or nuclear facility:

1. a safety-analysis report for decommissioning made in accordance with Chapter 4.1 of these Rules;

2. a decommissioning programme as referred to in Article 50 of these Rules;

3. an opinion by an approved radiation and nuclear safety expert concerning the radiation and nuclear safety of the facility, drawn up based on the documents referred to in items

1 and 2 of this Article;

4. a construction-permit design file or, in the case of mining works, a mining-works design file, compiled in accordance with Article 40 of these Rules;

5. documentary evidence demonstrating that any subcontractors comply, during the decommissioning works, with radiation or nuclear safety standards equivalent to those undertaken to be complied with by the facility operator.

Article 32

(contents of an application for a permit

for commencement of decommissioning of a facility)

(1) The following shall be enclosed with an application for a permit for commencement of decommissioning of a radiation or nuclear facility:

1. a safety-analysis report for decommissioning made in accordance with Chapter

4.1.1 of these Rules;

2. management system documents as referred to in Article 55 of these Rules;

3. a decommissioning programme as referred to in Article 500 of these Rules;

4. a radioactive-waste or spent-fuel management programme according to the regulation governing radioactive-waste or spent-fuel management;

5. programmes of SSC maintenance, testing and inspection according to the regulation governing the ensuring of operational safety of radiation and nuclear facilities during operation;

6. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in items 1 to 5 of this paragraph;

7. the results of radioactivity monitoring following the termination of operation;

8. a decommissioning project-design file made based on the programme of decommissioning and in accordance with the provisions of Article 40 of these Rules;

9. information concerning the granting of the construction permit for removal of the radiation or nuclear facility;

10. a list of written procedures covering the commencement of decommissioning in conformance with the regulation governing the ensuring operational safety of radiation and nuclear facilities in operation;

11. documentary evidence demonstrating that any subcontractors will comply, during the decommissioning works, with radiation or nuclear safety standards equivalent to those undertaken to be complied with by the facility operator;

12. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety and amendments that have entered into force following the termination of operation of the facility, as a separate and confidential document in accordance with the regulations governing data confidentiality.

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with the application for a permit for commencement of decommissioning any documents or data in effect for the facility that have not changed since their submittal.

(3) In the case of a disposal facility, the application referred to in the first paragraph of this article applies to the processing and storage facilities that will be decommissioned.

(4) The Administration shall approve the documents referred to in items 1, 3 and 4 of the first paragraph of this article as part of the procedure for granting a permit for the commencement of decommissioning.

(5) A permit for commencement of decommissioning of a radiation or nuclear facility shall be granted following the granting of the construction permit for the facility's removal.

(6) Until the permit for commencement of decommissioning of a radiation or nuclear facility is granted, the decommissioning works on the radiation or nuclear facility may not commence solely on the basis of the construction permit for facility removal.

Article 33

(contents of an application for a permit for completion of decommissioning)

(1) The following shall be enclosed with an application for a permit for completion of the decommissioning of a radiation or nuclear facility other than a disposal faciltiy:

1. a safety-analysis report on the completion of decommissioning of the radiation or nuclear facility;

2. a report on the final inspection of the radiological condition of the facility site demonstrating that radioactivity levels, including dose rates and contamination of soil with alpha, beta and gamma emitters, are below the statutory limits;

3. certificates of disposal, treatment or reuse of all the radioactive materials generated during decommissioning;

4. the results of radioactivity monitoring during decommissioning;

5. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety for the facility, drawn up based on the documents referred to in items 1 to 4 of this paragraph;

6. the type and method of filing of documents on all past activities related to the facility at the site.

(2) In the case of a disposal facility, the permit for completion of decommissioning applies to the processing and storage facilities that will be decommissioned.

(3) The safety-analysis report on completion of the decommissioning of a radiation or nuclear facility referred to in item 1 of the first paragraph of this article shall include:

1. a facility description;

2. decommissioning objectives;

3. clearance criteria;

4. decommissioning activities;

5. a description of SSCs, buildings or areas not cleared;

6. the final radiological condition;

7. clearance of the site;

8. wastes;

9. personnel doses;

10. a description of events in the course of decommissioning;

11. a description of lessons learned in the course of decommissioning.

(4) The Administration shall approve the documents referred to in items 1 and 2 of the first paragraph of this article as part of the procedure of granting a permit for completion of the decommissioning of the radiation or nuclear facility.

3.5. OTHER CONSENTS AND PERMITS FOR SPECIFIC FACILITIES

Article 34

(closure of a radioactive-waste or spent-fuel disposal facility)

The closure of a radioactive-waste or spent-fuel disposal facility shall involve those works on the depot facilities of the disposal facility necessary to bring them into a condition suitable for long-term surveillance and maintanance.

Article 35

(contents of an application for consent for closure works on a radioactive-waste or spent-fuel disposal faciltiy)

(1) The following shall be enclosed with an application for consent for closure works on a radioactive-waste or spent-fuel disposal faciltiy:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. management system documents as referred to in Article 55 of these Rules;

3. the programme of closure for the disposal faciltiy, which shall provide measures and procedures for closure and propose phases and deadlines for these works;

4. the results of the operational monitoring of radioactivity;

5. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety for the facility, drawn up based on the documents referred to in items 1 to 4 of this paragraph;

6. the as built design of the facility, which shall reflect the actual condition of the facility, drawn up in accordance with Article 40 of these Rules;

7. the design file for the execution of closure works, drawn up in accordance with

Article 040 of these Rules;

8. a list of written procedures covering the closure of the facility in conformance with the regulation governing the ensuring of operational safety of radiation and nuclear facilities;

9. a physical-protection plan made in accordance with the act governing protection against ionising radiation and nuclear safety, and amendments that have entered into force during the facility's operation, as a separate and confidential document in accordance with the regulations governing data confidentiality;

10. documentary evidence demonstrating that any subcontractors comply, during the decommissioning works, with radiation or nuclear safety standards equivalent to those undertaken to be complied with by the facility operator.

(2) Without prejudice to the provisions of the previous paragraph, the applicant does not need to enclose with an application for consent for closure of a radioactive-waste or spent-fuel disposal facility any documents or data in effect for the facility that have not changed since their submittal.

(3) In the case of the surface processing facilities of a disposal facility, the application referred to in the first paragraph of this article applies to the granting of consent for completion of their decommissioning.

(4) In the case of the underground processing facilities of a disposal facility constructed according to the methods of mining works, the application referred to in the first paragraph of this article applies to the granting of consent for the mining works.

(5) In the case of a disposal facility constructed in a modular manner, closure works may be undertaken in phases, by depot units.

Article 36

(contents of an application for a permit for closure of a radioactive-waste or spent-fuel disposal facility)

(1) The following shall be enclosed with an application for a permit for closure of a radioactive-waste or spent-fuel disposal facility:

1. a safety-analysis report for the closure of a radioactive-waste or spent-fuel disposal facility, made in accordance with Chapter 4.1 of these Rules;

2. a safety-analysis report on disposal's facilities for the period following the closure of the disposal facility, made in accordance with Chapter 4.1 of these Rules;

3. a plan for long-term surveillance of the disposal's facilities;

4. the results of the monitoring of radioactivity during the closure works;

5. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety of the facility, drawn up based on the documents referred to in items 1 to 4 of this paragraph;

6. the design file of the completed works, drawn up in accordance with Article 40 of these Rules;

7. records of the radioactive waste or spent fuel disposed of;

8. a declaration by the operator stating the compliance of the executed works with project documents and documentary evidence of the quality of the installed materials in accordance with the quality-assurance programme.

(2) The Administration shall approve the documents referred to in items 1, 2 and 3 of the previous paragraph as part of the procedure for granting a permit for the closure of the disposal facility.

Article 37

(contents of an application for a permit for the closure

of a disposal facility of mining or hydrometallurgical tailings)

(1) The following shall be enclosed with an application for a permit for closure of a disposal facility of mining or hydrometallurgical tailings:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. a plan for long-term surveillance of the disposal facility of mining or hydrometallurgical tailings;

3. records of the mining or hydrometallurgical tailings disposed of;

4. the results of the monitoring of radioactivity during the closure works;

5. an opinion by an approved radiation and nuclear safety expert concerning the radiation and nuclear safety of the facility, drawn up based on the documents referred to in items 1 to 4 of this paragraph;

6. the design file of the completed works, drawn up in accordance with Article 40 of these Rules;

7. a declaration by the operator stating the compliance of the executed works with project documents and documentary evidence of the quality of the installed materials in accordance with the quality-assurance programme.

(2) The Administration shall approve the documents referred to in items 1 and 2 of the previous paragraph as part of the procedure of granting a permit for closure of the disposal facility.

Article 38

(contents of an application for a permit for completion of mining works)

(1) The following shall be enclosed with an application for a permit for completion of mining works under the termination of extraction of nuclear mineral resources:

1. a safety-analysis report made in accordance with Chapter 4.1 of these Rules;

2. certificates of disposal, reprocessing or reuse for all radioactive materials generated in the course of mining works;

3. a report on the final inspection of the radiological condition of the facility site demonstrating that radioactivity levels, including dose rates and contamination of soil with alpha, beta and gamma emitters, are below the statutory limits;

4. an opinion by an approved radiation and nuclear safety expert concerning the aspects of safety for the facility, drawn up based on the reports referred to in items

1 and 3 of this paragraph;

5. a final expert report on the extraction of mineral resources including information on any residual reserves at the site.

(2) The Administration shall approve the safety-analysis report and the final-inspection report referred to in the previous paragraph as part of the procedure for granting a permit for the completion of mining works.

(3) A permit for completion of mining works shall be granted following granting of the consent for termination of extraction of mineral resources.

3.6. REFERENCE DOCUMENTS AND PROJECT DOCUMENT FILES

Article 39 (reference documents)

At the request of the Administration, the investor or operator of a radiation or nuclear facility shall make available for inspection the consent or permit-application reference documents referred to in articles 21 to 28, articles 30 to 33 and articles 35 to 38 of these Rules.

Article 40

(contents, method of compiling and revision of project documents)

(1) The contents, method of compiling and revision of project documents for less important radiation facilities, radiation facilities and nuclear facilities and other facilities in areas of limited use of space due to nuclear facilities shall conform, in the case of construction subject to the act governing building construction, with the provisions of the regulation governing project documents, and in the case of mining works subject to the act governing mining works, with the provisions of the rules governing the method of compilation, sequence, contents and revision of mining-works project documents.

(2) A project document file shall be compiled in compliance with the design bases made pursuant to the provisions of Chapter 2 of these Rules.

(3) For less important radiation facilities, radiation facilities and nuclear facilities, the project document file, describing project solutions with direct or indirect relevance to nuclear or radiation safety, shall be reviewed in full.

(4) The investor of a radiation or nuclear facility shall submit the design bases referred to in the first paragraph of Article 0 of these Rules to an authorised nuclear and radiation safety expert for expert opinion.

Article 41

(instructions for use, operation and maintenance)

In the case of radiation and nuclear facilities, the granting of the relevant permit referred to in chapters 3.2, 3.3, 3.4 and 3.5 of these Rules shall be deemed to mean the compilation of instructions laying down the rules of use, operation and maintenance of the facility, pursuant to the act governing building construction.

4. SAFETY DOCUMENTS

4.1. SAFETY-ANALYSIS REPORT

Article 42

(purpose and appplication)

(1) The safety-analysis report is an integral document providing the bases for the regulatory surveillance of the safety of a radiation facility or nuclear facility or facility of the national infrastructure in accordance with the act governing radiation protection and nuclear safety, after closure of disposal.

(2) The safety-analysis report is first compiled by the facility investor, to demonstrate the safe operation of a facility, and is approved by the Administration as part of the procedure of granting consent for construction of the radiation or nuclear facility.

(3) The safety-analysis report shall demonstrate with justification that the facility fulfils relevant safety requirements. The safety report shall convey a clear understanding and traceability of justifications, choices and decisions that affect safety. It shall provide information on the facility at a level of detail allowing independent assessment of the safety of the facility.

(4) The operator of a radiation or nuclear facility shall apply the safety-analysis report as a basis for continuous support of safe operation of the facility. The safety-analysis report shall also serve as a basis for assessment of the potential influences of any modifications to the facility, changes in the environment or methods of the facility's operation on the safety of the facility.

(5) The operator of a radiation or nuclear facility shall update the safety-analysis report at least once per year to reflect modifications to the facility, and to consider all new knowledge and facts, including the information related to the characteristics of the site and the site environment, as well as the changes based on new regulatory requirements. The update shall also take into account the operator's own experience, new regulatory requirements, new standards or new ways of utilizing them, and development of science and technology in a timely manner after the new information is available and applicable.

Article 43

(contents of a safety analysis report)

(1) A safety-analysis report of a radiation facility, nuclear facility or facility of the national infrastructure in accordance with the act governing radiation protection and nuclear safety, under construction, trial operation, operation, following termination of operation, in the safe enclosure mode, under decommissioning or long-term surveillance and maintenance in case of disposal facility, shall contain:

1. a site description, a general description of the facility and its normal operation and a description of how the facility's safety is ensured;

2. a description of the programme for trial operation;

3. a description of the technical characteristics of the radiation or nuclear facility and a description of performance in all the states of the facility;

4. a description of the facility's design and of the accomplishment of basic safety objectives, a description of the design bases of the radiation or nuclear facility and a description of their methods of fulfilment;

5. a detailed description of safety functions, of all safety systems, of safety-related SSCs, their design bases and the performance of all safety-related SSCs in all states of the facility;

6. a list of regulations and standards applied as the basis for descriptions and safety analyses covered in the safety-analysis report;

7. a description of the internal organisational set-up of the facility operator intended for the ensuring of nuclear safety;

8. an assessment of safety aspects related to the facility's siting;

9. a description of safety analyses undertaken to assess the safety of the radiation or nuclear facility in response to anticipated operational occurrences, design-basis events, and for nuclear power plants also the extended design conditions with a comparison against safety criteria and radiological release limits. Safety margins shall be described as well.;

10. a description of probabilistic safety analyses carried out in accordance with the

Article 16 of these Rules;

11. a description of emergency operating procedures, as well as of severe accident management guidelines for facilities, in which severe accident can occur;

12. a description of the protection against internal fires in accordance with item 3 of

Annex 1 of these Rules;

13. a description of the protection and rescue plan for the facility and of the operator's internal organisational set-up in emergency events and of alignment with the national protection and rescue plan in case of nuclear or radiological accident;

14. a description of the measures providing for SSC inspection, testing and surveillance, a description of the programme of application of operational experiences and a description of the ageing-management programme;

15. a description of the training and education of the personnel;

16. operating conditions and limits of safe operation compiled in accordance with articles 46 and 47 of these Rules, and technical bases explaining expert bases for each operating condition or limit;

17. a description of the strategy for protection against radiation, a description of the methods and measures for protection of exposed personnel against ionising radiation, including an assessment of their protection against radiation and an assessment of the population and environment exposure;

18. a description of any radioactive and nuclear materials and other sources of radiation;

19. a description of the radioactive-waste and spent-fuel management programme;

20. a description of all activities in the facility's operational phase planned to facilitate termination of operation and decommissioning;

21. a description of the quality-assurance and management programme;

22. an outline of the physical protection of the facility, nuclear and radioactive substances;

23. the anticipated and maximum allowable releases of radioactive substances into the environment;

24. a programme of meteorological measurements and radioactivity monitoring in operation;

25. in the case of a radioactive-waste disposal facility, a spent-fuel disposal facility, a hydrometallurgical-tailings disposal facility or a mining-tailings disapoal facility, a plan for long-term surveillance.

(2) The descriptions, assessments and arrangements mentioned in the safety-analysis report shall consider the site as a whole, to take into account hazards, which may challenge all installations within a short period of time and which arise from harmful interactions between installations.

Article 44 (reference documents)

(1) In assessing radiation or nuclear facility safety, chapters of the safety-analysis report may refer to reference documents.

(2) The reference documents shall be mutually aligned and aligned with the safety-analysis report and with other project documents of the radiation or nuclear facility.

Article 45

(arrangement of information in a safety-analysis report)

(1) A safety-analysis report shall be arranged according to the graded-approach principle: matters that are more important for safety shall be elaborated in more detail and more comprehensively, and less important matters in less detail.

(2) Information shall be arranged in the safety-analysis report as follows:

1. each chapter of the safety-analysis report shall be an integral topical whole;

2. if the safety-analysis report is divided into several volumes, each volume shall be provided with a table of contents for the whole report;

3. information in drawings, diagrams and schematics shall be legible, with all symbols and abbreviations explained.

(3) To amend the safety-analysis report, the page, which includes a change, shall be substituted with a new, modified page, where revision number shall be indicated.

4.2. OPERATING CONDITIONS AND LIMITS FOR SAFE OPERATION

Article 46

(bases for operating conditions and limits)

(1) The operating conditions and limits, for safe operation (hereinafter reffered as operating conditions and limits) as included in the safety-analysis report referred to in Article 43 of these rules shall:

1. be based on design bases, testing results and safety analyses, with allowance for the uncertainties of analyses of this kind;

2. ensure safe operation of the radiation or nuclear facility in compliance with design bases and the safety-analysis report, while in the case of a radioactive waste disposal, also ensure compliance with the requirements for safety after its closure;

3. specify conditions that must be fulfilled to prevent the occurrence of circumstances that may lead to an accident and to mitigate consequences of a potential accident;

4. contain the requirements concerning the performance of safety systems and safety functions in all states of the facility;

5. ensure preparedness for managing design extension conditions.

(2) Operating conditions and limits shall be specified for all the operating states of the facility.

In the case of a nuclear power plant, such states include the states of power operation, shutdown and refuelling, and any of the intermediate states, as well as any

circumstances arising during maintenance and testing.

Article 47

(contents of operating conditions and limits)

(1) Operating conditions and limits shall contain:

1. a definitions of terms;

2. safety limits;

3. parameter limiting settings for safety systems;

4. limiting conditions for operation and requirements for minimum performance of equipment, including requirements of the extent of operation or the operational readiness of the radiation or nuclear safety related SSCs;

5. necessary measures in cases of exceeded operating conditions and limits, and the time available for taking such measures;

6. requirements for the testing, calibration and inspection of SSCs, which ensure that the SSC can fulfill its functionality; and

7. requirement for the minimum number of licensed personnel to ensure safe operation in different states of the facility.

(2) Operating conditions and limits for a radioactive waste or a spent fuel storage shall contain:

1. the environmental conditions within the storage (e.g., temperature, humidity, contaminants);

2. the effects of heat generation from waste or spent fuel, covering both each individual waste and spent fuel packages or unpackaged spent fuel elements as well as the entire storage;

3. potential effects of gas generation from waste or spent fuel, in particular the hazards of fire ignition, explosion, waste or spent fuel package or unpackaged spent fuel elements deformations and radiation protection aspects;

4. measures to prevent criticality, which includes both individual waste and spent fuel packages or unpackaged spent fuel elements as well as the entire storage,valid during normal and abnormal operation and during accidents; and

5. suitability of the storage for handling and retrieval.

Article 48

(determination of safety limits, operating conditions and safety-system settings)

(1) Appropriate margins shall be provided between the safety-limit values, safety-parameter limiting settings and alarm levels and operating conditions, to avoid frequent undue actuation of safety systems.

(2) Safety limits shall be determined in a conservative manner, with due consideration of the assumptions and uncertainties of safety analyses.

4.3. RADIATION OR NUCLEAR FACILITY DECOMMISSIONING PROGRAMME

Article 49

(records for decommissioning)

A radiation or nuclear facility operator shall keep, in all the phases of the facility's operation and decommissioning, appropriate records to keep the inventory of all radioactive substances in the facility and thereby facilitate decommissioning.

Article 50

(programme of decommissioning)

(1) The programme of decommissioning of radiation or nuclear facility (hereinafter referred as programme of decommissioning) shall be based on safety analyses, assessment of the radiological condition of the facility and up-to-date facility data.

(2) To ensure radiation and nuclear safety, the development of a decommissioning programme shall follow the graded-approach principle so that the strategy of decommissioning and corresponding plans are aligned with the complexity of the facility, inventory of radioactive substances contained within and the stage of the facility's life- cycle in which the programme is being developed.

(3) The decommissioning programme shall set out written procedures for all decommissioning works relevant for radiation or nuclear safety and describe the personnel needs for decommissioning.

(4) The decommissioning programme shall identify the main existing systems and equipment used during decommissioning and ensure that they are available. It shall identify possible changes and replacement of these systems and the need for new structures for the implementation of decommissioning and waste management.

(5) The decommissioning programme shall be regularly reviewed and updated, at a minimum within the periodic safety review of the radiation or nuclear facility. Changes in the strategy of decommissioning, developments in technology, amendments in legislative provisions, modifications to the facility that may have impacts on the facility's decommissioning, progress of the decommissioning works, deviations from the planned programme and the needs for decommissioning realisation shall be taken into account.

(6) The facility's decommissioning programme shall be reviewed and updated according to the procedure laid down for the approval of modifications in accordance with the act governing protection against ionising radiation and nuclear safety.

(7) The decommissioning programme shall minimise the dependence of the radiation or nuclear safety of the facility, following its transition to the safe enclosure mode, on active systems.

(8) If several facilities are located at the same site it shall be ensured that in each facility decommissioning plan any interactions and interdependencies between the facilities are taken into account.

(9) The radiation or nuclear facility operator shall in the decommissioning program document the decommissioning strategy of the facility, including a description of the options, overall timescales for the decommissioning of the facility and the end‐state after completion of all decommissioning activities. The reasons for the preferred option shall be explained, and options not involving immediate dismantling shall be rigorously justified.. The decommissinoing strategy shall be consistent with existing related national strategy and the regulations governing the decommissioning or radioactive waste management.

(10) The decommissioning programme shall also provide for a programme of monitoring and surveillance, to be made prior to transfer of the facility to the safe enclosure mode, to ensure safety in that phase and allow for final decommissioning in the future.

(11) In addition to the provisions of the first to seventh paragraphs of this article, the development of the decommissioning programme for the Krško nuclear power plant shall also be in conformance with the clauses of the Agreement between the Government of the Republic of Slovenia and the Government of the Republic of Croatia on the Arrangement of the Status and other Legal Relationships under the Investment in the Krško Nuclear Power Plant, its Exploitation and Decommissioning (Official Gazette of RS - International Agreements, No. 23/03).

Article 51

(contents of decommissioning programme for a radiation or nuclear facility) (1) The decommissioning programme referred to in the previous article shall cover:

1. a description of the facility;

2. the decommissioning strategy;

3. a description of project management;

4. a description of decommissioning activities;

5. a description of inspections and maintenance;

6. a description of waste management;

7. costs of decommissioning and source of funding;

8. a safety assessment;

9. a description of impacts on the environment;

10. a description of safety at work;

11. a description of quality management;

12. a protection and rescue plan;

13. a description of physical protection of nuclear and radioactive substances;

14. a final inspection of the radiological condition.

(2) The decommissioning program, attached to the application for approval of the construction by licensee shall show that the decommissioning is feasible and that it can be safely conducted using proven techniques or ones being developed.

5. MANAGEMENT SYSTEM

Article 52

(integrated management system)

(1) Management of the investor or operator of a radiation or nuclear facility shall establish, implement and continually improve an effective and integrated management system to ensure radiation and nuclear safety.

(2) The integrated management system from the previous paragraph of this Article shall integrate all elements of management, including safety, security, quality, health, environmental and economic elements, taking into consideration societal factor with the view that safety is not compromised.

(3) The integrated management system from the first paragraph of this Article shall ensure the achievement and continuous improvement of nuclear and radiation safety of the facility by:

- bringing together all requirements for managing the radiation or nuclear facility;

- describing the planned and systematic actions necessary to ensure that all requirements are met;

- ensuring that the requirements relating to health, environmental, security, quality, economic and societal elements are not considered separately from the requirements for radiation and nuclear safety in order to avoid any negative impact of other requirements on radiation and nuclear safety.

(4) Safety of a radiation or nuclear facility shall be paramount within the management system, overriding all other demands. Safety aspects shall be overriding priority in all decisions.

(5) Management of the investor or operator of a radiation or nuclear facility shall ensure that the management system includes the conditions of normal operation, anticipated operational events and possible accidents and takes into account the safety in the design, construction, operation, decommissioning and closure of a nuclear and radiation facility, and in the case of radioactive waste disposal facility also after the closure.

(6) The management system shall base on:

- regulatory requirements governing nuclear and radiation safety;

- compliance with formal agreements with stakeholders;

- standards and guidelines, adopted for use by the investor or operator of a radiation or nuclear facility.

(7) The investor or operator of a radiation or nuclear facility shall be able to demonstrate effective fulfilment of its management system requirements.

(8) The investor or operator of a radiation or nuclear facility shall ensure that decisions influencing radiation or nuclear safety are adopted on time and that, before adopting them the analyses and consultations are performed to consider all relevant safety aspects. Any safety-related matters shall be reviewed by qualified experts that are not directly involved in the preparation or adoption of the decisions.

(9) The investor or operator of a radiation or nuclear facility shall provide a system for continuous monitoring and assuring of radiation and nuclear safety in order to maintain safety and improve it if necessary.

(10) The investor or operator of a radiation or nuclear facility shall ensure that relevant operating experiences, international development of safety standards and new knowledge gained through research and development projects are analyzed and continuously used to improve radiation and nuclear safety of the facility as well as the performance of its personnel.

Article 53 (safety culture)

(1) The investor or operator of a radiation or nuclear facility shall with the management system:

- ensure that management and personnel promote activities to ensure the safety and contribute to the continuous improvement of safety culture;

- establish and support the desired and expected behaviours and attitudes, that promote a strong safety culture, whereby the desired and expected attitudes and behaviours also apply to suppliers;

- ensure that safety related tasks implemented by individuals and teams are carried out safely and successfully taking into account interactions between individuals, technology and organization;

- provide the means by which the organization continuously seeks to develop, upgrade and improve its safety culture.

(2) All individuals in the organization of the investor or operator of a radiation or nuclear facility shall support and foster a strong safety culture by reinforcing the following:

- both individual and collective commitment to safety;

- an acceptance of personal responsibility for safety;

- an organizational culture that encourages trust, collaboration and free communication and the reporting of human and organizational problems;

- the reporting of any deficiencies in SSCs to avoid degradation of safety;

- the prompt acknowledgement of identified problems, feedback for identified problems and suggestions for improvement;

- the means by which the organization continuously seeks to develop and improve safety and safety culture;

- the allocation of responsibilities and accountabilities of organizations and individuals for safety at all levels;

- the measures to encourage a questioning and learning attitude and critical thinking at all levels of the organization;

- measures to discourage complacency;

- a common understanding of the key aspects of safety and safety culture within the organization;

- an awareness of the risks and hazards relating to the work and to the work environment, and an understanding of potential consequences of such hazards;

- safety driven conservative decision making in the implementation of all activities.

(3) The management of the operator of a radiation or nuclear facility shall regularly commission an independent assessment and self-assessment of safety culture and its own management.

(4) The management of the operator of a radiation or nuclear facility shall communicate the results of the assessments from the previous paragraph to all employees and shall ensure continuous improvement and encourage open communication, collaboration, questioning, critical thinking and continuous learning of employees at all levels of the organization.

(5) The operator of a radiation or nuclear facility shall ensure that its suppliers and subcontractors whose work can affect the safety of a radiation or nuclear facility, carry out their activities in accordance with the first and second paragraph of this article.

Article 54

(graded approach of the management system)

(1) The principles for grading the application of the management system requirements shall be applied to the products, services and activities of all processes related to radiation and nuclear safety.

(2) The criteria and requirements for each grade arising from graded approach, which exploit available resources shall be documented and included in the management system. The following shall be taken into account:

- the significance and complexity of a single process or activity;

- the potential hazards and potential safety impacts (risks) and radiation effects associated with the implementation of processes or activities;

- the potential harmful impacts and consequences on safety if the process or activity is improperly carried out or if an unexpected event occurs.

Article 55

(documentation of the management system) (1) The management system shall be documented.

(2) Documentation of the management system shall include as a minimum:

1. a policy statement on management policy and objectives, including the values and expectations of management;

2. a safety policy stating that the priority is given to the protection of public and the environment against ionizing radiation sources;

3. a description of the organizational structure of the investor or operator;

4. a description of how the management system complies with the regulatory requirements relating to the activity of the investor or operator;

5. a description of the responsibilities, levels of authorities and interactions of those managing, performing and assessing work;

6. determination of the responsibilities and the necessary arrangements to ensure safety;

7. a description of the main processes;

8. a description of how the individual activities are developed, reviewed, implemented, documented, verified and improved;

9. a description of the interactions with interested parties;

10. a description of the oversight over the work of subcontractors/suppliers;

11. a description of recording and reviewing knowledge, information and data on all matters relating to safety, as well as a description of method for managing and keeping these records;

12. a description of the requirements for ensuring the transfer of knowledge to personnel in the different phases of the facility.

(3) In addition to the above documents, management system documents may also include procedures, instructions, specifications, drawings, training materials and other texts that describe processes, specify requirements or establish product specifications.

(4) The documentation of the management system shall be drawn up as to be understandable to those who use it. Documents shall be controlled, valid, regularly reviewed and updated, readable, readily identifiable and easily available at the point of use.

(5) The documentation of the management system shall reflect the characteristics of the organization and its activities and be proportionate to the complexity of processes and their interactions.

Article 56

(archives of documentation of nuclear and radiation faciliy)

(1) The operator of a radiation or nuclear facility shall determine retention periods of documentation in the internal documents in line with the significance for radiation and nuclear safety, taking into account the following:

- five years for documentation, which is less important for radiation and nuclear safety;

- the operational lifetime of nuclear or radiation facility for documentation, which is important for radiation protection and nuclear safety;

- retention of documentation after the shutdown of a nuclear or radiation facility, if it is so for certain types of documents specified by other regulations.

(2) The operator of a radiation or nuclear facility shall archive documentation from the previous paragraph in suitable climatic conditions, protected against burglary, fire, water, biological, chemical, physical and other harmful effects, and ensure the availability of the entire retention period.

(3) The first and second paragraph of this Article shall apply meaningful to the contractor of a long-term surveillance of a disposal.

Article 57 (management commitment)

(1) Management at all levels in the organization shall demonstrate its commitment to the establishment, implementation, assessment and continuous improvement of the management system and shall allocate resources for the implementation of these activities.

(2) Management shall establish and develop a common personal and organizational values and expectations of the organization to support the implementation of the management system and promote safety culture.

(3) Management at all levels shall inform the employees with the need to adopt common personal values, organizational values and expectations of the organization and to carry out activities in accordance with the management system.

(4) Management at all levels shall foster the involvement of all employees in the implementation and continuous improvement of the management system.

(5) The management of the operator of a radiation or nuclear facility shall ensure that it is clear, when, how and who makes decisions in the context of the management system.

Article 58

(interaction with interested parties)

(1) The management of the operator of a radiation or nuclear facility shall identify the interested parties for its organization and define a strategy for communicating with them.

(2) The management of the operator of a radiation or nuclear facility shall ensure in the management system appropriate, timely and effective communication with interested parties to recognize their expectations and to ensure that interested parties are informed timely with the important information.

Article 59 (management policy)

(1) The management of the operator of a radiation or nuclear facility shall establish and develop organizational management policy and document it in the management system.

(2) The management policy of the previous paragraph shall be adjusted to the organization's activities and shall support the safety policy set out in Article 60 of these Rules.

Article 60 (safety policy)

(1) The management of the investor or operator of a radiation or nuclear facility shall issue as part of the management system a written safety policy document (hereinafter: safety policy) to demonstrate its commitment to implementing a high level of radiation and nuclear safety performance.

(2) Safety policy shall be issued as a separate document or a significant part of the overall policy of the organization.

(3) The safety policy shall:

1. specify the investor’s or operator’s commitment to provide appropriate resources to accomplish the set objectives;

2. clearly establish an overriding priority to safety in all activities in the facility;

3. include a commitment to continuously improve safety;

4. demand directives for implementing the safety policy and set out the method of monitoring its efficiency;

5. contain safety objectives, formulated in such a way that they can be easily monitored, followed up and taking appropriate measures if necessary by the facility management;

6. contain key directives relating to the cyber security;

7. contain requirements for continuous improvement of radiation protection and nuclear safety on the basis of:

- regular (permanent activity aimed at reviewing and analyzing facility project and its operation, and identify opportunities for improvement) and comprehensive assessment of the safety of a radiation or nuclear facility, taking into account operating experience, R&D in the field of nuclear and radiation safety as well as new developments in science and technology;

- the timely introduction of identified improvements, if it is proved to be meaningful;

- immediate dealing with new data that may be relevant for the safety of a nuclear or radiation facility;

8. contain the aspects of protection and control of nuclear materials and safeguards.

(4) Safety policy shall be communicated to all facility personnel. Personnel involved in tasks important for safety shall understand safety policy and be involved in its implementation.

(5) The safety policy shall also be communicated to subcontractors in such a way that they understand the operator’s expectations and are able to fulfil them.

(6) The investor or operator shall, at regular intervals, shorter than the interval of periodic safety reviews, verify the adequacy and efficiency of its safety policy.

Article 61 (planning)

(1) The management of the investor or operator of a radiation or nuclear facility shall determine the strategies, goals, objectives and plans of the organization that are consistent with the organization's safety policy from the previous Article.

(2) Strategies, goals, objectives and plans of the organization of the previous paragraph:

- shall be developed with the participation of all employees;

- shall not compromise nuclear and radiation safety;

- shall be established in a comprehensive way that their overall effects on safety are understandable and manageable.

(3) The management of the investor or operator of a radiation or nuclear facility shall establish measurable safety goals for the different organizational levels that are in accordance with the strategies, objectives and plans of the organization.

(4) The management of the investor or operator of a radiation or nuclear facility shall ensure that the documents referred in the first paragraph of this Article are regularly reviewed for compliance with the safety goals and that actions are taken to remedy any deviations from the documents referred to the first paragraph of this Article.

Article 62

(responsibility and authority for the management system)

(1) The management of the investor or operator of a radiation or nuclear facility is responsible for the establishment, implementation, maintenance and continuous improvement of the management system in order to ensure safety and meet all regulatory requirements.

(2) The management of the investor or operator of a radiation or nuclear facility shall assign a designated individual who is responsible for coordinating the development, implementation and maintenance of management systems.

(3) The designated individual from the previous paragraph shall have a direct access to management.

(4) The investor or operator of a radiation or nuclear facility shall retain overall responsibility for the management system even when the external organization is involved in the work of developing the overall management system or a part of it.

Article 63 (resource management)

(1) The management of the investor or operator of a radiation or nuclear facility shall determine the amount of resources necessary (personnel, infrastructure, work environment, information, knowledge, suppliers and financial resources) and the required competencies to carry out the activities and to establish, implement, assess and continually improve the management system and to ensure nuclear and radiation safety.

(2) The investor or operator of a radiation or nuclear facility shall assess within the management system through self-assessment and assessments of the managemet system the adequacy of resources and the effectiveness of resource utilization of the previous paragraph.

Article 64

(personnel of radiation or nuclear facility)

(1) The management of the investor or operator of a radiation or nuclear facility shall determine the competence requirements for individuals at all levels of organization and shall provide training or take other actions to achieve the required level of competence.

(2) The management of the investor or operator of a radiation or nuclear facility shall implement the evaluation of the effectiveness of the implemented measures from the previous paragraph.

(3) The management of the investor or operator of a radiation or nuclear facility shall ensure that individuals are competent to perform their assigned work and that they understand the consequences for radiation or nuclear safety of their activities and meet the goals of the organization. Individuals shall have received appropriate education and training, and shall have acquired suitable skills, knowledge and experience to ensure their competences. Training shall ensure that individuals are aware of the relevance and importance of their activities and of how their activities contribute to safety in the achievement of the organization’s objectives.

(4) The management of the investor or operator of a radiation and nuclear facility shall by means of a systematic analysis determine the required number of personnel and their competencies to ensure radiation or nuclear safety.

(5) Management of the investor or operator of a radiation or nuclear facility shall regularly verify and document the sufficiency of the number of the personnel and their competencies for work related to radiation or nuclear safety.

(6) Management of the investor or operator of a radiation or nuclear facility shall make a ten- year staffing plan for areas that are important for radiation and nuclear safety. Staffing plan shall be updated at intervals no longer than three years.

(7) Each planned change in the number of personnel, which might be significant for radiation and nuclear safety, shall be justified and evaluated in advance and after implementation.

(8) The investor or operator of a radiation or nuclear facility shall always have in house sufficient personnel with suitable competencies to understand the design basis of a radiation or nuclear facility, as to understand the actual design and operation of the facility in all its states, to develop project tasks and acceptance criteria to outsource works relevant to the radiation or nuclear safety to contractors, to supervise the execution of such works and to evaluate them upon acceptance.

Article 65 (organizational structure)

(1) The investor or operator of a radiation or nuclear facility shall establish an organizational structure to ensure safe and reliable operation of the facility, and an appropriate response in emergencies. The organizational structure shall be documented and its efficiency regularly evaluated.

(2) The investor or operator of a radiation or nuclear facility shall clearly define and document authorities, responsibilities, hierarchical links between persons, organizations and organizational units involved in activities important to safe operation of the radiation or nuclear facility and to emergency management.

(3) Any changes of the organizational structure established to implement activities essential for the safe operation of a radiation or nuclear facility or to deal with the emergency preparedness that influence or may indirectly influence the contents of the safety- analysis report shall be controlled and managed in accordance with the requirements applying to the approval of modifications to the facility pursuant to the act governing protection against ionizing radiation and nuclear safety. Each modification to the organizational structure shall be first justified in accordance with Article 64 of these Rules.

(4) The investor or operator of a radiation or nuclear facility shall analyze, control and monitor the implementation of organizational changes from previous paragraph and thereby ensure, that modification does not affect radiation or nuclear safety. The investor or operator of a radiation or nuclear facility shall inform all interested parties about the changes and keep records of changes.

Article 66 (process management)

(1) The investor or operator of a radiation or nuclear facility shall identify, establish and manage the processes that are needed to achieve the safety goals of the organization.

(2) The management of processes and single activities shall ensure that:

1. processes are developed and documented and that necessary additional documentation is maintained;

2. an effective interaction between interfacing processes is established;

3. the process documentation is consistent with existing documents;

4. records required to demonstrate the process results are specified in the process documentation;

5. processes are monitored and the performance of the processes is reported;

6. process’ improvements are promoted;

7. processes, including any subsequent changes to them, are aligned with the goals, strategies, plans and objectives of the organization.

(3) Processes or modifications to existing processes shall be designed, established and implemented so that safety is not compromised and that requirements are fulfilled.

(4) Processes shall be implemented, assessed and continually improved. Also, feedback information shall be taken into the consideration.

(5) The sequence of processes, interfaces and interactions between the processes shall be defined so that safety is not compromised. Special attention should be given to the interfaces within the organization and with external organizations.

(6) New processes or modifications to existing processes shall be designed, verified, approved and implemented in such a way that safety is not compromised.

(7) The process shall also include aspects of security so that safety is not compromised.

(8) The documentation necessary for the implementation of the processes referred to in the first paragraph shall be controlled. Changes to this documentation shall be reviewed and recorded and shall be subject to the same level of approval as the original documentation. The use of valid reviewed documents shall be ensured.

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

(10) The management system shall provide for the recording of activities and control over such records. Throughout their period of validity, all records shall remain legible, integral, identifiable and accessible.

(11) Any activities for inspection, testing, verification and validation, their acceptance criteria and the responsibilities for carrying out such activities shall be specified. It shall be specified when and at what stages independent inspection, testing, and verification and validation are required to be conducted.

(12) For determining inspection, testing, verification and validation as well as acceptance criteria graded approach shall be taken into account.

(13) Any activity that may affect safety shall be carried out under controlled conditions, by using understandable and approved procedures, instructions, drawings or other appropriate means that are validated before the first use and periodically reviewed to ensure their adequacy and effectiveness. Personnel carrying out the activities relating to the processes shall be involved in the validation procedure.

Article 67

(supervision of subcontractors and suppliers)

(1) The investor or operator of a radiation or nuclear facility shall retain responsibility also for any works executed by subcontractors;

(2) The investor or operator of a radiation or nuclear facility shall in its management system foresee the control of processes or activities within individual processes executed by subcontractors. The investor or operator of a radiation or nuclear facility shall:

1. request that subcontractors establish, document and implement the management systems;

2. carry out assessments of subcontractors and suppliers of services and products;

3. select subcontractors and suppliers according to criteria set in advance;

4. check if suppliers and subcontractors understand the safety requirements, and if works are carried out in compliance with safety requirements;

5. assess if the subcontractor has the resources and expertise to ensure the safety and quality of products or services;

6. ensure that the products and services of subcontractors and suppliers are in accordance with safety requirements;

7. check if suppliers and subcontractors have control of their subcontractors;

8. ensure that contract requirements including safety requirements are specified;

9. through inspections, testing, verification and validation confirm prior to adoption, installation or operation that the activities or the products meet the prescribed criteria;

10. create and specify criteria for product procurement and document them. A supplier shall furnish the investor or facility operator with documented evidence of compliance with criteria, before the product is used;

11. when ordering safety-related SSCs, request the manufacturers or suppliers to provide the means for supervision by its staff or administration staff, including access to the manufacturer’s or supplier’s premises for inspection.

Article 68

(quality of instaled equpiment)

The investor or operator of a radiation or nuclear facility shall establish procedures for identifying inadequate, counterfeit, fraudulent or suspicious items and shall strengthen the activities in the supply chain to prevent the entry of such items.

Article 69 (monitoring and measurement)

(1) To ensure favorable outcomes of the organization the investor or operator of a radiation or nuclear facility shall monitor and measure the effectiveness of the management system, in order to, based on the feedback, identify weaknesses and strengths of the organization and opportunities for improvements as well as to increase the radiation and nuclear safety.

(2) The investor or operator of a radiation or nuclear facility shall define and use performance indicators in order to evaluate the effectiveness of the management system and confirm that the processes and activities of the organization are appropriate for achieving the desired results. Trends of performance indicators should be regularly assessed and analyzed. Reviews shall be made on the basis of risk with respect to potential radiation impacts arising from the particular process or activity. The investor or operator of a radiation or nuclear facility shall decide on the bases of risk assessment whether changes of the management system are needed.

Article 70

(non-conformances and corrective and preventive actions)

(1) The management of the investor or operator of a radiation or nuclear facility shall ensure that the causes of non-conformances, events and safety issues shall be determined and remedial actions shall be taken to prevent their recurrence. The corrective and preventive measures should be introduced in a timely manner to eliminate the causes of non- conformances and to prevent new or similar discrepancies. The status and effectiveness of all corrective and preventive actions shall be monitored and reported to management at an appropriate level in the organization.

(2) Products and processes that do not meet the specified requirements shall be identified, segregated, controlled, recorded and reported to an appropriate level of management within the organization. The impact of non-conformances shall be evaluated and non- conforming products or processes shall be either: accepted, reworked or corrected within a specified time period; or rejected and discarded or destroyed to prevent their inadvertent use.

Article 71 (self-assesment)

All levels of management of the investor or operator of a radiation or nuclear facility shall regularly conduct self-assessment:

- to identify and correct weaknesses with the aim of continuous improvement of the management system;

- to confirm that the management system ensures the required radiation and nuclear safety;

- to strengthen the management, safety culture and effectiveness of processes and activities.

Article 72 (independent assessment)

(1) The management of the investor or operator of a radiation or nuclear facility shall ensure that independent assessments of the management system, including audits are conducted regularly. With the independent assessment the effectiveness of the management system is measured.

(2) The purpose of the independent assessments is:

- to evaluate the effectiveness of processes in meeting and fulfilling strategies, goals, objectives and plans;

- to assess the compliance with the relevant safety standards and integration of safety requirements into the management system;

- to determine the adequacy of the implementation of work and management;

- to assess the organization’s management and safety culture;

- to monitor the quality of products and services;

- to assess the adequacy of resources in order to enable personnel to meet the requirements and to achieve their objectives in accordance with the strategies and plans;

- to identify opportunities for improvements.

(3) The management of the investor or operator of a radiation or nuclear facility shall evaluate the assessment results and take appropriate measures. The results of the evaluations and measures shall be documented. The results of the evaluations, measures and reasons for the measures shall regularly be communicated to the personnel.

(4) Plans to conduct independent assessment and self-assessment of the management system shall take into account difficulties in the implementation of the management system and the introduction of possible improvements.

(5) The operator of a radiation or nuclear facility shall set up an internal group or an individual responsible for implementation of an independent assessment. In the case of a nuclear power plant, such a group shall be set up as an organizational unit for independent assessment.

(6) An independent organizational unit or individual referred in the previous paragraph shall be vested with appropriate authorities to carry out its tasks. Organizational unit or individual shall have the possibility of direct management reporting.

(7) An individual involved in an independent assessment of the management system shall not participate in the assessment of their own activities.

Article 73 (management system review)

(1) A management system review shall be conducted by the investor or operator of a radiation or nuclear facility at planned intervals to ensure the effectiveness of the management system and its ability for achieving the objectives set for the organization.

(2) The review shall cover but shall not be limited to:

- results from all forms of assessments and evaluations;

- results delivered and objectives achieved by the organization and its processes;

- non-conformances, preventive and corrective actions;

- operating experience;

- lessons learned from the organization and other organizations and causes for the events occurring;

- opportunities for improvement;

- good practices and achievements in the profession.

(3) The review from the first paragraph of this Article shall identify whether there is a need to make changes to or improvements in policies, goals, strategies, plans, objectives and processes.

Article 74 (improvements)

(1) The management at all levels shall encourage the identification of opportunities for the improvements of the management system. Actions to improve the processes shall be selected, planned and documented.

(2) Improvement plans of investor or operator of a radiation or nuclear facility shall include plans for the provision of adequate financial, human or other resources. Actions for improvement shall be monitored through to their completion and the effectiveness of the improvement shall be assessed.

6. TRANSITIONAL AND FINAL PROVISIONS

Article 75

(requirements applying to existing facilities)

(1) Procedures undertaken prior to the entry of these Rules into force pursuant shall be completed pursuant to such previous legislative provisions.

(2) For the Krško nuclear power plant, the provision of the Item 6 under 1.9 of Annex 1 of these Rules shall enter into force on 31 December 2018.

(3) For the Krško nuclear power plant, the provision of the Item 4 and 5 of Annex 1 of these

Rules shall enter into force on 31 December 2021.

(4) Management systems already introduced at the time of the entry into force of these Rules shall be aligned with the provisions of these Rules within two years from the entry into force of these Rules.

Article 76 (revocation)

On the date of entry into force of these Rules, the Rules on Radiation and Nuclear Safety Factors (Official Gazette of the Republic of Slovenia, No. 92/09 and 9/10 corr. shall cease to be valid.

Article 77 (entry into force)

These Rules shall enter into force on the fifteenth day after their publication in the Official

Gazette of the Republic of Slovenia.

No. 007-502/2015

Ljubljana, on 18 November 2016

EVA 2015-2550-0191

Irena Majcen Minister of the Environment and Spatial Planning

Annex 1: Nuclear Power Plant Design Bases

1. ESSENTIAL DESIGN BASES

1.1. Nuclear power plant design bases

1. Risks to the population due to the operation of a nuclear power plant shall be comparable or lower than those due to the generation of electric power from other sources.

2. The design of a nuclear power plant shall limit the core damage frequency to less than

10-5 per year and the frequency of large or early (before it is possible to carry out the evacuation of the public) uncontroled release of radioactive substances from the power

plant, from all possible sources less than 10-6 per year.

3. In the case that the core damage frequency is less than 10-5 per year but more than 10-6 per year or the large uncontroled release frequency is less than 10-6 per year but more than 10-7, the investor or operator shall provide substantiated proof that any further reduction of the level of frequency is either impossible or not reasonable.

4. The design shall ensure that the core damage accidents with core melt, which would lead to early or large releases are practically eliminated. For core damage accidents, which can not be practically eliminated practical solutions shall be available, which shall ensure that only limited protective measures are needed for public (no permanent relocation, no need for evacuation immediate vicinity of the plant, limited sheltering and long-term food restrictions) and that sufficient time is available to implement these measures.

5. The design bases shall determine the necessary capabilities of the nuclear power plant to withstand all the facility states and ensure compliance with the requirements of radiation protection. The design bases shall include:

- normal operational conditions;

- conditions under anticipated operational occurrences and design basis events;

- SSC safety classification;

- substantial assumptions, and

- selected analytical methods.

6. The design bases shall take into account the internal and external initiating events characteristic for the power plant and and their credible and still plausible combinations, which shall be treated in accordance with the site conditions. External postulated initiating events in addition to natural external events referred to in item 5 of this Annex shall also include human made external hazards including airplane crash and other nearby transportation, industrial activities and site area conditions that might lead to fires, explosions or other threats to the safety of the nuclear power plant.

7. Without prejudice to the provision in item 2 under 1.1 of this Annex, the design of the Krško nuclear power plant shall limit the core damage frequency during power operation to less than 10-4 per year and the large release frequency during power operation to less than

510-6 per year.

8. For Krško nuclear power plant the provisions referred to in item 4 under 1.1 of this Annex shall be used as a reference for the timely implementation of reasonably practicable safety improvements within the framework of periodic safety reviews.

1.2. Safety functions

1. A nuclear power plant shall during normal operation (including start-up, power operation, shutting down, shutdown, maintenance, testing and refuelling), anticipated operational occurrences and design-basis events achieve the following safety functions:

- control of core reactivity;

- removal of heat from the reactor core and from the spent fuel, and

- containment of radioactive material and prevention of their uncontrolled spread into the environment.

2. No failure of any system necessary for the normal operation of the nuclear power plant may affect safety functions.

1.3. Safety analyses

In addition to the conditions referred to in Article 16 of these Rules, the safety analyses shall take into account:

- for the purposes of achieving and maintaining the state of safe shutdown, only those

SSCs classified according to item 2.1 of this annex;

- as additional aggravating circumstances, in all the facility states and under all design- basis accidents, jamming of the control rod with the highest reactivity value in the fully withdrawn position, and the total blackout of the plant;

- for nuclear power plants, derogations from the third paragraph of Article 16 are only allowed in safety analyses of accidents exceeding design basis events.

1.4. Technical acceptance

The design shall specify, as a minimum:

1. radiological and other technical acceptance criteria for the classification of postulated initiating events into different operating conditions of the nuclear power plant (usually: facility states, design-basis events, additional postulated failures, severe accidents). Postulated initiating events with higher probability of occurrence shall have negligible or no radiological consequences, while those events that may cause severe accidents shall have extremely low probabilities of occurrence;

2. fuel-cladding protection criteria, which include:

- fuel temperature;

- departure from nucleate boiling, except for gas-cooled reactors;

- cladding temperature;

- fuel-element integrity including limitation of the release of fission products in operation, which shall not increase notably even in abnormal operating states;

- the maximum allowable fuel damage as a result of a design-basis event, provided that the fuel elements remain in their positions and the damage does not obstruct efficient heat removal following the accident;

3. criteria for primary coolant system pressure-boundary protection, which include:

- the maximum coolant temperature;

- the maximum coolant pressure;

- the number of transients that impose thermal or pressure loading on the primary coolant system, rates of changes of temperature and pressure during such transients, the minimum and maximum temperatures and pressures reached, etc.;

- material stresses;

4. for a pressurised water reactor power plant, criteria for protection of the secondary coolant system, which shall include parameters analogous to those set out in the previous item;

5. containment-protection criteria, which include:

- temperatures in the containment;

- pressures in the containment;

- leakage.

1.5. Reactor trip systems

To ensure safe shutdown of the reactor, at least two diverse systems shall be provided. At least one of these systems shall be capable of autonomously bringing, within four seconds, the reactor into subcritical condition with an appropriate reactivity margin, from any state of the facility and in a design-basis event. This shall be achieved also under the assumption of a single failure.

Sub-criticality shall be ensured and sustained:

- in the reactor after planned reactor shutdown during normal operation and after anticipated operational occurrences, as long as needed;

- in the reactor, after a transient period (if any) following a design basis accident;

- for fuel storage during normal operation, anticipated operational occurrences, and design basis accidents.

1.6. Residual-heat removal

Residual-heat removal from the reactor core shall be provided following its shutdown from any facility state, as well as for fuel storage during and after anticipated operational occurrences and design basis accidents, even under the assumptions of a single failure and the loss of off-site power.

1.7. Containment

1. A nuclear power plant shall be fitted with a containment that shall ensure any releases of radioactive substances during a design-basis event into the environment remain below statutory limits. The containment system shall comprise:

- leaktight structures enclosing all the essential components of the primary reactor- coolant system;

- containment temperature- and pressure-monitoring systems;

- devices for isolation, containment and removal of radionuclides, hydrogen, oxygen and other substances that may be released into the containment atmosphere.

2. Any line that penetrates the containment and forms a part of the reactor coolant-pressure boundary or is in direct contact with the containment atmosphere shall be isolated automatically and reliably upon any event that leads to a design-basis accident. Each such line shall be fitted with at least two appropriate isolation valves in series. Isolation valves shall be located as close to the containment as practicable.

3. Any line that penetrates the containment and does not form a part of the reactor coolant- pressure boundary and is not in direct contact with the containment atmosphere shall be fitted with at least one appropriate isolation valve. The isolation valve shall be located outside the containment, as close to the containment as practicable.

4. The containment building shall retain its functionality even in an event of a direct crash of a large commercial aircraft; in the case of the Krško nuclear power plant, all reasonable measures to mitigate the consequences of a crash of a large commercial aircraft shall be taken.

1.8. Instrumentation and controls

1. Instrumentation shall allow measurement of all the main variables of the nuclear power plant that may affect the fission process, the reactor-core integrity, the reactor coolant system the containment and the state of the spent fuel storage. Instrumentation shall also allow the acquisition of all the plant information necessary for safe and reliable operation and for determining the status of the plant in design basis accidents. All safety-related parameters shall be automatically recorded and archived.

2. Instrumentation and controls shall be qualified for application under all ambient conditions for which they are intended and shall be mutually electromagnetically compatible.

3. Any unauthorised access or ingress to the instrumentation and control systems shall be prevented by appropriate physical, technical and administrative measures.

4. Instrumentation and control systems shall be designed and implemented in a way that prevents any impact of a failure of faulty data transfer on the correct operation of safety systems.

5. The design, installation and testing of software and hardware of computer-supported systems relevant to safety shall apply appropriate standards. Software for digital instrumentation and control shall be verified, validated and tested. Due to the integral nature of computer-supported systems, an additional degree of conservatism is necessary in their analyses.

1.9. Control room

1. The main control room shall allow safe operation and monitoring of the nuclear power plant in normal operation, abnormal operation and design basis accidents. All actions necessary to keep the plant in a safe condition and the restoring of safe condition upon an anticipated operational occurrence or a design-basis event must be achievable from the main control room.

2. The design of a main control room shall follow the principles of ergonomics. All needed information from the instrumentation shall be presented in such a way, that enables timely assessment of the condition of facility and its safety functions during accidents.

3. The main control room shall provide appropriate visual and acoustic indications of the facility and process states that deviate from normal conditions and may affect safety.

4. The operator shall be provided with appropriate information necessary to manage the consequences of automatic actions.

5. The design shall envisage those events within and outside the nuclear power plant that might pose threats to the activities in the control room and minimise their potential impacts.

6. In an event of inaccessibility of the main control room, a supplementary control room shall be provided, physically, electrically and functionally separated from the main control room. The supplementary control room shall be fitted with sufficient monitoring and control equipment to allow safe shutdown of the reactor, maintenance of the safe shutdown condition, removal of residual heat from the reactor and spent fuel storage and monitoring of essential plant parameters, including the conditions in the spent fuel storage.

1.10. Protection system

1. The protection system shall be designed for a high level of reliability. The design shall follow the principles of redundancy and independence. The protection system shall meet, as a minimum, the following conditions:

- no single failure shall cause a failure of the protection system;

- no failure of any component or channel shall cause the loss of the minimum required redundancy.

2. The design shall allow the testing of all functions of the protection system (from measurement sensors and input signals to final actuators) in operation.

3. The risk that an operator’s action may compromise the effectiveness of the protection system in any state of the facility shall be minimised. The protection system shall not prevent or override proper actions by the operator during a design-basis event.

4. The computer-supported system constituting a part of the protection system shall meet the following requirements:

- its hardware and software shall meet the most stringent quality specifications, they shall ensure optimal operational performance and the highest reliability;

- the overall development process, including surveillance, testing and implementation of modifications of the design, shall be systematically reviewed and documented;

- the computer-supported system shall be submitted to an independent expert assessment to verify the confidence level of its reliability;

- if a high level of confidence in the system cannot be achieved, an alternative method of achieving all safety measures expected from the protection system shall be implemented.

1.11. Emergency power supply

The source of the emergency power supply shall be capable of supplying the necessary power to safety-related systems and components in all facility states and during a design-basis event. This shall also be accomplished under the assumptions of a single failure and a total blackout.

1.12. Fuel and radioactive-waste handling systems

1. The design shall provide for storage of spent fuel and procedures involved in the transport of fuel elements from the facility. Cooling of the irradiated fuel shall be provided. The option of removing the entire core from the reactor shall be available at all times.

2. The design shall provide for storage of irradiated fuel for extended time periods. The design of the fuel and radioactive-waste handling systems shall ensure:

- the prevention of inadvertent criticality through physical means, e.g. through appropriate geometry or permanent neutron absorbers;

- minimised risk of fuel loss or damage, the impact of falling heavy objects and excessive loads on the fuel elements;

- storage of damaged fuel elements, control of chemical condition and coolant activity, and provisions for periodic inspection and testing of the fuel;

- adequate physical protection against theft or sabotage and verification of the identity of individual fuel elements.

3. The design and operation of a nuclear power plant shall minimise the volume of the generated radioactive waste. The radioactive-waste handling systems shall minimise releases and bylaw limits through surveillance and monitoring. For solid and liquid radioactive waste, handling systems and on-site storage shall be provided.

4. To minimise the exposure of personnel to radiation and releases to the environment, the design shall provide for systems to protect against radionuclides and their decay. Means shall be provided to measure radioactive releases into the environment, such as release sampling and monitoring.

5. The design shall provide for the means of radioactive-waste handling, collection, processing, storage and transport from the site. The liquid radioactive-waste handling system shall provide for leakage detection and capture of released substances.

1.13. Physical and functional separation and isolation

1. Systems shall be physically separated in order to increase confidence in their independence, especially in relation to common cause failures. Physical separation comprises:

- a geometrical separation (distance or position in the area);

- separation with barrier;

- separation with combination of both the above mentioned separation;

2. Functional separation shall prevent harmful mutual influences of equipment and components of redundant or interlinked systems due to their normal or abnormal operation, or due to failure of any of these components.

3. Special attention shall be paid to the independence of the SSC, which perform the same safety function on different levels of defense in depth. The SSCs, which ensure safety functions at different levels of defense in depth shall carry out their functions independently of the operation or failure of other SSCs needed at other levels of defense in depth.

4. The independence between levels of defense in depth measures shall primariliy be based on the principle of diversity decribed in the fifth paragraph of Article 3 of these Rules.

5. For sites with multiple radiation or nuclear facilities, the appropriate independent between them shall be ensured. Mutual support by sharring support systems can be considered, as far as this is not detrimental for safety.

2. SSC SAFETY CLASSIFICATION AND CATEGORISATION

2.1. SSC safety classification

1. Each SSC shall be classified into a safety class according to its importance to safety.

SSCs shall be designed, manufactured and maintained so as to ensure reliability and quality adequate for the importance of the SSC for safety.

2. The classification of SSCs into safety classes according to their importance for safety shall be based on nuclear safety analyses carried out employing deterministic methods and supplemented with probabilistic methods and engineering judgement as appropriate.

3. For each safety class, the safety classification shall specify:

- regulations and standards to be applied in design, manufacture, installation and inspection;

- requirements for emergency power supply and SSC compatibility with anticipated environmental conditions;

- availability/unavailability of systems necessary to achieve a safety function upon initiating events postulated in safety analyses employing deterministic methods;

- quality-assurance requirements.

2.2. SSC categories

1. SSCs may be categorised.

2. SSC categorisation means the classification of SSCs into four safety categories according to their relevance to risks established in a probabilistic safety assessment:

- safety category one (SC-1) includes SSCs that are classified as safety related and are intended to achieve a function critical for safety;

- safety category two (SC-2) includes SSCs that are not classified as safety related and are intended to achieve a function critical for safety;

- safety category three (SC-3) includes SSCs that are classified as safety related and are not intended to achieve a function critical for safety;

- safety category four (SC-4) includes SSCs that are not classified as safety related and are not intended to achieve a function critical for safety.

3. A function critical for safety as referred to in the previous paragraph shall mean a function the loss of which may significantly compromise defence-in-depth or safety margins i.e. that may significantly increase risks.

4. The categorisation shall:

- comply with the results and findings of probabilistic safety assessments (PSA) of the plant. PSAs shall be carried out in a high-quality manner, independently verified and appropriately detailed and shall cover, as a minimum, severe accidents caused by internal events during power operation;

- determine the importance of SSCs for the achievement of functions critical for safety through an integral and systematic approach covering initiating events, SSC and plant states, including those not covered by the PSA. The process shall consider the actual condition of the plant, and internal and external operating experiences. Functions critical to safety shall include functions required under design-basis events, accidents and severe accidents;

- comply with the defence-in-depth principle;

- include analyses to guarantee an appropriate level of confidence in the existence of adequate safety margins for all the SC-3 category SSCs and in the negligible contribution of the categorisation to an increase of risk of core meltdown and large releases;

- cover the entire system and structures, not only selected system components and structures.

5. Categorisation shall be undertaken by a group of experts familiar with the plant and proficient, as a minimum, in PSA, other types of safety analyses, plant operation, specification of design bases and system design.

2.3. Application of categorisation

1. An assessment of SSCs in categories SC-1 or SC-2 shall confirm their capability to achieve their functions in accordance with their categories.

2. SSCs categorised in SC-3 shall be capable of achieving their functions with adequate reliability throughout the service life in all circumstances specified in the design bases.

3. SSCs in categories SC-3 or SC-4 are not subject to the requirements laid down in accordance with their safety classification in item 2.1 of these Annex.

2.4. Requirements for SSCs

The failure of a SSC of a lower safety class referred to in item 2.1 of this Annex shall not cause a failure of a SSC of a higher safety class. The same requirement applies to auxiliary systems that support safety-related equipment.

2.5. SSC qualification programme

1. The operator of a nuclear power plant shall adopt and implement a qualification programme for safety-related SSCs.

2. The qualification programme referred to in the previous paragraph shall be applied by the operator to confirm the capability of SSCs to achieve their design functions over the entire design service life.

3. The SSC-qualification programme shall include collection, documentation and maintenance of information to confirm the capability of SSCs to achieve their design functions over the entire design service life.

4. The qualification programme referred to in the previous paragraphs shall consider operating conditions such as vibration, temperature, pressure, water-jet impacts, electromagnetic disturbances, irradiation, moisture, earthquake and combinations thereof. Operating conditions shall cover normal operating conditions over the entire design service life, conditions of abnormal operation and the conditions during accidents for those SSCs that are necessary for monitoring or controlling emergency situation.

5. The qualification program shall ensure, that SSCs important to safety, in the case of changes, preserve their qualification.

2.6. Contents of an application for SSC categorisation

An operator of a nuclear power plant desiring to apply SSC categorisation shall enclose the following with the application for categorisation:

1. a description of the implementation of the SSC categorisation;

2. a description of the measures implemented to ensure adequate quality and detail of the analyses, and assessments of internal and external events during power operation and at plant shutdown, including analyses of severe accidents, for the purposes of categorisation;

3. results of the review of probabilistic safety analyses concerning their impacts on categorisation;

4. a description and appropriate bases for the acceptance of assessments to guarantee an appropriate level of confidence in the existence of adequate safety margins for all the SC-

3 category SSCs and in the negligible contribution of the categorisation to an increase of

risk of core meltdown and large releases. Such assessments shall cover influences with regard to sensitivity to interaction due to common causes and any influences of known degradation mechanisms on active and passive SSC functions. They shall cover internal and external events, power operation and shutdown states.

3. PROTECTION AGAINST INTERNAL FIRE

3.1. Fire-protection objectives

Fire protection shall observe the defence-in-depth principle, to ensure:

- measures to prevent occurrence of fire;

- fast detection, containment and suppression of fires;

- prevention of fire spreading and consequences in any area where they might compromise nuclear power plant safety, or of fire reaching such areas.

3.2. Fire-protection design bases

1. Safety related SSCs shall be designed and installed as to:

- minimise the risks of occurrence of fire and its consequences;

- ensure the capability of plant shutdown;

- ensure the capability of residual-heat removal;

- limit the spreading of radioactive substances;

- ensure control over the situation in the nuclear power plant during and following a fire.

2. The buildings containing safety-related SSCs shall be protected against fire in compliance with the findings of the analysis of fire risks referred to item 3.4. of this Annex.

3.3. Building fire safety

1. Buildings containing safety-related equipment or radioactive substances and buildings, in which a fire might compromise the nuclear power plant's safety, shall be designed so as to maximise their fire safety, and shall be divided into fire compartments as appropriate.

2. Fire compartments referred to in the previous paragraph shall prevent fire loads to safety- related equipment and separate redundant or diverse trains of a safety system.

3. In the case that the division into fire compartments referred to in the previous paragraph is not practicable or is not reasonable, division into fire cells shall be applied and balance

achieved between passive and active safety; this shall be confirmed by the fire-risk analysis referred to in Item 3.4 of this Annex.

4. Buildings containing radioactive substances and in which fire might result in radioactive releases shall be designed so as to minimise such releases in the event of fire.

5. The design shall provide for fire routes for all personnel involved in fire containment and evacuation routes for all plant personnel.

3.4. Fire-risk analysis

1. Fire protection of a nuclear power plant shall be demonstrated by a fire-risk analysis, which shall be updated upon any major modification or at intervals of a maximum of two years.

2. The fire-risk analysis shall confirm:

- achievement of all the fire-protection objectives;

- compliance with fire-protection design principles;

- proper design of fire-protection measures; and

- proper implementation of all the necessary administrative measures.

3. The deterministic part of the fire-risk analysis shall include, as a minimum:

- individual fires as well as their spreading in all locations where combustible materials are kept on a temporary or permanent basis, in all the facility states, including the shutdown states;

- possible combinations of fires and other postulated initiating events that may occur independently of the fire.

4. The fire-risk analysis shall demonstrate the method of consideration of any consequences of the fire or its suppression.

5. A fire-risk analysis shall further include a probabilistic safety analysis of fire risks; this analysis shall constitute part of the level-one probabilistic safety analyses. This analysis shall verify the adequacy of the fire-protection arrangement and measures, and estimate the risks posed by fires.

3.5. Fire-protection systems

The design shall ensure the compliance of the fire-protection system with the following requirements:

1. Each fire compartment or fire cell shall be fitted with fire detectors and alarm devices; the control room shall be fitted with alarm systems to indicate the presence and location of a fire. The system shall be supplied from an uninterruptible emergency power supply through fire-resistant cables.

2. A nuclear power plant shall be fitted with stationary and portable, automatic or manual fire extinguishers. Their design shall prevent their interference, through action or failure, with the achievement of the functions of safety-related SSCs.

3. The outdoor fire-hydrant distribution loop and fire taps within the buildings shall ensure adequate coverage of safety-related areas of the facility. Coverage shall be substantiated in the fire-risk analysis.

4. Ventilation system shall be implemented so as to maintain separation between fire compartments in an event of fire.

5. Outdoor sections of ventilation systems shall have fire properties equivalent to sections within fire compartments, as an alternative, outdoor sections shall be provided with fire dampers to ensure their isolation.

3.6. Fire-safety surveillance and maintenance

1. For fire-prevention purposes, the facility operator shall implement appropriate procedures to monitor and minimise the volumes of combustible materials and to minimise the number of potential sources of fires that might affect safety-related SSCs.

2. The facility operator shall adopt and implement procedures to ensure the feasibility of fire- protection measures.

3. The facility operator shall implement procedures for inspection, maintenance and testing of fire barriers, detection systems and suppression systems.

3.7. Fire-protection organisational arrangements

1. The facility operator shall implement measures to monitor and ensure fire safety in compliance with the findings of the fire-risk analysis. Such arrangements shall include nominating persons to be responsible for or have duties with respect to fire protection and shall set out the requirements for control of all activities that can have impact on fire safety.

2. The facility operator shall devise plans of action in the event of fire and update them at intervals of a maximum of two years, and provide for regular training of their implementation. These plans shall provide for measures in all the areas where a fire might affect safety-related equipment of the nuclear power plant or the protection of radioactive substances.

3. Written procedures for emergencies shall clearly specify responsibilities and actions of personnel in any fire in the nuclear power plant and shall be regularly updated at intervals of a maximum of two years.

4. If the system envisages the engagement of outside organisations in fire suppression, personnel of such organisations shall be informed of relevant risks in the nuclear power plant. Under such an arrangement, collaboration with outside organisations shall be appropriately organised and managed and covered by the fire-action plan.

5. If the system envisages the engagement of nuclear power plant personnel in fire suppression, the fire-protection organisational arrangements, the minimum number of personnel involved, and requirements for equipment, qualifications and training shall be documented. The adequacy of all the above arrangements shall be confirmed by a person qualified in accordance with according to the law governing fire protection.

4. DESIGN EXTENSION CONDITIONS

4.1. Selection of design extension conditions

1. A set of design extension conditions shall be derived and justified as representative, based on a combination of deterministic and probabilistic assessments as well as engineering judgement.

2. The selection process for design extension conditions A shall start by considering those events and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover:

- events occurring during the defined operational states of the plant;

- events resulting from internal or external hazards;

- common cause failures.

Where applicable, all reactors and spent fuel storages on the site have to be taken into account. Events potentially affecting all units on the site, potential interactions between units as well as interactions with other sites in the vicinity shall be covered.

3. The design extension conditions category B events shall be postulated and justified to cover situations, where the capability of the plant to prevent severe fuel damage is exceeded or where measures provided are assumed not to function as intended, leading to severe fuel damage (e.g. system failures, man induced events, etc.).

4.2. Safety analysis of design extension conditions

The design extension conditions analysis shall:

1. rely on methods, assumptions or arguments which are justified, and should not be unduly conservative;

2. be auditable, paying particular attention where expert opinion is utilized, and take into account uncertainties and their impact;

3. identify reasonably practicable provisions to prevent severe fuel damage (design extension conditions A) and mitigate severe accidents (design extension conditions B);

4. evaluate potential on-site and off-site radiological consequences resulting from the design extension conditions (given successful accident management measures);

5. consider plant layout and location, equipment capabilities, conditions associated with the selected scenarios and feasibility of foreseen accident management actions;

6. demonstrate, where applicable, sufficient margins to avoid “cliff-edge effects” that would result in unacceptable consequences, i.e. for design extension conditions A severe fuel damage and for design extension conditions B a large or early radioactive release;

7. reflect insights from probabilistic safety analysis;

8. take into account severe accident phenomena, where relevant;

9. define an end state, which should where possible be a safe state, and, when applicable, associated mission times for SSCs.

4.3. Ensuring safety functions in design extension conditions

General

1. In design extension conditions A, it is the objective that the plant shall be able to fulfil, the fundamental safety functions:

- control of reactivity, if it is lost, it shall be reestablished after a transient period;

- removal of heat from the reactor core and from the spent fuel; and

- confinement of radioactive material.

In design extension conditions B, it is the objective that the plant shall be able to fulfil confinement of radioactive material. To this end removal of heat from the damaged fuel shall be established.

2. For the fulfilment (or reestablishment) of the fundamental safety functions in design extension conditions A and design extension conditions B, the use of mobile equipment on-site can be taken into account, as well as support from off-site, with due consideration for the time required for it to be available.

3. It shall be demonstrated that SSCs including mobile equipment and their connecting points, if applicable for the prevention of severe fuel damage or mitigation of consequences in design extension conditions have the capacity and capability and are adequately qualified to perform their relevant functions for the appropriate period of time.

4. If accident management relies on the use of mobile equipment, permanent connecting points, accessible (from a physical and radiological point of view) under design extension

conditions, shall be installed to enable the use of this equipment. The mobile equipment, and the connecting points and lines shall be maintained, inspected and tested.

5. A systematic process shall be used to review all units relying on common services and supplies (if any), for ensuring that common resources (of personnel, equipment and materials) expected to be used in accident conditions are still effective and sufficient for each unit at all times. In particular, if support between units at one site is considered in design extension conditions, it shall be demonstrated that it is not detrimental to the safety of any unit.

6. The NPP site shall be autonomous regarding supplies supporting safety functions (e.g. fuel supplies) for a period of time until it can be demonstrated with confidence that adequate supplies can be established from off site.

Long-term sub-criticality

7. In design extension conditions, subcriticality of the reactor core shall be ensured in the long term and in the fuel storage at any time. Subcriticality might not be guaranteed during core degradation and later on during some time in a fraction of the corium.

Heat removal functions

8. Nuclear power plant shall have sufficient independent and diverse means (including necessary power supplies available) to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.

Confinement functions

9. Isolation of the containment shall be possible in design extension conditions. For those shutdown states where this cannot be achieved in due time, severe core damage shall be prevented. If an event leads to bypass of the containment, severe core damage shall be prevented.

10. Pressure and temperature in the containment shall be managed.

11. The threats due to combustible gases shall be managed.

12. The containment shall be protected from overpressure. If venting is to be used for managing the containment pressure, adequate filtration shall be provided.

13. High pressure core melt scenarios shall be prevented.

14. Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

15. In design extension conditions A, radioactive releases shall be minimised as far as reasonably practicable.

16. In design extension conditions B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

- allow sufficient time for protective actions (if any) in the vicinity of the plant; and

- avoid contamination of large areas in the long term.

Instrumentation and control for the management of design extension conditions

17. Adequately qualified instrumentations shall be available for design extension conditions for determining the status of plant including spent fuel storage and safety functions as far as required for making decisions on-site as well as off-site in case of design extension conditions B.

18. There shall be an operational and habitable control room (or another suitably equipped location) available during design extension conditions in order to manage design externsion conditions. The control room shall provide long-term habitability of operators even in case of severe accidents.

Emergency power

19. Adequate power supplies during design extension conditions shall be ensured considering the necessary actions and the timeframes defined in the design extension conditions analysis, taking into account external hazards.

20. Batteries shall have adequate capacity to provide the necessary DC power until recharging can be established or other means are in place.

21. Emergency power supply shall also be ensured for the purpose of communication during the accident both on-site, as well as with emergency personnel and organizations off-site.

4.4. Review of the design extension conditions

The design extension conditions shall regularly, and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the selection of design extension conditions is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

5. NATURAL HAZARDS

5.1. Objective

1. Natural hazards shall be considered an integral part of the safety demonstration of the plant, including spent fuel storage. Threats from natural hazards shall be removed or minimised as far as reasonably practicable for all operational plant states. The safety demonstration in relation to natural hazards shall include assessments of the design basis and design extension conditions with the aim to identify needs and opportunities for improvement.

5.2. Identification of natural hazards

1. All natural hazards that might affect the site shall be identified, including any related hazards (e.g. earthquake and floods). Justification shall be provided that the compiled list of natural hazards is complete and relevant to the site.

2. Natural hazards shall include:

- geological hazards;

- seismotectonic hazards;

- meteorological hazards;

- hydrological hazards;

- biological phenomena;

- forest fire.

5.3. Site specific natural hazard screening and assessment

1. Natural hazards identified as potentially affecting the site can be screened out on the basis of being incapable of posing a physical threat or being extremely unlikely with with the exceedance frequency less than 10-7 per year. Care shall be taken not to exclude hazards which in combination with other hazards (other natural hazards, internal hazards or human

induced hazards) have the potential to pose a threat to the facility. Consequential hazards and causally linked hazards shall be considered, as well as random combinations of relatively frequent hazards. The screening process shall be based on conservative assumptions. The arguments in support of the screening process shall be justified.

2. For all natural hazards that have not been screened out, hazard assessments shall be performed using deterministic and, as far as practicable, probabilistic methods taking into account the current state of science and technology. This shall take into account all relevant available data, and produce a relationship between the hazards severity (e.g. magnitude and duration) and exceedance frequency, where practicable. The maximum credible hazard severity shall be determined where this is practicable.

3. The following shall apply to hazard assessments:

- the hazard assessment shall be based on all relevant site and regional data. Par- ticular attention shall be given to extending the data available to include events beyond recorded and historical data;

- special consideration shall be given to hazards whose severity changes during the expected lifetime of the plant;

- the methods and assumptions used shall be justified. Uncertainties affecting the results of the hazard assessments shall be evaluated.

5.4. Definition of the design basis events

1. Design basis events shall be defined based on the site specific hazard assessment. These design basis events are individual natural hazards or combinations of hazards. The design basis may either be the original design basis of the plant, when it was commissioned, or a reviewed design basis for example following a periodic safety review.

2. The exceedance frequencies of design basis events shall be low enough to ensure a high degree of protection with respect to natural hazards. A common target value of frequency, not higher than 10-4 per annum, shall be used for each design basis event. Where it is not possible to calculate these probabilities with an acceptable degree of certainty, an event shall be chosen and justified to reach an equivalent level of safety (expert assessment of the sensitivity analysis, the assessment of the worst possible events and consequences, etc). For the specific case of seismic loading, as a minimum, a horizontal peak ground ac- celeration value of 0.1g (where ‘g’ is the acceleration due to gravity) shall be applied, even if its exceedance frequency would be below 10-4 per annum.

3. The design basis events shall be compared to relevant historical data to verify that historical extreme events are enveloped by the design basis with a sufficient margin.

4. Design basis parameters shall be defined for each design basis event taking due consideration of the results of the hazard assessments. The design basis parameter values shall be developed on a conservative basis.

5.5. Protection against design basis events

1. Protection shall be provided for design basis events. A protection concept shall include design bases event and design extension conditions and also emergency operation procedures and severe accident management guideline.

2. The protection concept shall be of sufficient reliability that the fundamental safety functions are conservatively ensured for any direct and credible indirect effects of the design basis event.

3. The protection concept shall:

a) apply reasonable conservatism providing safety margins in the design;

b) rely primarily on passive measures as far as reasonable practicable;

c) ensure that measures to cope with a design basis accident remain effective during and following a design basis event;

d) take into account the predictability and development of the event over time;

e) ensure that procedures and means are available to verify the plant condition during and following design basis events;

f) consider that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;

g) ensure that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

h) not adversely affect the protection against other design basis events (not originating from natural hazards).

4. For design basis events, SSCs identified as part of the protection concept with respect to natural hazards shall be considered as important to safety.

5. Monitoring and alert processes shall be available to support the protection concept. Where appropriate, thresholds (intervention values) shall be defined to facilitate the timely initiation of protection measures. In addition, thresholds shall be identified to allow the execution of pre-planned post-event actions (e.g. inspections).

6. During long-lasting natural events, arrangements for the replacement of personnel and supplies shall be available.

7. For Krško nuclear power plant the requirements under the first item under 5.5 of this Annex, related to seismic safety, can be met by determining its actual seismic capability and demonstrating its protection against seismic hazards in accordance with the requirements of item 2 under 5.4 of this Annex. In a similar way, Krško nuclear power plant can satisfy the requirements associated with extreme external temperatures.

5.6. Considerations for events more severe than the design basis events

1. Events that are more severe than the design basis events shall be identified as part of design extension conditions analysis. Their selection shall be justified. Further detailed analysis of an event will not be necessary, if it is shown that its occurrence can be considered with a high degree of confidence to be extremely unlikely.

2. To support identification of events and assessment of their effects, the hazards severity as a function of exceedance frequency or other parameters related to the event shall be developed, when practicable.

3. When assessing the effects of natural hazards included in the design extension conditions analysis, and identifying reasonably practicable improvements related to such events, analysis shall, as far as practicable, include:

- demonstration of sufficient margins to avoid cases where small change of a parameter could cause extensive and unacceptable consequences, such as loss of a fundamental safety function;

- identification and assessment of the most resilient means for ensuring the fundamental safety functions;

- consideration that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;

- demonstration that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

- on-site verification (typically by walk-down methods).

Annex 2: Research Reactor Design Bases

1. General

1. The design of a research reactor (hereinafter: reactor) shall cover, along with the reactor, also associated devices that may affect safety. Furthermore, it shall consider any influences that the reactor may have on such devices.

2. The design of safety systems shall take into account the operational modes of the reactor and its stability at different operating powers. Operational modes include e.g. operation on user demand (differing from continuous operation), operation at different reactor powers, in different core layouts, with different fuel types, etc.

3. In special cases, such as:

- research reactors with thermal power exceeding ten megawatts;

- fast reactors;

- reactors fitted with experimental rigs, such as high-pressure or high-temperature loops;

- cold neutron sources;

- hot neutron sources,

the design bases for a nuclear power plant, laid down in Annex 1 to these Rules, and extra safety measures shall be applied by analogy.

2. Graded approach

In ensuring the safety of research reactors, a graded approach may be applied. A decision to waive any requirements for a research reactor according to such an approach may be based on the following factors:

- reactor power;

- isotopic composition of releases in an event of accident;

- quantity and enrichment of nuclear substances;

- spent fuel elements, high-pressure systems, heating systems and storage of combustible substances that may affect reactor safety;

- type of fuel elements;

- type and mass of moderator, reflector and coolant;

- reactivity value that may be inserted and the rate of reactivity insertion, reactivity control, passive and active safety features;

- quality of the containment building or other means of limiting releases into the environment;

- utilisation of the reactor (experimental rigs, tests and reactor-physics experiments);

- location;

- proximity of population groups.

3. SSC safety classification

The safety classification of SSCs of research reactors is subject to the same requirements as the safety classification of SSCs in nuclear power plants, as laid down in Item 2.1 of Annex 1 to these Rules.

4. Protection against internal fire

The design of fire protection shall comply, by analogy, with the requirements for nuclear power plants, laid down in Item 3 of Annex 1 to these Rules.

5. Operating limits, protection and safety systems

1. Operating limits of parameters shall be specified for all operational modes and design- basis events. By comparison of event scenarios, the most unfavourable parameter values shall be established and these values, with appropriate extra margins, shall be taken into account in the design of individual SSCs, including experimental rigs.

2. Facility states shall be identified. The design shall consider requirements for the intended utilisation of the reactor and for its power stability to ensure safe operation or power reduction without the need to actuate safety systems. These requirements and limits shall be applied as the basis for the determination of operating limits and conditions.

3. The reactor design shall provide, for the postulated initiating events, the means of automatic actuation of safety systems, and providing the reactor operator with actions to ensure the long-term stable state of the reactor or to limit any release of radioactive substances. As far as practicable, the design shall minimise the need for operator action, in particular during and following design-basis events. Safety systems shall be resistant to extreme loads and environmental conditions during events or accidents.

4. Safety systems shall be designed primarily to limit and mitigate consequences of anticipated operational occurrence or design-basis events. The safety systems shall be established by a safety analysis, which shall demonstrate the capability of the safety systems to achieve their functions. Furthermore, support systems shall be provided to ensure proper operation of the safety systems. Design bases shall specify the modes of safety-system operation, including the extent of automatic operation, and the necessary conditions in the case of a need for manual control of their operation. The following shall be taken into account:

- component reliability, system interdependence, redundancy, fail-safe features, diversity and physical separation of redundant systems;

- use of materials resistant to design-basis event conditions;

- options for the inspection, testing and maintenance and functionality and readiness verification of the safety system, its reliability and effectiveness.

5. The highest allowable levels of non-availability of safety systems and components shall be specified to ensure the required reliability of safety functions.

6. In the case of extended shutdown of a research reactor for modifications or preparations for decommissioning, provisions shall be made for the maintenance of conditions for nuclear fuel, coolant or moderator, for inspections, testing and maintenance of important SSCs and for physical protection.

7. All safety-related SSCs shall be designed with a safety margin to account for the effects of ageing and potential degradation due to ageing. All facility states, maintenance and shutdown states shall be considered. The design shall allow monitoring, testing, sampling and inspections to detect, assess, prevent and limit the effects of ageing.

6. Control room

1. The control room design shall meet ergonomic criteria. Appropriate visual and acoustic indications of safety-related parameters shall be provided. The design shall minimise the need for operator action.

2. If required, a supplementary functionally separated command room shall be provided for personnel to use in an accident, which shall provide an indication of reactor data and radiological conditions within the facility and in the environment.

7. Research reactor utilisation and modification options

1. The design shall consider the different modes of a research reactor operation in response to varying experimental demands. A system of effective facility configuration control shall be provided. Special emphasis shall be placed on the experimental rig.

2. Any modifications to the experiments or to the reactor that are relevant to safety shall be designed according to the same principles that apply to the reactor. Standards and solutions applied shall be comparable to solutions for the reactor as regards applied materials, structural integrity and suitability for radiological protection. Account shall be taken of the contents of radioactive substances in and energy releases from the experimental rig. In the case that the experimental rig protrudes into the area of the reactor, provisions shall be made for the maintenance of reactor integrity and protection. Any protection and safety systems of the experimental rig shall protect both the rig and the reactor.

8. Reactor core and reactivity control systems

1. Regarding the fuel elements, reflector and other core components the design shall consider neutron, thermohydraulic, mechanical, material, chemical and irradiation aspects. Analyses, supported by experimental data, shall demonstrate the acceptability of irradiation conditions and limitations, and prevent fuel swelling or deformation. Long-term handling of irradiated fuel shall also be considered.

2. Regarding the reactor-core (fuel elements, reflectors, cooling-channel geometry, irradiation devices and structural components) the design shall consider all possible core compositions. It shall ensure containment of parameters within prescribed ranges in all operational states and prevent excessive fuel damage in a design-basis event.

3. Regarding the core the design shall allow reactor shutdown, cooling and achieving subcriticality with an appropriate safety margin in all facility states.

4. Regarding the reactivity-control device the design shall take into account wear and irradiation effects, such as burnout, variations of physical properties and generation of gases.

5. The maximum reactivity value inserted shall be determined either with the reactivity- control system or by an experiment. The reactivity-control system shall function correctly in all facility states and shall be designed to discontinue chain reaction in the event of failure.

9. Reactor-trip system

The design shall provide for an automatic reactor-trip system. According to the reactor characteristics, supplementary independent reactor-trip system or several such systems may be provided. The effectiveness, expeditiousness and shutdown margin of the reactor-trip system shall ensure compliance with the prescribed limits and conditions. The reactor-trip system shall achieve its function even in the case of a single failure (e.g. failure to insert the reactor-trip rod with the highest reactivity value). One or several channels of manual actuation of reactor trip shall be provided. Verification of proper condition of the means for reactor trip shall be provided through instrumentation and testing. Computer-supported reactivity-control system software shall be verified and validated.

10. Reactor protection system

1. The operation of the reactor protection system shall be automatic and independent from any other system; the means shall also be provided to actuate reactor trip by a manual signal from the control room or from another location. Automatic actuation of the safety measures shall safely discontinue any further evolution of the event for all the postulated

initiating events. The design shall take into account single failures of system components. Manual interventions by the operator may be complied with provided that enough time is available, information is properly processed and indicated to allow easy diagnosis of the event and decisions for further action, and the operator is not under stress.

2. Manual actions may not obstruct or prevent automatic functions of the reactor protection system. The design shall ensure that no manual action is required within a short period immediately following the event. Automatically actuated safety measures shall continue to their completion and may not be cancelled. To restore operation, deliberate action by the operator shall be required. Safety-measure interlocks shall be assessed. Unnoticed actuation of such interlocks shall not be possible under any circumstances.

3. The reactor protection system design shall comply with the redundancy and independence principles to ensure automatic safety measures even in the case of single failures. Where required, fail-safe and diversity principles shall be applied to prevent a loss of the reactor- protection function. The reactor protection system shall achieve a safe state even in a case of a common-cause failure. Means shall be provided for functional testing of all system components.

4. The design shall specify the limitations to allow a sufficient safety margin between the point of safety-system actuation and the safety limit to allow the system to control the process before the safety limit is reached. The safety limit shall be determined based on instrumentation accuracy, calibration uncertainties, instrument drifts and instrument and system response times.

5. The computer-supported system constituting a part of the safety system, if any, shall meet the following requirements:

- its hardware and software shall meet the most stringent quality specifications, and shall ensure optimal operational performance and the highest practicable reliability;

- the overall development process, including surveillance, testing and implementation of modifications of the design, shall be systematically reviewed and documented;

- the computer-supported system shall be submitted to an independent expert assessment to verify the confidence level of its reliability;

- if a high level of confidence in the system cannot be achieved, an alternative method of achieving all the safety measures expected from the safety system shall be implemented.

11. Reactor-coolant system and associated systems

1. The reactor-coolant system shall ensure adequate cooling of the reactor core, with an additional margin. Long-term and reliable transfer of heat from the fuel to the ultimate heat sink shall be ensured.

2. Means shall be provided to allow testing and inspection of the systems containing reactor coolant, to detect any leaks, rapid development of cracks or brittle fracture spots and to monitor the rate of fault development. The principle of multiple barriers to radioactivity may be applied (e.g. primary system, contained within a pool, or a special design to contain any leakage). The reactor-coolant boundary shall allow pre-operational and operational inspection and testing.

3. The design of a water-cooled reactor shall consider the risk of core uncovering. For devices above the core with penetrations, special equipment shall be applied to prevent coolant loss (siphons, isolation devices).

4. If the core is cooled by means of a special system upon reactor trip, the primary cooling system shall be supplemented with a second residual-heat removal system of adequate reliability.

5. In cases of systems applying convection-coolant circulation as a safety system, several redundant devices shall be provided to comply with the single-failure principle. Operation of such systems shall be monitored and appropriate signals shall be transmitted to the reactor-protection system.

6. Where two systems under different pressures are linked, then either both of the systems shall be designed to withstand the higher of the two pressures, or means shall be provided to prevent excessive build-up of pressure in the lower pressure system in the event of a single failure.

7. Reactor coolant and moderator properties (e.g. pH, water conductivity) shall be monitored and controlled, and radioactive substances, including fission products, removed from the coolant.

8. Where necessary, an emergency core-cooling system shall be provided to prevent fuel damage in the event of loss of the primary coolant. The accident that the system is supposed to withstand shall be identified and analysed to demonstrate fulfilment of the system functions. The emergency core-cooling system shall maintain the fuel temperature within safety limits for an adequate period of time.

9. The emergency core-cooling system shall prevent fuel damage within the entire range of design-basis events involving loss of primary coolant. Special procedures shall be applied for beyond-design-basis events.

10. The emergency core-cooling system shall reliably fulfil design principles and achieve the expected function even in the event of a single failure. Means shall be provided for periodic functional testing of system components to verify the system's functionality.

12. Means for confinement (containment)

1. Where necessary, means shall be designed for the confinement of spreading of radioactivity to prevent the releases of radioactive substances (fission and activation products) that exceed allowable limits following an accident involving core damage. Means for confinement, which may consist of physical barriers enclosing the main components of a research reactor, shall prevent or limit inadvertent releases of radioactive substances in all facility states and in accident conditions. The physical barriers may consist of the reactor building, sumps and tanks for collecting effluents, ventilation system with filters for accident conditions, isolation devices on barrier penetrations and a point for releases into the environment at appropriate elevation. The design shall comply with the requirements for protection and safety systems set out in Item 5 of this Annex.

2. For proper performance of the confinement systems, the pressure within the barrier shall be set so as to prevent uncontrolled releases of radioactive substances into the environment. Variations of atmospheric conditions shall be taken into account. The design of confinement means shall take into account effects of emergency conditions (e.g. an explosion within the barrier) and environmental conditions at the time of accident, including circumstances caused by external and internal events (e.g. fire). The design shall provide an adequate margin for the highest pressure and temperature loads during an accident.

3. The acceptable rate of releases during an accident shall be determined taking into account isotopic composition of releases and factors such as filtering, point of release, environmental conditions, and pressure and temperature during an accident. Each penetration of the barriers shall be automatically and reliably isolated if, following an accident, conditions (including circumstances leading to pressure rises) arise that demand control of effluents to prevent radioactive releases into the environment exceeding allowable limits. Means shall be provided for initial and periodic testing to verify the functionality of the air-leak flow-rate and the ventilation system. If filtering is provided, filter efficiency shall be tested.

4. Research reactors representing major potential hazards to the environment shall be fitted with a containment to maintain releases during design-basis events, and internal and external events within specified limits. Special procedures shall be developed for the purposes of limiting consequences of beyond-design-basis events.

13. Experimental rig

1. The design of an experimental rig shall prevent the rig from compromising reactor safety in any facility state. The operation or failures of the experimental rig may not lead to an unacceptable variation of the reactor reactivity, obstruction to the core cooling or unacceptable irradiation. For each experimental rig, directly or indirectly associated with the reactor, design bases shall be specified, taking into account the radioactivity inventory of the rig and any energy generation or releases. A safety analysis shall be carried out, as well as an analysis of rig damages arising from the postulated reactor-initiating events.

2. In cases of the linking of the experimental rig with the reactor protection and safety system, the quality of the reactor protection and safety system shall be retained, and any potential harmful impacts on this system assessed.

3. Where necessary for reactor and experimental safety, means shall be provided to monitor experimental parameters in the reactor's control room.

4. Conditions shall be specified for safe use of the experimental rig, as well as criteria for reporting on devices and experiments to the Administration. Operational limits shall be specified for the devices, as well as conditions and limits for their safe operation, and these limits shall be incorporated in the research reactor's operational limits and conditions.

5. A preliminary plan of decommissioning shall be developed for the experimental rig.

14. Instrumentation and controls

1. The reactor shall be fitted with instruments to allow monitoring of operational parameters and process systems, and the recording of parameters important to safety. The reactor shall be provided with manual and automatic controls to maintain parameters within operational limits. The instruments designed to indicate and record reactor parameters during normal and abnormal operation shall also be suitable for design bases accidents. The design shall provide the means for inspection, testing and maintenance of instruments that are important for safety.

2. The required reliability of instrumentation and controls, which shall be specified depending on their importance for safety, shall be ensured through appropriate design, testing and verification of compliance with design specifications. Environment conditions shall be considered in the use and storage of instrumentation and control equipment, as well as potential effects of ambient factors (humidity, increased temperature, electromagnetic fields, etc.).

3. The design and testing of the software and hardware of computer-supported systems relevant to safety shall meet appropriate standards. Software for digital instrumentation and control shall be verified, validated and tested. Due to the complex nature of computer- supported systems, an additional degree of conservatism is necessary in their analyses.

15. Radiological protection

1. Radiological protection shall be ensured in all the states of a research reactor, through protection, ventilation and filtering, radioactive-substance decay systems (decay tanks), as well as the monitoring of radiation and presence of radioactive substances in the air. Furthermore, protection shall be provided for the experimental rig and ancillary facilities, taking into account the risk analysis.

2. Materials in structures close to the reactor core shall be selected so as to minimise irradiation loads on personnel during operation, inspection, testing, maintenance and decommissioning. The planning of radiological protection shall also consider radionuclides generated through neutron activation in the reactor systems. Areas within the facility shall be identified and marked according to the levels of radiological risks. Surfaces shall be designed to allow decontamination.

3. SSCs designed for radiological protection shall allow operational monitoring of radioactivity in all facility states, as well as, so far as is practicable, during beyond-design-basis events. Functions include:

- measurement of dose rates at locations normally occupied by personnel, and at selected locations during transients and accidents;

- measurement of activity in the atmosphere and rooms where there is a risk of spreading of radioactive substances in the air;

- measurement of the concentration of selected radionuclides in liquid process systems and in liquid samples taken from the environment and the facility during operation and during accidents;

- monitoring of radioactive effluents before and during their release into the environment;

- provision of surface and personnel contamination-measurement devices and personnel dose-measurement devices;

- monitoring of radioactivity at the points of access to the reactor to prevent unauthorised propagation of radioactive substances.

4. Indication of the measurement instruments referred to in the previous paragraph shall be provided in the control room and at other control points, if any. Based on monitoring measurements, propagation of contamination shall be prevented.

16. Fuel-handling systems

1. The design shall provide for the storage of spent fuel and procedures involved in the transport of fuel elements from the facility. Cooling of the irradiated fuel shall be provided. Limits and requirements for periodic testing specified in operational conditions and limits, and in the safety analysis report, shall be complied with. The option of removing the entire core from the reactor shall be available at all times.

2. The design shall provide for storage of irradiated fuel for extended time periods. The design of the fuel and radioactive-waste handling systems shall ensure:

- the prevention of inadvertent criticality through physical means, such as appropriate geometry or permanent neutron absorbers;

- minimised risk of fuel loss or damage, of impact from falling heavy objects and excessive loads on the fuel elements;

- storage of damaged fuel elements, control of chemical condition and coolant activity and provisions for periodic inspection and testing of the fuel;

- adequate physical protection against theft or sabotage, and verification of the identity of individual fuel elements.

17. Electrical power supply systems

1. Design bases shall be specified for the normal and emergency power-supply systems.

The design bases shall include the availability of a reliable power supply for the achievement of essential safety functions during design-basis events. The design shall

also consider uninterruptible power-supply sources. A reliable source shall be provided for the power supply of safety-related systems, taking into account start-up loads for the

equipment supplied from the source. Means shall be provided for the functional testing of the emergency power-supply system.

2. Maximum periods of the loss of alternating and direct current power-supply sources shall be specified. The selection and laying of power and instrumentation cables shall consider the risk of common-cause failures and prevent such failures by means of the separation and redundancy principles or by means of the selection of suitable materials.

18. Radioactive-waste handling systems

1. The design and operation of a research reactor shall minimise the volume of radioactive waste generated. The radioactive-waste handling systems shall minimise releases and maintain them below statutory limits through surveillance and monitoring. For solid and liquid radioactive waste, handling systems and on-site storage shall be provided.

2. To minimise the exposure of personnel to radiation and releases to the environment, the design shall provide for systems to protect against radionuclides and their decay. Means shall be provided to measure radioactive releases into the environment, such as release sampling and monitoring.

3. The design shall provide for the means of radioactive waste handling, collection, processing, storage and transport from the site. The liquid radioactive-waste handling system shall provide for leakage detection and capture of released substances.

19. SSCs

1. Safety-related SSCs shall be designed for all facility states, and when possible also for beyond-design-basis events. The design shall prevent levels of radiation and radioactive releases on the site and into the environment from exceeding regulatory limits, and shall minimise them. In accordance with the safety analysis of the reactor and its utilisation, the tightness of the reactor building and other buildings and structures containing radioactive substances shall be specified. Furthermore, requirements for the ventilation system shall be specified.

2. Reactor safety may not be compromised by a failure of any SSC, regardless of the importance of that SSC to safety. Appropriate measures shall be taken to prevent releases of radioactive substances into the environment in an event of a failure of an SSC containing radioactive substances.

3. Where reasonable, a notification system shall be arranged for the purposes of the safety of the reactor and associated facilities.

Annex 3: Low and intermediate level radioactive waste storage design bases

1. Requirements for SSCs

Safety related SSCs shall be designed to withstand the impacts of natural phenomena such as earthquakes, tornadoes, lightning, floods or combinations thereof, and to prevent massive collapse of building structures or falls of heavy objects on radioactive waste, or safety-related SSCs due to such collapse.

2. Containment barriers and systems

Adequate ventilation systems shall be provided to ensure confinement of airborne radioactive particles during normal and abnormal events.

Containment systems shall be monitored to an extent allowing the facility operator to detect the need for corrective measures for maintainance of the safe storage.

Radioactive waste shall be packaged in such a way to allow safe handling without discharges of radioactivity into the environment or irradiation exceeding allowable limits. Each package shall be designed at least for the entire operating lifetime of the storage or disposal.

3. The safety classification of SSCs

The safety classification of SSCs for radioactive waste storage shall use the same requirements as safety classification of SSCs of nuclear power plants referred in Item 2.1 of Annex 1 of these Rules.

4. Protection against internal fires

The design of fire protection shall comply, by analogy, with the requirements for nuclear power plants, laid down in Item 3 of Annex 1 to these Rules.

5. Radioactive waste package handling

The design of package-handling SSCs shall consider measures of protection against ionising radiation, easy maintenance and minimising of the risks of consequences of events or accidents.

A design of a storage facility shall always provide reserve storage capacity to allow for inspection, maintenance or remedial work, retrieval and waste management during emergency events.

The project shall ensure appropriate equipment and packaging for handling damaged radioactive waste packages in a short time upon the detection of the damage.

The operator shall implement a procedure for managing anomalies realted to loss of package integrity or their degradation to a level where they no longer meet the criteria for disposal or storage.

The storage facility shall be designed so as to allow removal of all waste within a reasonable time period following the termination of operation of the facility or in the conditions of intervention measures.

The operator's written procedures describing the method of acceptance of radioactive waste shall include instructions for safe handling of radioactive waste that does not meet criteria for acceptance in storage.

6. Recovery capabilities

The project of storage systems in the radioactive waste storage facility shall ensure accessibility of each radioactive waste package for inspection, relocation and maintenance, and ensure its readiness for further treatment or disposal.

For Krško nuclear power plant's storage of low and intermediate level radioactive waste the operator shall prepare special written procedures and measures to handle radioactive waste packages, which are not accessible.

Annex 4: High level radioactive-waste or spent-fuel storage design bases

1. Requirements for SSCs

Safety related SSCs shall be designed to withstand the impacts of internal and external events, including natural events which are site specific and events that are associated with human activity. The evaluation of external natural events shall consider earthquakes, flooding, extreme weather conditions, i.e. the effects of high and low temperatures, snow, ice, high winds, lightning, ice, and combinations thereof. Massive collapse of building structures or consequential heavy load drops on spent fuel, high level radioactivee waste, or other safety- related SSCs shall be prevented.

2. Design Extension Conditions

The design of the high level radioactive waste or spent-fuel storage shall comply with, by analogy, design extension conditions requirements laid down in Item 4 of Annex 1 to these Rules.

3. Containment barriers and systems

Sub-criticality, heat removal and prevention of uncontrolled releases of radioactive material shall be ensured in all facility states, including accidents.

Dry storage of spent fuel or high-level radioactive waste storage shall be based primarily on the use of passive / natural residual heat removal.

Spent-fuel cladding shall be protected against degradation that might lead to larger cracks, or, as an alternative, fuel shall be enclosed in some other way to prevent such degradation during storage as will compromise the safety of its later removal from the storage.

In the case of storing spent fuel or radioactive waste submerged in water, which serves for protection against ionising radiation and for containment of radioactive substances, the design of water-conditioning systems and water-level control systems for the storage pool shall prevent the risk of their abnormal operation or failure compromising water-safety limits.

Residual-heat removal shall be provided duringnormal operation, anticipated operational occurrences, design basis accidents and design extension conditions category A, taking into account the assumption of a single failure and the loss of off-site power.

Appropriate ventilation systems shall be provided to ensure the confinement of airborne radioactive particles during operational states, design basis accidents and design extension conditions category A.

Containment systems shall be monitored to an extent allowing the facility operator to detect the need for corrective measures to maintain safety of storage.

In the case of dry storage of spent fuel and high-level radioactive waste, the monitoring referred to in the previous paragraph shall be implemented in accordance with the design bases for spent-fuel or high-level radioactive waste containers. Periodic inspections as well as operational monitoring shall be ensured.

Spent fuel and high-level radioactive waste shall be packaged in the way ensuring their safe handling without releases of radioactivity into the environment or irradiation exceeding allowable limits. Each package shall be designed at least for the entire operating life of the storage facility.

For the large commercial aircraft crash to the storage facility the functionality of equipment important to safety shall be ensured during and after such an event.

4. The safety classification of SSCs

For the SSC safety classification of storage apply the same requirements as for SSC safety classification of nuclear power plants referred in Item 2.1 of Annex 1 of these Rules.

5. Protection against internal fires

The design of fire protection shall comply, by analogy, with the requirements for nuclear power plants, laid down in Item 3 of Annex 1 to these Rules.

6. Spent-fuel or high-level radioactive-waste package handling

The design of package-handling SSCs shall consider measures of protection against ionising radiation, easy maintenance and minimising of the risks of consequences of events or accidents.

A project of a storage facility shall always provide reserve capacity to allow inspection, maintenance or remedial work, and retrieval of spent fuel, unpackaged spent fuel elements or high-level radioactive, as well as their handling during emergency events.

The project shall ensure appropriate equipment and packaging for handling damaged spent- fuel or high-level radioactive-waste packages within a reasonable time period upon detection of the damage.

The operator shall implement a procedure for managing anomalies related to loss of package integrity or their degradation to a level where they no longer meet the criteria for disposal or storage.

A storage facility shall be designed so as to allow removal of all spent fuel or high-level radioactive waste within a reasonable time period following the termination of operation of the facility or in conditions of intervention measures.

The operator’s written procedures regulating the method of acceptance of spent fuel or high- level radioactive waste shall include instructions for safe handling of spent fuel or high-level radioactive waste that does not meet criteria for acceptance in storage.

7. Recovery capabilities

The project of storage systems in a spent fuel or high-level radioactive-waste storage shall ensure accessibility of the spent fuel, high-level radioactive waste or each radioactive-waste package for inspection, relocation and maintenance, and ensure its readiness for further treatment or disposal.

8. Safeguards of nuclear materials and surveillance measures

When storing spent fuel or high-level radioactive waste special attention shall be given to ensure adequate physical protection measures set out in the nuclear facilities' physical protection plan and procedures arising from the plan.

Storage facilities that are intended for the handling and storage of spent fuel or high-level radioactive waste shall be adequately protected to avoid unauthorized access or unauthorized removal of spent fuel or high level radioactive waste.

The operator shall ensure the undisrupted functioning of the equipment for surveillance of nuclear materials and activities (safeguards) installed by international organizations or the competent authorities, nuclear material record keeping, identity verification of individual fuel elements or containers of nuclear materials, and designing storage systems in a way to enable access to spent fuel and high-level radioactive waste for inspection.

9. Ageing Management

The operator shall ensure that processes of early detection, mitigation and removal of possible ageing mechanisms and their effects (including wear-out mechanisms and potential age related degradations) on SSCs important to safety are in place. .

The design phase of a storage consider all aspects of storage and structural properties of materials, such as corrosion, creep, concrete shrinkage, fatigue and changes due to the effects of radiation.

10. Prepardeness for decommissioning

The design shall provide prepardeness for decommissioning of a storage facility after the end of its operation, which is especially important in order to facilitate its decommissioning by minimizing radiation risk of personnel and population and with the smooth decontamination and dismantling of SSC, with the aim of reducing the amount and activity of radioactive waste generated.

It is necessary to maintain all the details of the storage required for its decommissioning, namely from design onwards.

Annex 5: Radioactive-waste or spent-fuel disposal design bases

1. Site characteristics

1. The site shall be located in an area of seismic and tectonic activity low enough to pose no threat to the disposal facility's isolating capability.

2. The frequency of surface processes such as disposal facility site flooding, landslides or erosion shall be low enough not to compromise the facility's capability of meeting safety requirements.

3. An integral characterisation of the facility's geological environment shall be carried out.

4. The site's geological environment shall contribute to the isolation of radioactive waste, to the confinement of releases of radionuclides into the environment and to the disposal facility's stability; it shall ensure adequate volume capacity and shall have characteristics favourable for facility project implementation.

5. The geometric, physical and chemical characteristics of the disposal facility site's geological environment shall retard migration of radionuclides from the facility into the environment in all phases of the facility's lifetime.

6. The dependency of the characteristics of the bedrock on future geodynamic phenomena (climate change, neotectonics, seismicity, volcanism, diapirism) shall be low enough to prevent their unacceptable compromising of the isolating capability of the entire facility.

7. Hydrogeological characteristics and the hydrogeological environment shall retard underground water flows, involve long transfer routes to limit migrations of radionuclides and contribute to safe isolation of waste for the specified time period.

8. Physical-chemical and geochemical characteristics of the geological and hydrogeological environment shall retard releases of radionuclides from the disposal facility into the environment and shall not significantly reduce the durability of technical barriers.

2. Design Extension Conditions

The design of the spent-fuel disposal facility shall comply with, by analogy, design extension conditions requirements laid down in Item 4 of Annex 1 to these Rules.

3. General requirements

1. Any provisions to facilitate reversal of disposal operations, or retrieval of waste packages disposed of, have no unacceptable effects on post‐closure safety.

2. The operator shall define and implement an appropriate program investigations, modelling, testing and monitoring activities with the purpose of providing an understanding of the evolution of the disposal system adequate for the safety case.

3. The operator shall ensure that any measures necessary for the purpose of accounting for and control of nuclear material shall not unacceptably affect operational and post‐closure safety.

4. The operator shall gather information during construction and operation to improve the knowledge of the intrinsic properties of the host environment and the response of the host environment to the presence of the disposal facility.

4. Design and construction conditions

1. The site's surface and underground characteristics shall allow optimisation of the design of surface structures or mining works in compliance with legal provisions regulating construction and mining works.

2. If construction, operation, decommissioning or closure activities of the disposal facility are taking place concurrently, the activities shall be performed in such a way, that they do not have an unacceptable effect on operational or post‐closure safety.

3. The investitor or the operator of the facility shall design the disposal facility in a way that the disposal system ensures operational and post‐closure safety. The project shall take into account the characteristics of the waste to be disposed, the characteristics of the site, as well as the feasibility of technical variants.

4. The investitor or operator shall design the disposal facility giving due consideration to potential changes or disturbances of the disposal system, which may affect post‐closure safety.

5. The investitor or operator shall design the disposal facility so that the engineered components (including barriers) are, to an adequate extent, physically and chemically compatible with each other, with the waste disposed of and with the host environment.

5. Human activities

1. The disposal facility site shall be selected with consideration of existing and potential future human activities on the site and in its immediate vicinity. The risk of a compromise of the facility's isolating capability or unacceptable consequences of human activities shall be minimised.

2. Land use and ownership of the site shall consider the envisaged development and regional-development plans in the region of the site.

3. Potential risks to the existing and envisaged future population of the site region arising from the disposal facility shall be acceptable.

4. The overall social impact of the facility in the site region shall be acceptable. Favourable impacts of the disposal facility site's selection on the region shall be maximised to the practicable extent. Negative social impacts shall be minimised.

6. Environmental protection

The disposal facility site shall ensure appropriate environmental protection over the entire period of operation and following the closure of the facility. Potential harmful impacts shall be mitigated to acceptable levels, considering economic, social and environmental aspects.

7. Transport of wastes

The disposal facility site shall be accessible through transport routes that allow transport of radioactive waste or spent fuel at the minimum risk to members of the public and allow the maintenance of exposure to irradiation and impacts on the environment during the transport of wastes to the site within allowable limits.

8. Limitation of doses

Following its closure, a disposal facility shall not impose a burden exceeding 0.3 mSv/year on a member of the public under the normal evolution of a disposal facility. In cases of alternative evolution of a disposal facility, the following measures shall be implemented, depending on the burden imposed on a member of the public:

a) for up to 10 mSv/year, no disposal facility optimisation measures are required;

b) for above 10 mSv/year, measures to minimise the probability of an alternative evolution of a disposal facility are required;

c) for above 100 mSv/year, measures to minimise the consequences of the alternative evolution of a disposal facility are required.

9. SSC safety classification

For the SSC safety classification of radioactive-waste or spent-fuel disposal facility apply the same requirements as for SSC safety classification of nuclear power plant in Item 2 Annex 1 of these Rules. The investor or operator of a disposal facility shall recognize and classify SSCs in accordance with their importance for operational or post‐closure safety.

10. Protection against internal fires

The design of fire protection shall comply, by analogy, with the requirements for nuclear power plants, laid down in Item 3 of Annex 1 to these Rules.

11. Common conditions for storage and disposal facility

In addition to the conditions laid down in items 1 to 7 of this Annex, a disposal facility and any facilities for the preparation or treatment prior to disposal shall also, by analogy, comply with design bases for radioactive-waste storage set out in Annex 3 of these Rules.

Annex 6: Mining-or hydrometallurgical-tailings disposal facility design bases

1. Disposal facility site

The selection of a disposal facility site shall consider the distance from populated areas, options to stabilise the facility, and minimisation of risks of erosion and of impacts and propagation of radioactive substances due to natural forces.

2. Slope inclinations

The design of inclinations of slopes of embankments and cover beds shall ensure their long- term geotechnical stability.

3. Cover bed

The design of the disposal facility cover bed shall ensure:

- durability and long-term resistance to erosion;

- isolating capabilities for preventing releases of radon from the disposal facility body and penetration of oxygen into the disposal facility body;

- protection against gamma radiation;

- proper regulation of atmospheric-precipitation infiltration;

- prevention of erosion through technical solutions;

- in the case of application of a clay isolating layer, presence of a water-containing layer to maintain humidity;

- minimised risk of punctures or loss of integrity of landfill liners because of the deep roots of the plants or animal operation (biointrusion).

4. Draining

Draining of surface and underground water from the disposal facility shall ensure the long-term stability of the disposal facility body and of the cover bed.

Annex 7: Design bases for irradiation rig or particle accelerator other than that for medical or veterinary application

1. Protection

The design of protection shall consider potential heat generation.

2. Radiation monitoring and warning

1. The locations and sensitivities of measurement instruments shall be designed so as to allow their immediate detection of excessive radioactive radiation levels and the prevention of irradiation of personnel through appropriate warnings and the discontinuation of radiation.

2. In the facility, at least 15 seconds before development of an area of elevated radiation, a visual and acoustic warning signal shall be emitted, and the visual signal shall remain active throughout the period of elevated radiation level.

3. Access control

The safety-related SSCs shall be designed to prevent any unauthorised access in the periods when the facility is not in the state of operation.

4. Radioactive source-carrier systems and restoration of the state of protection

1. The design of the source-carrier systems for irradiation devices shall prevent any development of corrosion on the source or on the contact of the source with the system, prevent damages to the radiation source in the event of power loss, dropping or impact, and prevent the risk of jamming of the movement mechanisms or minimise the risks of such jamming to personnel.

2. The design of a carrier system shall minimise the time of restoration of the state of protection in the event of power loss.

5. Emergency stop

A particle accelerator shall be fitted with a clearly marked emergency stop button in the area of elevated radiation.

Annex 8: Protection of digital computer and communication systems and networks

1. The opeartor shall provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat in accordance with requirements of this Annex.

2. The operator shall protect digital computer and communication systems and networks associated with:

a) safety-related and important-to-safety functions;

b) security functions;

c) emergency preparedness functions, including offsite communications; and

d) support systems and equipment which, if compromised, would adversely impact safety, security, or emergency preparedness functions.

3. The operator shall protect the systems and networks identified in Item 2 of this Annex from cyber attacks that would:

a) adversely impact the integrity or confidentiality of data and/or software;

b) deny access to systems, services, and/or data; and

c) adversely impact the operation of systems, networks, and associated equipment.

4. To accomplish requirements the above Items, the operator shall:

a) analyze digital computer and communication systems and networks and identify those assets that shall be protected against cyber attacks to satisfy requirements the above Items of this Annex;

b) establish, implement, and maintain a cyber security program for the protection of the assets identified in Item 4a of this Annex; and

c) incorporate the cyber security program as a component of the physical protection program.

5. The cyber security program shall be designed to:

a) implement security controls to protect the assets identified by Item 4a of this

Annex from cyber attacks;

b) apply and maintain defense-in-depth protective strategies to ensure the capability to detect, respond to, and recover from cyber attacks;

c) mitigate the adverse affects of cyber attacks; and

d) ensure that the functions of protected assets identified by Item 4a of this Annex are not adversely impacted due to cyber attacks.

6. As part of the cyber security program, the operator shall:

a) ensure that appropriate facility personnel, including contractors, are aware of cyber security requirements and receive the training necessary to perform their assigned duties and responsibilities;

b) evaluate and manage cyber risks;

c) ensure that modifications to assets, identified by Item 4a of this Annex, are evaluated before implementation to ensure that the cyber security performance objectives identified in Item 2 of this Annex are maintained.

7. The cyber security program shall describe how the requirements of this Annex will be implemented and shall account for the site-specific conditions that affect implementation.

8. The cyber security program shall include measures for incident response and recovery for cyber attacks. The cyber security program shall describe how the operator will:

a) maintain the capability for timely detection and response to cyber attacks;

b) mitigate the consequences of cyber attacks;

c) correct exploited vulnerabilities; and

d) restore affected systems, networks, and/or equipment affected by cyber attacks.

9. The licensee shall develop and maintain written procedures to implement the cyber security program.