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NUCLEAR SAFETY RULES FOR REACTOR INSTALLATIONS OF NUCLEAR POWER PLANTS

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NUCLEAR SAFETY RULES FOR REACTOR INSTALLATIONS OF NUCLEAR POWER PLANTS

Federal Environmental, Industrial and Nuclear Supervision Service of Russia
Moscow, 2007

These federal standards and rules “Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants” establish requirements for nuclear safety ensurance of reactor installations of nuclear power plants during design, engineering, construction and operation.

These federal standards and rules are issued to substitute the Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants PBYa RU AS-89 with Alteration №1 and Section 4 of Nuclear Safety Rules for Nuclear Power Plants PBYa-04-74*.

* The document has been developed by specialists at SEC NRS, taking into account comments and proposals produced by the following organizations: FSUE ”State Research and Design Institute Atomenergoproekt”, FSUE “Russian State Concern for the Production of Electrical and Thermal Energy at Nuclear Power Plants”, FSUE “All-Russia Research Institute for Inorganic Materials” named after A.A. Bochvar, FSUE “All-Russia Design and Research Institute for Integrated Power Technology”, FSUE “Experimental Design Bureau “Gidropress”, FSUE “State Research Center of the Russian Federation Institute for Physics and Power Engineering named after A.I. Leipunsky”, Joint Stock Company “Novosibirsk Chemical Concentrates Plant”, FSUE “State Research Center of the Russian Federation Research Institute of Atomic Reactors”, Joint Stock Company “Mashinostroitelny Zavod”, Joint Stock Company “Murmansk Shipping Company”, Joint Stock Company “TVEL”, FSUE “Experimental Design Bureau of Industrial Engineering named after A.A. Afrikantov”, FSUE “Mining and Chemical Combine”, FSUE Production Association “Mayak”, FSUE “Research and Development Institute of Power Engineering named after N.A. Dollezhal”, Balakovo NPP, Beloyarsk NPP, Bilibino NPP, Kalinin NPP, Kola NPP, Kursk NPP, Leningrad NPP, Smolensk NPP Department of Nuclear and Radiation Safety of the Federal Atomic Energy Agency, RRC “Kurchatov Institute”,.

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LIST OF ABBREVIATIONS

AZ	Emergency Protection
BN	fast neutron reactor
BSC (BSP)	Back-up Shutdown Center (Panel)
CPS	Control and Protection System
CSS	Controlling Safety Systems
ECR	Emergency Control Room
EGP-6	graphite-moderated loop-type reactor
FA	Fuel Assembly
FFLS	Failed Fuel Location System
FR	Fuel Rod
MCR	Main Control Room
NHP	Nuclear Heating Plant
NOCS	Normal Operation Control Systems
NPP	Nuclear Power Plant
PP	Preventive Protection
RBMK	high power channel-type reactor
RI	Reactor Installation
SAR	Safety Analysis Report
SIS	Safety Important System
VVER	Water-Water Power Reactor

BASIC TERMS AND DEFINITIONS

1. **Accident** shall mean the operational event at a NPP where radioactive substances and(or) ionizing radiation have been released beyond design boundaries of RI and NPP envisaged for normal operation and in quantities which exceed safe operation limits. An accident is characterized by an initiating event, development paths and consequences.

2. **AZ rod** shall mean the device to affect reactivity which is used in AZ (emergency protection system).

3. **Control and protection system** shall mean the combination of technical, software and information support means to ensure safe behavior of the chain fission reaction.

   The control and protection system is the safety important system combining normal operation functions and safety functions and consisting of elements pertaining to normal operation control systems, protective, controlling and supporting safety systems.

4. **Control train** shall mean the combination of sensors, communication circuits, signal processing and/or parameter display means which is designed to exercise control in the scope envisaged by the design.

5. **Core refueling (refueling)** shall mean the nuclear hazardous operations at RI related to loading, withdrawal and moving of FEs (FRs), reactivity controls and other components affecting reactivity for their repair, replacement and dismantling.

6. **CPS actuator** shall mean the device consisting of a drive mechanism, control rods and connecting elements which is designed to change reactivity of the reactor.

7. **CPS drive** shall mean the device intended for changing position of the CPS mechanical rod and keeping it in fixed position.

8. **CPS rod** shall mean the reactivity controls used by CPS.

9. **CPS rod position indicator** shall mean the device to determine a position of a CPS rod in the reactor core.

10. **Diagnostics** shall mean the control function which purpose is to determine whether the object of diagnostics is workable (unworkable) or in order (failed).

11. **Diversity principle** shall mean the principle of reliability improvement through application in different systems (or within one system in different channels) of different means and/or similar means based on diverse functional principles.

12. **Emergency alarm signal** shall mean the signal generated and recorded by the control and monitoring equipment to trigger preventive protection and warn the personnel on possible operational events.

13. **Emergency protection (AZ)** shall mean:
    - the safety function allowing for fast rendering the reactor subcritical and maintaining it as such;
    - the complex of safety systems performing the AZ function.

14. **Emergency protection hardware set** shall mean the control and protection system hardware which performs functions of monitoring and control over emergency protection in the scope envisaged by the RI design.

15. **Emergency protection signal** shall mean the signal generated by AZ hardware to actuate AZ rods and which is sent to recording devices as well as to MCR and BCR for the personnel notification.

16. **Equivalent FR cladding oxidizing depth** shall mean the total thickness of equivalent layer

which would react with water steam in the assumption that all locally consumed oxygen would have been used to generate the stoichiometric zirconium dioxide ZrO_2 as reduced to the cladding initial thickness. In the event of the cladding loss of integrity oxidizing of both outer and inner cladding surfaces is considered.

17. **Fuel assembly** shall mean the engineering item containing nuclear materials, which is intended for generation of thermal energy in a nuclear reactor through the controlled nuclear reaction.

18. **Fuel rod (FR)** shall mean the individual assembly unit containing nuclear materials, which is intended for generation of thermal energy in a nuclear reactor through the controlled nuclear reaction and (or) accumulation of nuclides.

19. **Fuel rod damage** shall mean violation of at least one of design damage limits established for fuel rods.

20. **Fuel rod destruction** shall mean the loss of integrity of the fuel rod structure to result in a loss of the FR geometry which ensures its design cooling.

21. **Fuel rod leak** shall mean the fuel rod damage associated with loss of integrity of its cladding of a gas leak type or direct contact of nuclear fuel with the coolant.

22. **Group of CPS rods** shall mean one or several CPS rods combined together for the purposes of control to be moved simultaneously and affect reactivity.

23. **Independence principle** shall mean the principle of the system reliability improvement through application of functional and/or physical separation of trains (elements) where, if implemented, a failure of one train (element) does not lead to a failure of another train (element).

24. **Maximum design fuel rod damage limit** shall mean maximum permissible values of fuel rod parameters and characteristics under conditions of design basis accidents, which, if exceeded, may cause destruction of fuel rods.

25. **Maximum reactivity margin** shall mean reactivity which may occur in the reactor when all reactivity controls and other removable absorbers are withdrawn from the core at the moment of a fuel cycle and in the reactor state featuring the maximum value of the effective multiplication factor.

26. **Monitoring** shall mean the part of control function which is to evaluate a parameter value or determine (identify) a state of the process or equipment subject to monitoring.

27. **Nuclear accident** shall mean the accident involving a fuel rod damage in excess of the safe operation limits and/or personnel exposure in excess of the permissible limits, as initiated by:
    - disruption of monitoring and control over the nuclear fission reaction in the reactor core;
    - criticality during refueling, transportation and storage of nuclear fuel;
    - disruption of heat removal from FRs;
    - other causes leading to FR damage.

28. **Nuclear hazardous operations** shall mean the operations being performed at RI, which under certain conditions may lead to a nuclear accident.

29. **Nuclear safety** shall mean the RI and NPP property to prevent a nuclear accident with a certain probability.

30. **Preventive protection** shall mean the function performed by the NPP unit normal operation control system to prevent AZ system actuation and/or violation of safe operation limits and conditions.

31. **Reactivity controls** shall mean the engineering means such as solid, liquid or gaseous absorbers (moderators, reflectors) which provide a change in reactivity in the reactor core if

their position or state is changed in the reactor core.

32. **Reactor** shall mean the device for carrying out the controlled nuclear fission reaction to generate thermal energy.

33. **Reactor core** shall mean the section of the reactor where nuclear fuel, moderator, absorber, coolant, reactivity controls and structural components designed to provide for the controlled nuclear chain fission reaction and to transfer energy to the coolant are located.

34. **Reactor installation** shall mean the complex of systems and components of NPP designed to convert nuclear energy into thermal energy. It includes the reactor and directly related systems necessary for its normal operation, emergency cooling, emergency protection, maintaining it in safe condition provided the required auxiliary and supporting functions are performed by other NPP systems. The RI boundaries are established for each NPP in the design.

35. **Reactor shutdown system** shall mean the system intended for rendering the reactor subcritical and maintaining it as such using the reactivity controls.

36. **Redundancy principle** shall mean the principle of the system reliability improvement through application of structural, functional and information redundancy in the scope that is maximum necessary and sufficient for the systems to perform the designated functions.

37. **RI operator** shall mean the individual from the operating personnel who directly controls RI at MCR (BCR).

38. **Severe core damage** shall mean the beyond design basis accident where fuel rods damage exceeds the maximum design limit and the permissible limit of radioactive substance emergency release to the atmosphere can be achieved.

39. **Withdrawal of reactivity controls** shall mean such movement or alteration of state of the reactivity controls which leads to introduction of positive reactivity (insertion of the reactivity controls leads to introduction of negative reactivity).

1. Purpose and scope

1.1. These Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants shall cover all NPPs which are under design, engineering, construction and operation in Russia.

1.2. These Rules establish requirements for design, characteristics and operation conditions of RI systems and components as well as organizational requirements regarding nuclear safety ensurance during design, engineering, construction and operation of RIs and NPPs.

1.3. The Rules have been developed on the basis of the General Safety Provisions for Nuclear Power Plants (GSP) and the experience gained in design, development, construction and operation of NPPs. The Rules specify the GSP requirements as regards RI and NPP nuclear safety ensurance except for nuclear fuel storage and transportation requirements.

1.4. Nuclear safety of a RI and NPP is determined by technical perfection of designs; required quality of manufacturing, assembling, aligning and testing of safety important systems and components; their operational reliability; diagnostics of technical conditions of the equipment; quality and timeliness of maintenance and repair of the equipment; monitoring and control over processes during operation; organization of work; and qualifications and discipline of the personnel.

1.5. Nuclear safety of RI and NPP is ensured by a system of technical and organizational measures envisaged by the defense-in-depth concept, including:

- implementation and further development of inherent safety features;
- use of safety systems built on the basis of the principles of independence, diversity and redundancy, and single failure criterion;
- use of reliable, field-proven technical solutions and justified methodologies, calculation analyses and experimental studies;
- following the RI and NPP safety norms, rules and standards, and design requirements;
- stability of processes;
- implementation of quality assurance systems at all stages of creation and operation of NPP;
- building and implementing safety culture at all stages of creation and operation of NPP.

2. Nuclear safety requirements for reactor and other safety important systems

2.1. General requirements

2.1.1. Design, construction and operation of an NPP power unit, as well as development and manufacturing of RI and NPP components shall be carried out in compliance with the applicable regulatory documents on NPP safety.

2.1.2. The RI design and NPP deign development shall precede the NPP construction. The RI and NPP designs shall identify safety important systems, their main characteristics, reliability, service life, as well as their operating sequence, operating conditions, relevant means of monitoring and diagnostics.

2.1.3. Changes to composition, design and/or characteristics of RI and its safety important systems as well as operating conditions of the NPP shall not be done without corresponding modifications introduced to the designs of RI and NPP.

2.1.4. In the RI design development and(or) modernization of the reactor core, when new designs of fuel assemblies and new fuel compositions are applied, the improvement of control and protection system and other safety important systems, a required scope of bench and in-pile tests shall be carried out. The RI design shall demonstrate that the sufficient number of studies has been conducted to prove that the required safety criteria are met.

2.1.5. Quality assurance programs shall be developed for all stages of RI and NPP life cycle.

2.1.6. To maintain and verify design characteristics the safety important RI and NPP systems (components) shall be subjected to inspections and tests during their manufacturing, assembling, aligning, as well as to periodic in-service inspections.

The RI and NPP designs shall provide for tooling, devices, methodologies and frequencies of safety important systems checks against their design characteristics, including comprehensive testing (signal sequence and transmission time including those of AZ response, switching over to emergency power supply sources, performance of safety functions, etc.).

The RI and NPP designs shall contain lists of systems and components which performance and characteristics are to be verified at the operating or shutdown reactor, along with a description of RI and safety important RI and NPP systems' conditions.

Devices and methodologies for inspection of safety important systems and their components shall not affect NPP safety.

2.1.7. The Safety Analysis Report of Nuclear Power Plant (corresponding SAR NPP sections) shall be the main document justifying nuclear safety of RI. In case of NPPs for which no safety analysis report has been developed, such document shall be an existing Technical Safety Analysis (TSA) or an In-depth Safety Analysis Report (ISAR). A SAR NPP shall be developed by the operating organization while assuring that SAR NPP is consistent with RI and NPP designs.

2.1.8. The RI and NPP designs shall establish and SAR NPP shall contain a list of initiating events of design basis accidents and a list of beyond design basis accidents; a classification of design and beyond design basis accidents by their frequency of occurrence and severity of consequences; as well as an analysis of design basis and beyond design basis accidents and their consequences. Among beyond design basis accidents the severe core damage accidents shall be considered.

2.1.9. During RI design process one shall pursue that the cumulative frequency of severe beyond design basis accidents estimated on the basis of the probabilistic safety assessment would not exceed 10^{-5} per reactor-year.

2.1.10. The RI and NPP designs shall contain an analysis of possible failures of safety important systems (components) identifying failures dangerous for RI and NPP and evaluating their consequences on the basis of probabilistic and deterministic safety analyses.

2.1.11. The RI and NPP designs shall list and justify operational limits and conditions, safe operation limits and conditions and design limits established for design basis accidents.

2.1.12. For each design basis accident or a group of accidents the RI and NPP design shall set corresponding design limits for design basis accidents, which shall not be exceeded considering actuation of safety systems.

2.1.13. The RI designs shall demonstrate that the maximum design fuel damage limit established for design basis accidents with the severest consequences is not exceeded.

The RI designs shall establish design fuel damage limits for other design basis accidents; their values shall be less than the maximum design fuel damage limit.

The fuel damage limits for NPPs with the most commonly used reactor installations in Russia are given in Appendix.

2.1.14. The RI and NPP designs shall list and justify nuclear hazardous operations.

2.1.15. The RI and NPP designs shall list methodologies and codes used for safety justification, and those used in SIS, along with the scope of their application. The codes and methodologies in use shall be verified and certified in accordance with the established procedures.

2.2. Core and its structural components

2.2.1. The core shall be designed such that any reactivity changes during normal operation and operational events, including design basis accidents, do not lead to violation of the corresponding fuel damage limits.

Requirements for reactivity coefficients of NPP reactors with RI types most commonly used in Russia are given in Appendix.

2.2.2. The RI and NPP designs shall demonstrate that during design basis accidents involving reactivity fast build-up fuel enthalpy, as averaged over a pellet transversal cross-section (average radial), shall not exceed the limiting value established in the design basing on experimental data, as well as fuel rods and fuel assemblies destruction shall be ruled out. Conditions shall be given for beyond design basis accidents under which destruction of some fuel rods and fuel assemblies could be possible.

2.2.3. The RI design shall establish correspondence between fuel damage limits and primary coolant activity as regards the reference radionuclides considering efficiency of the coolant clean-up systems.

2.2.4. To justify that the requirements stating that the safe operation limits for fuel damage should not be exceeded are met in case of operational events, the RI design shall contain an analysis of thermal and engineering reliability of the core along with a justification that the margins envisaged in the RI design are sufficient.

2.2.5. The FR cladding oxidizing during the RI operation shall not lead to their over-embrittlement. The RI design shall justify (on the basis of the experimental data) and contain values of the equivalent oxidizing depth of the FR cladding for normal operation and operational events, including design basis accidents.

2.2.6. For fast neutron reactors with sodium coolant it shall be demonstrated that sodium coolant voids are excluded during normal operation and operational events, including design basis accidents.

2.2.7. The core and its components' (including fuel rods and fuel assemblies) design and implementation shall be such that during normal operation and operational events, including design basis accidents, the corresponding fuel damage limits would not be exceeded considering:

- RI design operational modes, their number and design course;
- force (mechanical), thermal and radiation impact to the core components;
- physical and chemical interaction of the core structural materials and coolant;
- limiting deviations from design, process characteristics and process parameters;
- shock and vibration impacts, thermal cycling, radiation and thermal creep and

material aging;

- influence of the coolant fission products and admixtures on strength and corrosion resistance of fuel rods;
- other factors that degrade mechanical properties of the core structural materials and integrity of the FR cladding.

2.2.8. The RI and NPP design shall justify and provide for engineered means for the possibility to remove damaged core components after a design basis accident.

2.2.9. The core and CPS actuators shall be designed so that control rod jamming, ejection or their spontaneous disengaging with CPS drives are excluded.

2.2.10. The RI design shall demonstrate that in case of unanticipated movement of the rod of highest worth or a group of CPS rods the fuel rods are not damaged to result in violation of the safe operation limits considering AZ actuation without the CPS rod of highest worth.

2.2.11. During normal operation and operational events including design basis accidents the possibility shall be excluded for unanticipated movements and(or) deformations of the core components inducing reactivity growth and degrading of heat removal which lead to fuel damage in excess of the corresponding design limits.

2.2.12. The RI and NPP designs shall demonstrate and justify that in case of seismic impacts typical of the NPP site the unobstructed insertion of the control rods and AZ rods into the core and reliable cooling are ensured.

2.2.13. The core and reactivity controls' characteristics shall be so that the insertion of reactivity controls into the core and(or) reflector for any combination of positions thereof during normal operation and operational events, including design basis accidents, ensures introduction of negative reactivity at any section of their moving route.

2.2.14. The FA design shall be so that changes in shape of fuel rods and other FA components, which are possible during normal operation, operational events and design basis accidents, do not lead to blocking of FA flow area resulting in fuel damage in excess of the corresponding limits and do not disrupt normal performance of CPS rods.

2.2.15. The FA design shall have distinguishing markings showing nuclide composition and enrichment of fuel in FRs. This marking shall be identified visually and(or) with the help of refueling devices.

2.2.16. Fuel rods of different enrichment, those containing burnable absorber in fuel, those with mixed fuel, and the like, special burnable absorbers within FAs shall bear distinguishing markings legible for visual monitoring means and(or) industrial monitoring means during assembling of FAs.

2.2.17. The RI and NPP designs shall provide for engineered means and monitoring methods for fuel cladding leaktightness to be applied at the shutdown and(or) operating reactor. Such engineered means and monitoring methods shall ensure reliable and timely detection of leaking FAs (fuel rods). There shall be criteria for rejecting leaking fuel rods (FAs). The RI and NPP design shall list and justify methodologies used for fuel cladding leaktightness monitoring at shutdown and(or) operating reactor.

2.3. Control and protection systems

2.3.1. General requirements

2.3.1.1. The RI shall incorporate control and protection systems designed for:

- control over reactivity of the reactor core and RI power;
- monitoring of the neutron flux (power) density, its rate of changes, process

parameters necessary for protection and control over reactivity of the reactor core and RI power;
- rendering the reactor subcritical and maintaining it as such.

2.3.1.2. Composition, architecture, characteristics and operational sequence of control and protection systems shall be justified in the RI design. The RI design shall include a quantitative reliability analysis to demonstrate that CPS reliability indicators meet the requirements of regulatory documents they are subject to.

2.3.1.3. The RI design shall include an analysis of CPS response to external and internal impacts (fires, earthquakes, flooding, electromagnetic noise, etc.), to possible malfunctions and failures (short circuits, insulation degrading, voltage drops and pickups, false actuation, loss of controls, etc.), which demonstrate that there are no responses that may be hazardous for RI.

Should the control and protection system responses hazardous for RI be revealed during operation, RI shall be shutdown and the measures shall be taken to exclude such responses. The operating organization shall ensure that the corresponding changes are done to the RI design in accordance with the established procedure.

2.3.1.4. The RI design shall provide for, at least, two reactor shutdown systems, each one being capable, independently from the other, of rendering the reactor subcritical and maintaining it in this state considering single failure criterion or human error. These systems shall be designed in accordance with the diversity, independence and redundancy principles.

3.2.1.5. During normal operation, operational events including design basis accidents at least one of the reactor shutdown systems (which does not perform the AZ function) shall have:
- efficiency that is sufficient for rendering the reactor subcritical and maintaining it as such, considering possible reactivity release;
- response time sufficient for rendering the reactor subcritical without violation of design fuel damage limits established for design basis accidents (taking into account performance of the emergency core cooling systems).

3.2.1.6. The RI design shall determine and justify the number, worth, location, group composition, operating positions, sequence and moving speed of CPS rods (including emergency protection rods) as well as the number of their drives.

3.2.1.7. The RI design shall determine and justify methods and conditions of tests, replacement and repair of CPS rods, their drives and other reactivity controls.

3.2.1.8. The CPS actuators shall have indicators of intermediate positions of their rods, terminal position indicators and terminal circuit breakers which are actuated, if possible, directly by the rod. Other prompt reactivity controls shall have positioning and(or) terminal position indicators.

3.2.1.9. In case the RI design envisages the use of an additional (to the standard) CPS system during the first criticality of the reactor, this system shall meet the requirements of Section 2.3 as regards the CPS system.

2.3.2. Emergency protection system

2.3.2.1. At least one of the two reactor shutdown system shall perform the emergency protection function.

2.3.2.2. The RI design shall demonstrate that the rods performing the AZ function possess the features below without a rod of highest worth:

- response time sufficient to render the reactor subcritical without violating safe operation limits in case of operational events;
- worth sufficient for rendering the reactor subcritical and maintaining it in this state in case of operational events, including design basis accidents.

In case the AZ effectiveness is not sufficient enough to maintain the reactor subcritical for a prolonged period of time, the RI design shall provide for automatic actuation of the other reactor shutdown system (systems) having higher effectiveness and sufficient for rendering the reactor subcritical considering possible release of positive reactivity.

2.3.2.3. The emergency protection shall have at least two independent rod groups.

2.3.2.4. The emergency protection shall be designed so that the initiated protective action shall be fully performed considering the requirements of para. 2.3.2.2, and monitoring of the emergency protection function performance shall be ensured.

2.3.2.5. The RI design shall contain a procedure for determining and eliminating causes that led to actuation of the emergency protection as well as a sequence of operating personnel actions to restore normal operation of RI after AZ has actuated.

2.3.2.6. At an AZ signal the AZ rods shall be actuated irrespectively of their current position, operating or intermediate.

2.3.2.7. The design reactivity controls, as provided in the RI design shall exclude introduction of positive reactivity if the emergency protection rods are not placed in operating position. The RI design shall determine the operating positions of AZ rods and their withdrawal sequence.

2.3.2.8. When the reactivity controls perform the combined functions of normal operation and emergency protection the RI design shall contain and justify their functioning procedure. At this, the AZ priority functioning shall be ensured.

2.3.2.9. The AZ structure shall be selected so as to provide compliance with the mandatory criteria (single failure, common cause failure) and meet reliability indicators.

2.3.2.10. The AZ hardware shall include at least two independent sets.

2.3.2.11. Each AZ hardware set shall be designed so that the protection is ensured over the range of neutron flux density changes from 10^{-7} % up to 120% of nominal value as regards:

- neutron flux density – by at least three independent trains;
- neutron flux build-up rate – by at least three independent trains.

2.3.2.12. When it is necessary to divide the neutron flux density measurement range into several sub-ranges it shall be arranged so that the measurement sub-ranges are overlapped within not less than one decimal exponent in the neutron flux density units of measure and provided with automatic switching of sub-ranges.

The possibility of connecting a recording device to each neutron flux density measurement channel shall be provided.

2.3.2.13. Each AZ hardware set shall be designed so that the emergency protection is ensured over the entire range of process parameter measurements by not less than three independent channels for each process parameter at which the protection is required.

2.3.2.14. Each emergency protection hardware set shall be designed basing on the majority logic which is to be selected through the reliability analysis given in the RI design. The minimal majority rate shall be 2 out of 3.

The each set’s control commands designated for AZ actuators shall be transmitted at least via two trains.

2.3.2.15. The RI design shall justify that it is permissible to combine, in each emergency protection hardware set, the measuring sections of neutron flux monitoring channel and the measuring sections of the neutron flux build-up rate monitoring channels.

2.3.2.16. The emergency protection shall be segregated from NOCS to a degree at which a termination of operation or failure of any of NOCS components does not affect capabilities of the emergency protection to perform designated functions.

2.3.2.17. A failure in the parameter display channel, information recording channel and diagnostics channel shall not affect capabilities of that channel to perform emergency protection functions.

2.3.2.18. For each of the channels and for the whole set the possibility shall be provided of testing the emergency protection signal generation and transmitting time without actuation of AZ rods.

2.3.2.19. The emergency protection system shall provide for automatic monitoring and performance diagnostics of the emergency protection hardware sets and protection channel, the channel failure information being displayed at MCR, as well as for AZ signals indicating failures of channels or sets.

2.3.2.20. The RI design shall justify and list methodologies of metrological certification and checks of the AZ hardware.

2.3.2.21. The RI design shall justify permissibility and conditions of terminating operation of one set or one channel in the AZ set (duration, RI power, conditions of other sets etc.).

2.3.2.22. When one channel of one of the AZ hardware sets is put out of operation while the set remains operative the emergency signal for this channel shall be automatically generated.

2.3.2.23. A list of parameters at which the emergency protection functions shall be performed, emergency protection actuation settings and conditions as well as signal transmitting time before AZ rods actuate shall be justified in the RI design. Emergency protection actuation settings and conditions shall be selected so as to prevent violation of safe operation limits.

2.3.2.24. The RI design shall contain and justify a list of initiating events requiring actuation of the emergency protection.

2.3.2.25. As minimum, the AZ shall actuate in the following cases:

- when the AZ neutron flux density setting is approached;
- when the AZ neutron flux density build-up rate setting is approached;
- when voltage is lost at any emergency protection hardware set and CPS power supply buses;
- when any two out of three protection channels for neutron flux density and neutron flux build-up rate fail in any AZ hardware set remaining in operation;
- when the AZ process parameter settings requiring actuation of protection features are approached;
- when the emergency protection is actuated by the MCR (BCR) switch.

2.3.2.26. The RI design shall justify whether the use of the preventive protection (protections) in case of operational events not requiring AZ actuation is permitted along with it (their) conditions of use.

2.3.2.27. The emergency protection shall be designed so that the technical features exclude a possibility of manipulating the controls of emergency protection channels and alterating the settings (in the manner that is not envisaged by the RI design and safe operation regulations of the NPP power unit) without notifying the personnel and actuation of the emergency protection rods.

2.3.2.28. Performance of the reactor emergency protection functions shall not depend on availability and conditions of the power supply sources.

2.3.3. Neutron flux and reactivity control

2.3.3.1. To monitor neutron flux the reactor shall be equipped with the monitoring channels so as to provide for the monitoring over the entire range of changes of neutron flux density in the core from 10^{-7} % up to 120 % of nominal value by, as minimum:

- three independent from each other neutron flux density measurement channels with indicating instruments;
- three independent from each other neutron flux density change measurement channels.

2.3.3.2. The RI design shall justify that it is permissible to combine the measuring sections of neutron flux monitoring channel and the measuring sections of the neutron flux density change monitoring channels.

2.3.3.3. At least two out of three neutron flux density monitoring channels shall be equipped with recording devices which can be connected to any neutron flux density monitoring channel. These recording devices shall provide for the possibility of taking measurements and recording of instrument readings over the entire design range of neutron flux density changes.

2.3.3.4. Neutron flux density monitoring channels shall be calibrated over the entire design range of the reactor thermal power changes. The RI design shall justify, identify methodology and describe a conduct procedure of such calibration and its frequency during operation of the NPP power unit.

2.3.3.5. When the neutron flux density measurement range is divided into several sub-ranges it shall be arranged so that the measurement sub-ranges are overlapped within not less than one decimal exponent in the neutron flux density units of measure and provided with automatic switching over the sub-ranges.

2.3.3.6. If the neutron flux density monitoring channels indicated in para. 2.3.3.1 do not provide for neutron flux monitoring during loading (refueling) of the core, the reactor shall be equipped with an additional monitoring system. The additional monitoring system may be removable and be installed only for the core loading and refueling periods. It shall include not less than three independent channels for neutron flux density monitoring and displaying and recording devices.

2.3.3.7. To monitor reactivity changes the RI design shall provide for a reactivity measurement instrument fitted with sensors, prompt information displays, recorders and capabilities for automatic switching over neutron flux density and reactivity measurement ranges.

2.3.3.8. The RI design shall justify the methodology and errors of determining reactivity (number and locations of sensors, calculation algorithms and constants, measurement errors and ranges).

2.3.3.9. Reactivity monitoring channels shall be equipped with the means of automatic performance check and failure warning alarms.

2.3.3.10. The RI design shall justify and describe methodologies of metrological certification and checks of reactivity control channels.

2.3.3.11. The RI design shall justify and set up characteristics of the automatic RI power control system, which ensure RI operation without violation of operational limits. The possibility and permissible time of RI operation without the automatic power control system, in particular, when it fails, and the permissible RI power in this mode shall be justified in the RI design.

2.3.3.12. If several measurement channels are connected to the input of the automatic power control system, there shall be a device to receive signals from operating measurement channels which will ensure that there are no reactor power changes induced by the automatic control system in case of switching off or failure of one of these channels.

2.3.3.13. For RIs which are refueled when the reactor is shutdown there shall be technical measures in place which exclude a possibility of introducing positive reactivity by two and more design reactivity controls simultaneously, as well as introducing positive reactivity by reactivity controls during nuclear fuel loading (unloading).

2.3.3.14. The reactivity growth rate by reactivity controls shall not exceed 0.07 β_{eff}/s. For CPS rods with worth higher than 0.7 β_{eff} positive reactivity shall be introduced step-by-step, the step worth being not greater than 0.3 β_{eff} (is ensured by engineered measures). The RI design shall indicate the step worth, lag between steps and reactivity growth rate.

2.3.3.15. Before the reactor start-up the emergency protection rods shall be in on-position.

The reactor subcriticality at any point of the fuel cycle after the emergency protection rods have been put in on-position and other CPS rods inserted in the core shall be not less than 0.01 when the core features the maximum effective multiplication factor.

2.3.3.16. A failure of the neutron flux density and(or) density change rate monitoring channel shall be signaled on for the operator and recorded. At this, the signal indicating the failure of such channel shall be generated.

2.3.3.17. The RI design shall contain requirements for the means of ensuring, during operation, the prompt automatic detection and recording of current reactivity margins of the reactor core and their changes. The RI design shall justify a procedure for determining a total worth of reactivity controls, worth of emergency protection rods, worth of CPS rod groups, reactivity coefficients through parameters affecting reactivity (power, coolant temperature, moderator temperature, dissolved absorber concentration, etc.), as well as methodologies and errors of determining these values.

2.3.3.18. The RI design shall provide for means of and methodologies for the reactor core subcriticality monitoring.

2.3.3.19. The RI design shall provide for means of monitoring of power density non-uniformity in the reactor core and means of operative calculation of critical power ratio margins.

2.3.3.20. For the reactor cores where absence of neutron flux density oscillations has not been proved the RI design shall provide for means of monitoring and control over the neutron flux density oscillations and the oscillation control procedure that does not violate the design operational FR damage limits shall be described.

2.4. Normal operation control systems and controlling safety systems

2.4.1. The RI design shall describe and justify requirements for composition, architecture, main characteristics, number and location conditions of NOCS, CSS and their components as well as RI diagnostics systems.

2.4.2. The RI design shall justify and list:

- monitored parameters and signals on RI conditions;
- controlled parameters and controlling signals;
- PP settings and actuation conditions;
- locations of RI diagnostic system sensors;
- parameters determining whether actuation of safety systems is required.

2.4.3. The RI design shall demonstrate that the NOCS and CSS provide monitoring of the technical state and safe operation of RI during normal operation and operational events including design basis accidents.

2.4.4. The RI design shall include and justify lists of protective features and interlocks for RI equipment as well as technical requirements for conditions of their actuation.

2.4.5. NOCS and CSS shall include devices for generating the following signals, as minimum:

- emergency warning (siren with a distinguished sound pitch) – in cases anticipated by the RI design;
- emergency (visible and audible) – when parameters approach the emergency protection actuation settings and conditions;
- warning (visible and audible) – in cases of operational events at RI systems and components and when parameters approach the PP actuation settings and conditions;
- indicating – on availability of voltage in power supply circuits, state of equipment, instruments, etc.

2.4.6. Diagnostics of NOCS and CSS shall be provided for.

2.4.7. CSS and NOCS shall be designed so as to provide a possibility to identify initiating events of accidents, actual operating algorithms of RI safety important systems, deviations from standard algorithms and determine the personnel actions.

2.4.8. To meet the para. 2.4.7 requirement, the recording of the below shall be provided:

* parameters and indications of conditions of RI systems (components) which allow for confident determining of the initiating event;
* controlling signals;
* parameter changes which characterize conditions of RI safety important systems;
* parameters at which the protective features are actuated;
* positions of safety system valves;
* parameters characterizing the radiation situation;
* operating personnel actions including visual information;
* operating personnel conversations over the communication systems.

2.4.9. The RI design shall justify and contain data on the scope and frequency of recording and keeping the information indicated in para. 2.4.8.

2.4.10. The means of recording shall be available and ensure keeping the information in case of design basis and beyond design basis accidents (in a “black-box” type device).

2.4.11. The RI design shall establish:

- permissible reactor power values depending on NOCS performance when its function is partially lost;
- conditions for putting NOCS and CSS and their sections under repair.

2.4.12. For controlled and monitored parameters the ranges and rates of their changes shall be justified for normal operation and operational events including design basis accidents.

2.4.13. NOCS and CSS components shall be subject to metrological testing and certification.

2.4.14. The RI design shall include an analysis of response of NOCS and CSS to external and internal impacts, possible malfunctions and failures (short circuits, insulation degrading, voltage drops and uptakes, false actuation, loss of signals, etc.) and failures of RI major equipment, which demonstrate that there are no responses hazardous for RI. Should NOCS and CSS responses hazardous for RI be revealed during operation, RI shall be shutdown and the measures shall be taken to exclude such responses. The operating organization shall ensure that the corresponding changes are done to the RI design in accordance with the established procedure.

2.4.15. The use of programmable features and codes in NOCS and CSS shall be justified and confirmed by tests. The programmable features and codes shall be verified.

2.4.16. The RI and its systems shall be controlled from MCR and, if necessary, from local control stations.

2.4.17. At each power unit, in addition to MCR, there shall be a back-up control room (BCR) to provide for rendering the reactor subcritical and RI emergency cooldown as well as monitoring of process parameters necessary for RI safety should it be impossible from MCR due to any reasons (fire, etc.).

2.4.18. Requirements to composition of the equipment and hardware of MCR, BCR and local control stations shall be determined in the RI design.

2.4.19. The BCR shall receive and display information on the state of all systems and individual system components including, as minimum:

- neutron flux density in the core;
- parameters of coolant and systems involved in emergency cooldown;
- indicators of intermediate and terminal positions of CPS rods;
- indicators of conditions of reactivity controls (positions of pump valves and components which unambiguously show whether the reactivity controls are available for designated functions and the fact of their actuation, as well as the parameters of the liquid absorber solution (when used) - temperature, pressure, concentration etc.;
- indicators of valve positions and condition of systems which support cooldown.

2.4.20. A possibility of disabling MCR and BCR monitoring and control circuits due to a common cause failure under the considered initiating events shall be excluded; and a possibility of simultaneous control from MCR and BCR of each specific element shall be excluded through the use of engineered means.

2.4.21. The concentrations of liquid absorber as set up in the RI (NPP) design shall be ensured in the reactor, primary circuit, emergency liquid absorber tanks and all systems to be filled with a liquid absorber as per the RI (NPP) design. A technique and frequency of measurements of a neutron-absorbing nuclides concentration in the liquid absorber solution shall be determined in the RI (NPP) design.

2.4.22. Engineered features shall be provided to monitor the content of neutron-absorbing nuclides in the solution of a liquid or gaseous absorber, if used, in RI and emergency liquid absorber tanks during RI operation. Also, engineered features shall be provided to maintain homogeneous concentration of the absorber solution in its hosting tanks.

2.4.23. The engineered means or organizational measures shall provide for incoming inspection of the content of neutron-absorbing nuclides used in the reactivity controls against the design characteristics.

2.4.24. Each emergency liquid absorber tank shall be equipped with at least two channels to monitor the level and(or) measure pressure, with the resulted signal being displayed at MCR and BCR.

2.4.25. The NOCS and CSS shall be provided with reliable power supply during normal operation and operational events including design basis accidents (including the power unit blackout) in the scope justified in the RI design.

2.4.26. The NOCS shall include close-circuit television and means of communications between MCR, BCR and local control stations (telephone, loudspeakers, radio, etc.).

2.4.27. The NOCS and CSS shall include an operator's information support system.

2.4.28. The NOCS and CSS shall include means to transmit information to the Off-Site and On-Site Emergency Control Centers to control NPP during beyond design basis accidents, as required for assessment of the situation and decision-making.

2.4.29. The RI design shall include organizational and(or) technical measures to exclude unauthorized access to NOCS and CSS.

2.5. RI coolant circuit (primary circuit)

2.5.1. The RI design shall define the primary circuit boundaries.

2.5.2. The RI design shall justify operational reliability of the primary systems and components during the design service life considering physical and chemical, thermal, force and other impacts possible during normal operation and operational events including design basis accidents. Number and nature of impacts to be considered to determine the design service life shall be described and justified in the RI design.

2.5.3. The design shall demonstrate that the strength of the reactor pressure vessel is ensured during normal operation and operational events including design basis accidents throughout the NPP power unit service life.

2.5.4. The layout of equipment and geometry of the primary circuit shall provide for natural circulation of coolant in the primary circuit when the forced circulation is lost or unavailable including that during design basis accidents.

2.5.5. The primary circuit pipelines shall be equipped with devices for monitoring and preventing impermissible movements under reactive forces resulted from breaks. The RI design shall justify strength and effectiveness of such devices under design basis accidents.

2.5.6. The heat exchanging equipment used to transfer heat from the RI primary circuit shall have the heat exchanging surface that is sufficient to compensate for a degrading of its heat transfer properties during operation.

2.5.7. When the forced circulation is used, the pumps providing for such circulation, if de-powered or in case of AZ actuation at any reactor power level, shall have sufficient momentum to ensure forced primary coolant flow until the moment when the gravity circulation will ensure removal of residual heat without exceeding the fuel damage operational limits.

2.5.8. The RI design shall provide for the means of:

- automatic protection against impermissible overpressure in the primary circuit during normal operation and operational events including design basis accidents;
- compensating for coolant amount changes due to temperature alterations;
- compensating for coolant losses in case of leaks. Maximum leak to be compensated for by this means is established in the RI design.

2.5.9. The RI design shall provide for leak limiters in the piping running from the main circulation pipeline. A waiver to install leak limiters shall be justified in the RI design.

2.5.10. The primary circuit components shall be equipped with devices reducing seismic impacts. A waiver to equip the primary circuit components with such devices shall be justified in the RI design.

2.5.11. The RI and NPP designs shall establish the coolant quality indicators; its chemical composition and permissible content of radionuclides during operation; the design shall envisage engineered features and organizational measures for their maintaining and monitoring. Technical solutions and organizational measures to ensure the coolant quality, as well as those related to methods and means of its monitoring, shall be justified in the RI and NPP designs.

2.5.12. The RI design shall provide for technical measures to protect the primary circuit against coolant drainage which is not envisaged by the NPP power unit safe operation process regulations. A partial drainage during repair and refueling shall be justified in the RI design.

2.5.13. The RI design shall provide for means and methods of detection of the primary coolant leaks locations and flow rate with accuracy justified in the design.

2.5.14. The technical and organizational measures shall exclude an unanticipated ingress of clean condensate and liquid absorber solution having concentration which is less than determined by the RI (NPP) design into the primary coolant and other systems which, according to the RI (NPP) design, shall be filled with the liquid absorber solution.

2.6. Emergency core cooling systems

2.6.1. The RI and NPP designs shall provide for the emergency core cooling systems. A composition, structure and characteristics of the emergency core cooling systems shall be justified in the RI and NPP designs.

2.6.2. The emergency core cooling systems shall be designed taking into account independence and redundancy principles and be capable of performing their function of preventing violation of the fuel damage design limits in case of design basis accidents.

2.6.3. A list of parameters, settings and conditions of emergency cooling systems' actuation shall be justified in the RI (NPP) design on the basis of an analysis of design basis accidents.

2.6.4. Acceptability of and conditions for disabling one channel of the emergency core cooling system shall be justified in the RI (NPP) design.

2.6.5. The RI (NPP) design shall consider all possible impacts to the systems (components) related to actuation and operation of the emergency core cooling systems.

2.6.6. The RI (NPP) design shall specify technical and organizational measures targeted to prevent unauthorized access to the emergency core cooling systems.

2.6.7. The RI (NPP) design shall contain a justification of reliability indicators of the emergency core cooling systems.

2.6.8. When the reactor is subcritical, the actuation and operation of the emergency core cooling systems shall not render it otherwise.

2.6.9. The emergency core cooling systems shall provide for cooling down and long-term maintaining the reactor core at the coolant parameter values justified in the RI (NPP) design.

2.7. Refueling equipment and core refueling procedure

2.7.1. Refueling equipment

2.7.1.1. The RI design shall justify and describe the refueling equipment and the requirements they are subject to, fulfillment of which ensures safe handling of FAs and other core components during refueling and also in case of failures and damages to the refueling equipment.

2.7.1.2. The FAs being reloaded shall be provided with heat removal without exceeding the fuel rod temperature parameters set forth by the RI design for the refueling operations being conducted in normal operation and failures.

2.7.1.3. The refueling equipment shall be designed so that in their normal operation and failures the normal operation conditions of RI and at-reactor fuel storages are not violated.

2.7.1.4. The RI and NPP designs shall specify the requirements for assembling, operation, maintenance, repair, tests and periodic inspections of the refueling equipment and also requirements to its reliability.

2.7.1.5. The refueling equipment shall be designed (engineered) to be accessible for inspection, repair, tests and maintenance.

2.7.1.6. When designing the refueling equipment the measures shall be provided to prevent damages to, deformations, destruction or drop of FAs and other core components and to prevent impermissible forces to them during their withdrawal or insertion. Values of maximum permissible forces shall be specified in the RI design. The use of refueling equipment which is not foreseen by the design is prohibited.

2.7.1.7. When designing the refueling equipment it shall be provided that a termination of power supply does not result in FA or other reloaded core component drop.

2.7.1.8. The RI design shall justify and set forth permissible speeds of fuel assemblies and other core components’ movements by the refueling equipment.

2.7.1.9. There shall be engineered features (interlocks etc.) ensuring that the refueling equipment moves within the permissible boundaries.

2.7.1.10. The RI design shall foresee the equipment for reliable transfer of FAs and other components of the core to safe locations in case of a failure or violation of operating conditions.

2.7.1.11. The refueling equipment shall feature boards (panels) with indicators to present information on a position (condition) and orientation of FAs, other components of the core being reloaded, and grips.

2.7.1.12. A possibility for movements of the refueling equipment at the moment of its connection with the process channel or during insertion of FAs or other components of the core being reloaded (withdrawn from the core) shall be excluded.

2.7.1.13. Interlocks shall be provided to prevent movements of the refueling equipment while FAs and other reloaded components of the core are in the positions which are not foreseen by the design.

2.7.1.14. A closed-circuit television system to monitor the refueling shall be provided. The RI and NPP designs shall define a list of refueling operations subject to the closed-circuit television monitoring.

2.7.2. Refueling sequence

2.7.2.1. The RI design shall justify:

- refueling methods;
- refueling frequency, scope and procedures;
- engineered means and organizational measures targeted to ensure nuclear safety during refueling, including monitoring of the neutron flux density;
- working concentration of the liquid absorber (if used), sampling locations, concentration monitoring and maintaining means.

2.7.2.2. In addition to failures of refueling system equipment, the RI and NPP design, as well as SAR NPP, shall consider possible loading (refueling) errors and their consequences, and measures aimed at errors elimination shall be developed.

2.7.2.3. The core refueling procedure shall be defined by a refueling program and(or) manual, work schedule and refueling map developed by the NPP personnel, approved by the NPP administration and agreed upon in accordance with the established procedure.

2.7.2.4. During refueling and repair operations, the organizational measures and, as possible, engineered means shall prevent alien objects from getting into the inner space of the equipment, valves and pipelines of the reactor installation.

2.7.2.5. For the reactors where refueling is carried out when the CPS rods are disengaged, the refueling shall be carried out having the CPS rods and other reactivity controls inserted into the core. At that, the reactor minimum subcriticality during the refueling shall be not less that 0.02, considering possible errors.

2.7.2.6. For the reactors where refueling is carried out having the CPS rods disengaged and reactivity is compensated for by a liquid absorber, the refueling shall be carried out when the CPS rods and other reactivity controls are inserted into the core. A liquid absorber concentration shall be brought up to the value which would ensure (taking into account possible errors) the reactor subcriticality of not less than 0.02 (without taking into account the inserted CPS rods).

2.7.2.7. For the reactors where the required subcriticality during refueling is ensured by a liquid absorber, the engineered means and organizational measures shall be provided to prevent the clean condensate supply to the reactor and primary circuit during refueling.

2.7.2.8. For the pressure vessel reactors with the upper location of CPS rod drives, the reactor and CPS actuators' design shall ensure that the CPS rods are disengaged when the upper part of the reactor is removed; at that, the means of diagnostics shall record the disengaged state.

2.7.2.9. The RI design shall include engineered measures that exclude the "uprising" of CPS rods during refueling.

2.7.2.10. Reloading of FAs and other core components at the shutdown channel-type reactor shall be carried out with AZ rods put in on-position. At this, the reactor minimum subcriticality during refueling shall be not less that 0.02, considering possible errors.

2.7.2.11. As regards the reactor installations where power refueling takes place, the permissible operational modes (power, coolant flow rate, etc.) during refueling shall be justified and defined in the RI design. Also, worth of means used to suppress excess reactivity, which can be introduced due to fueling errors or reactivity effects, shall be justified.

2.7.2.12. During power refueling the primary circuit integrity shall be intact, and the means shall be provided to check that there are no leaks of the primary coolant.

2.7.2.13. As regards the reactors with partial refueling, after refueling the tests (measurements) shall be conducted to verify the main design and calculated neutronics characteristics of the core. For the reactors with on-line refueling the frequency of tests (measurements) shall be justified in the RI design.

During the tests the checks shall be done to verify whether the experimental measurement results agree with the calculated parameters; it shall be done against the criteria established in the RI design.

3. Nuclear safety ensurance during the NPP unit commissioning

3.1. Reactor first criticality

3.1.1. During the first criticality process the experimental data on the reactor neutronics, reactivity effects, worth of control rods and AZ etc. shall be obtained.

3.1.2. The first criticality of the reactor, including its loading with nuclear fuel, shall be carried out in accordance with the First Criticality Program. The First Criticality Program shall be developed and approved by the operating organization.

3.1.3. The First Criticality Program shall include:

- a list of systems and equipment needed for the first criticality in the reactor;
- a procedure for loading the reactor with FAs (FRs);
- a procedure of reaching criticality;
- a description of tests (measurements) and their conduct;
- anticipated values of critical charges, critical positions (conditions) of reactivity controls, their worth, assessments of impacts to reactivity produced by loaded FAs (FRs) and coolant;
- test and measurement techniques;
- nuclear safety measures during the first criticality.

3.1.4. Readiness for the first criticality is checked on by:

- a Working Commission designated by the operating organization;
- a Commission designated by a state nuclear and radiation safety authority.

3.1.5. The Working Commission shall check:

- whether the work performed complies with the RI and NPP designs;
- equipment operability, availability of equipment test records, the pre-startup and alignment operations completion records;
- availability and formats of the operating documentation;
- availability of work permits for the shift personnel; and availability of records demonstrating that the operators-physicists have passed the exams.

The Working Commission shall generate a record on the systems and equipment readiness and personnel preparedness for the first criticality. The record shall be approved by the operating organization in accordance with the established procedure.

3.1.6. The Commission designated by a state nuclear and radiation safety authority shall check on:

- technical preparedness of the NPP power unit for the first criticality;
- design and operating documentation;
- personnel training adequacy for the first criticality.

3.1.7. The first fuel delivery to the site of the NPP power unit under commissioning shall be carried out subject to availability of the NPP operation license issued by a state nuclear and radiation safety authority and as determined by results of the inspection carried out by a state nuclear and radiation safety authority to check on the NPP power unit readiness for the nuclear fuel delivery.

3.1.8. A decision to achieve the first criticality shall be made in accordance with the established procedure on the basis of the Working Commission’s record stating that the systems and equipment are ready and that the personnel are prepared for the first criticality, and on the basis of the report from the operating organization stating that it has eliminated the deficiencies following the results of a state nuclear and radiation safety authority’s inspection related to the NPP unit readiness for the first criticality.

3.1.9. Should an abnormal situation arise in the course of the tests (measurements) during the first criticality, the tests shall be terminated and the reactor shall be rendered subcritical.

3.1.10. Results of the core loading with FAs (FRs) and also results of tests conducted during the first criticality shall be documented in the records and reports to be submitted to a state nuclear and radiation safety authority in accordance with the established procedure.

3.2. NPP first power

3.2.1. The NPP power unit first power involves a step-by-step and gradual power build-up, determining and updating the NPP power unit and RI parameters, integrated testing of NPP power unit systems and equipment, performance of planned tests (measurements) at each stage and analysis of obtained results.

3.2.2. The first power of the NPP unit is conducted in accordance with the NPP Power Unit First Power Program which is revised, if necessary, on the basis of the first criticality results. The First Power Program shall be developed and approved by the operating organization.

3.2.3. The First Power Program shall include the implementation procedure, anticipated values of the reactor physical characteristics (effects of reactivity, etc.), RI thermal engineering characteristics, test methodologies, nuclear safety measures during the first power and the like.

3.2.4. The First Power Program shall provide for tests and probation of the NPP power unit operational modes, inspection of the safety systems in a scope and sequence which provide for gaining the nominal power level by the reactor in a safe manner including testing of safe and dynamically stable operation under transients at all stages of power gaining process.

3.2.5. The Working Commission shall check whether the NPP power unit is ready for the first power. The Working Commission shall check on the readiness of the NPP power unit systems and equipment for the first power, building up reactor power, turbine generator start-up and NPP power unit connection to the grid, manning level as regards the shift personnel, their preparedness and availability of work permits. The Commission shall generate a record on the NPP power unit readiness for the first power. The record shall be approved by the operating organization in accordance with the established procedure.

If necessary, a state nuclear and radiation safety authority designates a commission to check on whether the NPP power unit is ready for the first power.

3.2.6. The first power of the NPP power unit shall be carried out after the deficiencies outlined in the Working Commission’s record and a state nuclear and radiation safety authority’s (if there was a check by a state nuclear and radiation safety authority) record have been eliminated.

3.2.7. A decision to conduct the first power operation shall be made in accordance with the established procedure on the basis of the Working Commission’s record on the NPP power unit readiness for the first power and the operating organization’s report on elimination of the deficiencies following the results of a state nuclear and radiation safety authority’s inspection (if conducted) related to the NPP unit readiness for the first power.

3.2.8. Basing on the results of the first criticality and first power the operating organization shall issue a report and update SAR NPP, if necessary.

4. Nuclear safety during operation

4.1. The Process Regulations for Safe Operation of NPP Power Unit are the main document which provides for safe operation of the NPP power unit. They contain safe operation rules and techniques, general procedure for safety related operations as well as safe operation limits and conditions. The operating organization shall develop the Process Regulations for Safe Operation of NPP Power Unit.

4.2. The NPP power unit operation shall be carried out in accordance with the operating manuals developed by the NPP administration on the basis of the design and engineering documentation and Process Regulations for Safe Operation of NPP Power Unit as updated basing on the NPP commissioning results and taking into account the operating experience.

4.3. The operating organization shall formalize a Reactor Installation Certificate before commencing the operation.

4.4. The operating organization, on the basis of the RI and NPP designs and considering the requirements of the process regulations for safe operation of the NPP power unit, shall arrange for the development and issue of the following documents for the safety important systems:

- inspection and test manuals;
- schedules for maintenance, preventive maintenance and major overhauls of the systems and components;
- test schedules and schedules for safety systems’ performance tests.

4.5. State of the RI and its systems, and conditions under which the NPP operation is permitted shall be justified in the RI and NPP designs and specified in the Process Regulations for Safe Operation of NPP Power Unit.

4.6. Should the operational limits be violated by the operating personnel a certain sequence of actions, which is set forth in the RI (NPP) design and Process Regulations for Safe Operation of NPP Power Unit and aimed at bringing the NPP power unit to normal operation, shall be implemented. If normal operation cannot be restored, the NPP power unit shall be shutdown.

4.7. In case of an abnormal situation (accident), the NPP power unit shall be shutdown. Its causes shall be identified and eliminated and measures targeted to restore normal operation of the NPP power unit shall be taken. Operation of the NPP power unit can be continued only after the causes of the abnormal situation (accident) have been eliminated.

4.8. The operating organization shall conduct investigations of NPP occurrences and accidents in accordance with the federal standards and rules. It shall also submit the information about these events in accordance with the procedure established in the federal standards and rules.

4.9. In case of design basis accidents the personnel actions shall be defined by the NPP Power Unit Accident Elimination Manual, which is to be developed on the basis of SAR NPP by the operating organization. The Manual shall address design basis accidents and identify measures targeted to eliminate the accident consequences.

4.10. To manage beyond design basis accidents as per the RI and NPP designs and SAR NPP, the operating organization shall develop a guide on beyond design accident management.

4.11. The NPP Power Unit Accident Elimination Manual and Beyond Design Accident Management Guide shall specify the procedure for putting into effect the action plans for protection of the personnel and population in case of a beyond design basis accident.

4.12. Emergency drills shall be conducted for the NPP personnel to have them prepared to undertake actions in case of abnormal situations and accidents. Frequency and conduct procedure for these drills shall be approved by the operating organization.

4.13. It is strictly prohibited to open up the instrumentation and controls, change settings of the emergency and warning alarms and protection systems commencing the accident initiation until the Accident Investigation Commission starts working. The engineered means and organizational measures shall be provided for to prevent a loss of the recorded information and unauthorized access to devices and components, databases and archives of the control system which contain records of the equipment conditions before the accident initiation and during the subsequent period.

4.14. Safe operating conditions of the shutdown reactor with the nuclear fuel in the core, including loading and refueling modes, shall be justified in the RI design and described in the Process Regulations for Safe Operation of NPP Power Unit. As minimum, the following shall be identified for these modes:

- the scope of monitoring in accordance with the requirements of paras. 2.3.3.1, 2.3.3.3 and 2.3.3.6 of these Rules, including mandatory monitoring of the neutron flux density and liquid absorber concentration, if used in this RI type;
- availability requirements for the safety important systems.

4.15. For the reactors where nuclear fuel loading and refueling are carried out with the reactor, primary circuit and related systems filled with a liquid absorber, as well as during tests and repairs of the primary circuit equipment, valves and pipelines, the liquid absorber concentration shall be not lower than that set forth by the RI (NPP) design.

4.16. The operating organization, on the basis of the design documentation and operating experience shall develop a basic list of nuclear hazardous operations at the NPP power unit.

4.17. Operations at the safety important systems (components) associated with shutdown for repair and restart, as well as tests which are not foreseen by the Process Regulations for Safe Operation of NPP Power Unit and operating manuals, shall be considered nuclear hazardous operations.

4.18. Nuclear hazardous operations shall be conducted in accordance with a special Work Program approved by the NPP administration.

Nuclear hazardous operations which are not foreseen by the by the Process Regulations for Safe Operation of NPP Power Unit and operating manuals shall be conducted in accordance with a special Work Program approved by the operating organization and coordinated with RI and NPP designer-organizations.

The Work Program shall include:

- the nuclear hazardous operation objective;

- a list of the nuclear hazardous operations;
- technical and organizational measures to ensure nuclear and radiation safety;
- criteria and control over proper completion of the nuclear hazardous operations;
- a directive on designation of an individual responsible for conduct of the nuclear hazardous operations.

Generally, nuclear hazardous operations shall be conducted at the shutdown reactor.

4.19. While performing nuclear hazardous operation activities the shutdown reactor subcriticality shall be not less than 0.02 for the state of the reactor with the maximum reactivity margin (for the channel-type reactor AZ (EP) rods shall be in the on-position and the other CPS rods inserted into the core).

4.20. After the repair of the safety important equipment and systems is completed, a check of compliance of these systems' characteristics with the design characteristics shall be carried out. The check shall be carried out in accordance with the existing manuals or programs developed under the procedure established by the operating organization.

4.21. During any tests of safety important systems the check of compliance of the test results with the criteria set forth in the RI and NPP designs shall be carried out. The test results shall be documented in a record.

5. The Rules compliance control

5.1. The operating organization shall continuously control the compliance with the requirements of the Rules.

5.2. The operating organization shall arrange for periodic (at least once in two years) inspections related to control over the NPP's compliance with the requirements of the Rules and establish a procedure for conduct of NPP nuclear safety inspections by in-house commissions. Results obtained through inspections by the NPP operating organization shall be submitted to a state nuclear and radiation safety authority.

APPENDIX. Fuel Rod Damage Limits and Requirements for Russian NPPs with the Most Common RI Types

1. NPPs with VVERs

1.1. Operational limit for fuel damage:
- gas leakage defects – not more than 0.2% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.02 % of the fuel rods in the core.

1.2. Safe operation limit for fuel damage:
- gas leakage defects – not more than 1% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.1 % of the fuel rods in the core.

1.3. Maximum design fuel rod damage limit corresponds to the conditions where the following limiting parameters are not exceeded:
- FR cladding temperature shall be not more than 1200°C;
- equivalent depth of FR cladding oxidizing shall not exceed the limiting value established in the design on the basis of experimental data;
- a fraction of reacted zirconium in the core shall be not more than 1% of its mass in the FR cladding;
- maximum fuel temperature shall be less than the melting temperature.

1.4. Reactivity coefficient values in terms of the coolant specific volume and fuel temperature, total reactivity coefficient in terms of the coolant temperature and fuel temperature, as well as reactor power, shall not be positive in all critical states possible over the entire range of the reactor parameter changes during normal operation and operational events including design basis accidents.

2. NPP with RBMKs

2.1. Operational limit for fuel damage:
- gas leakage defects – not more than 0.2% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.02 % of the fuel rods in the core.

2.2. Safe operation limit for fuel damage:
- gas leakage defects – not more than 1% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.1 % of the fuel rods in the core.

2.3. Maximum fuel rod damage limit corresponds to the conditions where the following limiting parameters are not exceeded:
- FR cladding temperature shall be not more than 1200°C;
- equivalent depth of FR cladding oxidizing shall be not more than the limiting value set forth in the design basing on the experimental data;
- a fraction of reacted zirconium in the core shall be not more than 1% of its mass in the cladding;

- maximum fuel temperature shall be less than the melting temperature.

2.4. Reactivity coefficient values in terms of the fuel temperature and power shall not be positive over the entire range of the reactor parameter changes during normal operation and operational events including design basis accidents. The RI design shall justify the permissible range of safe values of steam coefficient of reactivity. One has to pursue to have the steam coefficient of reactivity values close to zero during normal operation and operational events, including design basis accidents. While operating the NPP the steam coefficient of reactivity shall be confirmed by measurements done in accordance with verified methodologies and with frequency established in the RI design.

3. NPP with the BNs

3.1. Operational limit for fuel damage:

- gas leakage defects – not more than 0.05% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.005 % of the fuel rods in the core.

3.2. Safe operation limit for fuel damage:

- gas leakage defects – not more than 0.1% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.01 % of the fuel rods in the core.

3.3. Maximum design fuel rod damage limit for fast sodium-cooled reactors with MOX-fuel and fuel cladding made of austenitic steel ChS-68KhDcorresponds to the situation where the following limiting parameters are not exceeded:

- fuel rod cladding temperature – 900^{o}C;
- fuel temperature – 2300^{o}C;
- volume swelling of FR cladding – 15%.

3.4. Reactivity coefficient values in terms of the fuel temperature and power of the reactor as well as a total coefficient of reactivity in terms of the coolant and fuel temperature shall be negative over the entire range of the reactor parameter changes during normal operation and operational events including design basis accidents. For beyond design basis accidents, the permissible range of the void coefficient shall be justified in the RI and NPP designs.

4. NPP with the NDHPs

4.1. Operational limit for fuel damage:

- gas leakage defects – not more than 0.2% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.02 % of the fuel rods in the core.

4.2. Safe operation limit for fuel damage:

- gas leakage defects – not more than 1% of the fuel rods in the core;
- direct contact of the nuclear fuel with the coolant – not more than 0.1 % of the fuel rods in the core.

4.3. Maximum design fuel rod damage limit corresponds to the conditions where the following limiting parameters are not exceeded:

- FR cladding temperature shall not be more than 1200^{o}C;
- equivalent depth of FR cladding oxidizing shall be not more than the limiting value established in the design basing on the experimental data;
- a fraction of reacted zirconium in the core shall be not more than 1% of its mass in the cladding;
- maximum fuel temperature shall be less than the melting temperature.

4.4. Reactivity coefficient values in terms of the coolant specific volume, fuel temperature, reactor power, a total reactivity coefficient in terms of the coolant temperature and fuel temperature shall not be positive in all critical states possible over the entire range of the reactor parameter changes during normal operation and operational events including design basis accidents.

5. NPP with EGP-6s

5.1. Operational limit for fuel damage (tube fuel rods with a fuel composition as uranium dioxide grits within a magnesium matrix:

- temperatures of the FR outer surface - 430^{o}C;
- clad non-integrities are not allowed.

5.2. Safe operation limit for fuel damage:

- loss of integrity of the outer cladding of at least one fuel rod;
- achievement of 50-fold exceedence of FFLS readings over the background for any fuel assembly in the reactor.

5.3. Maximum design fuel rod damage limit:

- cladding temperature of the fuel rod relived of internal pressure – 1100^{o}C;
- cladding temperature of the fuel rod under operating internal pressure – 930^{o}C;
- local depth of interaction between the FR outer cladding and matrix material is not more than 85%.

5.4. Reactivity coefficient values in terms of fuel temperature, void content in the coolant and power shall not be positive over the entire range of the reactor parameter changes during normal operation and operational events including design basis accidents.