Technical Code of Common Practice

TCP 171-2009 (02300)

Nuclear Safety Regulations for Reactor Facilities of Nuclear Power Plants (RNS RF NPP)

ПРАВІЛЫ ЯДЗЕРНАЙ БЯСПЕКІ
РЭАКТАРНЫХ УСТАНОВАК АТАМНЫХ СТАНЦЫЙ
(ПБЯ РУ АС)

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Foreword

Objectives, main principles, provisions on the government regulation and administration in the area of technical codification are established by the Law of the Republic of Belarus ‘On Technical Codification’.

1 ELABORATED by the State Scientific Institution ‘The Joint Institute for Power and Nuclear Research – Sosny’.

INTRODUCED by the Ministry of Energy of the Republic of Belarus

2 APPROVED by resolution of the Ministry of Emergency Situations of the Republic of Belarus of 17 February 2009 No. 14

3 FIRST EDITION (with abrogation of PNAE-G-1-024-90 ‘Nuclear Safety Regulations for Reactor Facilities of Nuclear Power Plants (RNS RF NPP – 89)’ approved by edict of the USSR *GosAtomNadzor* of 12 June 1990 No 14 and of NSRa-04-74 ‘Nuclear Safety Regulations for Nuclear Power Plants’ approved by the USSR *GosAtomNadzor* of 31 December 1974)

Published in Russian

Content

Introduction

Safe development of the nuclear power industry relies on:

– the location conditions of the nuclear power engineering facility;

– the choice of design concept;

– performance quality of the technological process;

– qualification of the service personnel;

– the regulatory authorities qualification;

– systematic supervision and control over observance of the nuclear safety requirements.

This technical code of common practice was developed in accordance with:

–regulation of the Council of Ministers of the Republic of Belarus ‘On the Approval of the List of the State Scientific and Technical Programs for 2006 – 2010’ of 4 January 2006 No 5;

–Government scientific and technical program ‘Nuclear Physics Technology for National Economy of Belarus’ approved by order of Chairman of the Government Scientific and Technical Committee of 6 July 2006 No 180;

–Plan of essential preparatory work to be fulfilled prior to construction of the nuclear power plant of the Republic of Belarus approved by decree of the Council of Ministers of the Republic of Belarus of 18 July 2006 No 905-9.

While developing this technical code of common practice, the experience of building and operation of the nuclear power industry facilities collected in the former USSR and the Russian Federation generalized in:

– NSRa-04-74 Nuclear Safety Regulations for Nuclear Power Plants approved by resolution of the USSR *GosAtomNadzor* of 31 December 1974;

– PNAE-G-01-011-97 General Provisions for Safety of Nuclear Power Plants (OPB-88/97, NP-001-97) approved by resolution of the *GosAtomNadzor* of Russia of 14 November 1997 No 9;

– PNAE-G-01-024-90 Nuclear Safety Regulations for Reactor Facilities of Nuclear Power Plants (NSRa RU AS – 89) approved by resolution of the USSR *GosAtomNadzor* of 12 June 1990 No 7;

– NP-082-07 Nuclear Safety Regulations for Reactor Facilities of Nuclear Power Plants approved by resolution of the Federal Agency for Environmental, Technological and Nuclear Inspection of 10 December 2007 No 4.

This technical code of common practice was developed in accordance with legislation of the Republic of Belarus:

– Law of the Republic of Belarus ‘On Protection of Environment’ of 26 November 1992 No 1982-XII (read with Law of the Republic of Belarus of 17 July 2002 No 126-3);

– Law of the Republic of Belarus ‘On Fire Safety’ of 15 June 1993 No 2403-XII;

– Law of the Republic of Belarus ‘On Sanitary and Epidemiological Welfare of the Population’ of 23 November 1993 No 2583-XII (read with Law of the Republic of Belarus of 23 May 2000 No 397-3);

– Law of the Republic of Belarus ‘On Specially Protected Natural Areas and Objects’ of 20 October 1994 No 3335-XII (read with Law of the Republic of Belarus of 23 May 2000 No 396-3);

– Law of the Republic of Belarus ‘On the Radiation Safety of the Population’ of 5 January 1998 No 122-3;

– Law of the Republic of Belarus ‘On Protection of Population and Territories against Natural and Technogenic Emergency Situations’ of 5 May 1998 No 141-3;

– Law of the Republic of Belarus ‘On Industrial Safety of Hazardous Facilities’ of 10 January 2000 No 363-3;

– Law of the Republic of Belarus ‘On the Use of Nuclear Power’ of 30 July 2008 No 426-3.

This technical code of common practice was developed consistent with IAEA recommendations stated in:

– code of regulations on safety of nuclear power plants ‘NPP Safety: Designing’ 50-C-D (Rev. 1), Vienna, 1990;

– code of regulations on safety of nuclear power plants ‘NPP Safety: Operation’ 50-C-O (Rev. 1), Vienna, 1990;

– IAEA instructions on design safety for the NPPs of 50-SG-D1 series – 50-SG-D14 (1981 – 1993);

– IAEA instructions on operating safety for the NPPs of 50-SG-O1 series – 50-SG-O12 (1980 – 1997);

– safety requirements NS-R-1 'NPP Safety: Designing’. Vienna, 2003;

– safety requirements NS-R-2 ‘NPP Safety: Operation’. Vienna, 2003;

– other IAEA documents (75-INSAG-3 Rev. 1, INSAG-12, INSAG-10).

TECHNICAL CODE OF COMMON PRACTICE

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ПРАВІЛЫ ЯДЗЕРНАЙ БЯСПЕКІ РЭАКТАРНЫХ УСТАНОВАК АТАМНЫХ СТАНЦЫЙ (ПБЯ РУ АС)

Nuclear safety regulations for reactor facilities of nuclear power plants (RSN RF NPP)

Effective date 2009-05-01

1 Field of application

This technical code of common practice (hereinafter – the technical code) specifies requirements for design, characteristics and operational conditions of the systems and elements of reactor facilities (hereinafter – RF) of nuclear power plants (hereinafter – NPP) as well as organizational requirements directed at providing nuclear safety during design, engineering, building and operation of RF NPP.

Requirements of this technical code cover all RF NPP of the Republic of Belarus being designed, engineered, built and operated.

Requirements of this technical code are compulsory for all departments, enterprises and organizations during design, building and operation of RF and NPP as well as during engineering and fabrication of elements for RF and NPP.

2 Terms and definitions

This technical code uses the following terms with corresponding definitions:

2.1 accident: Breach of normal operation of the nuclear power plant with escapement of radioactive material and (or) ionization radiation outside the normal operation borders designated by the NPP project in amount exceeding the safe operation established tolerance. An accident is characterized by an initial event, ways of passing, and after-effects.

2.2 emergency protection: Safety function that consists in quick transition of the reactor to subcritical state and sustaining it in subcritical state; a complex of safety systems performing function of emergency protection.

2.3 administration: Officials authorized with powers and duties who as well bear responsibility for operating NPP.

2.4 reactor core: Part of the reactor containing fission material, moderator, poison, coolant, instruments affecting reactivity and constructional elements designed for realization of controlled nuclear chain reaction and transfer of energy to the coolant.

2.5 nuclear power plant: Nuclear facility for production of electric and heat power in standard regimes and conditions of application, located within the limits of designated area where, for implementing this purpose, a nuclear reactor (reactors) is employed along with a complex of necessary for its functioning systems, devices, equipment and constructions

2.6 Nuclear plant unit: Part of nuclear power plant represented by a nuclear reactor with generating and other equipment that provides functions of the nuclear power plant in the scope determined by the project.

2.7 commissioning: Process in the course of which they bring systems and components of the constructed nuclear power plant to workable condition and estimate their compliance to the nuclear plant project.

2.8 inner self-protectiveness: Property to provide safety based on the natural back coupling and processes.

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2.9 decomissioning: Process directed at termination of the nuclear power plant further application for its intended purpose, in which course they ensure safety of workers (personnel) of the operating organization, citizens and environment.

2.10group of working elements of the control and protection systems: One or several working elements of the control and protection system conjoined in operation for synchronous cooperative movement and effect on the reactivity.

2.11diagnostics: Control function intended to ascertain operability (nonoperability) status or serviceability (failure) of the object being diagnosed.

2.12 **beyond design basis accident:** Accident caused by initial events left out of account for design basis accident or accompanied by additional as compared to design basis accident failures of the safety system over single failure, realization of erroneous decision of the personnel.

2.13 extraction of the instruments affecting reactivity: Such movement or change of state in condition of the instruments affecting reactivity that leads to introduction of positive reactivity (bringing in the instruments affecting the reactivity leads to introduction of negative reactivity).

2.14actuating device of the control and protection system: Device consisting of a drive, working elements and connections intended for changing the reactor reactivity.

2.15 control channel: The aggregate of sensors, communication circuits, processing means for representation of the signals and (or) parameters for providing monitoring in the scope determined by the project.

2.16 emergency protection equipment set: Control and emergency protection system instrumentation fulfilling function of monitoring and control in the scope determined by the RF project.

2.17 reactor coolant circuit (heat transport main circuit): Contour along with the pressurizer system for circulating coolant through the reactor core in specified by the project modes and operational conditions.

2.18 local criticality: Criticality reached in part of the reactor core, fission material storage or some other volume containing fissile nuclear materials.

2.19 maximum reactivity inventory: Reactivity that can realize in the reactor with removing out of the reactor core all the instruments affecting reactivity and removable absorbers for the company moment and the reactor condition with maximum value of effective multiplication factor.

2.20 maximum design tolerance of fuel-element failure: Admissible values of parameters and characteristics of the fuel-elements under conditions of design basic accident. Their excess may lead to the fuel-element destruction.

2.21 reactor facility certificate: Document that contains information on design, calculated and actual values of physical parameters and characteristics of the mechanical system of the reactor control and protection system for the current fuel cycle, principal characteristics of the reactor facility shut down, equipment for emergency protection and neutron flux monitoring.

2.22 reactor core refueling (refueling): Reactor facility nuclear-hazardous jobs on refueling, extraction and transposition of the fuel assembly (fuel elements), the instruments affecting reactivity and other reactivity affecting elements, with a view of their repair, substitution and disassembling.

2.23 fuel elements damage: Breach of at least one of the damage design tolerance specified for fuel elements.

2.24 subcritical state: Reactor core condition characterized by the value of effective multiplication factor less than one; absence of local criticality.

2.25 prelaunch adjustment and alignment: Nuclear plant commissioning phase by which the nuclear plant systems and elements completed by construction and assemblage are brought to

the state of operational availability with their conformance testing to the criteria and characteristics determined by the project. This phase concludes with NPP readiness for the reactor first criticality.

2.26 alarm system: Function performed by controlling system of the nuclear plant unit normal operation for preventing scram and (or) breach of safe operation limits and conditions.

2.27 drive of the control and protection systems: Device for changing position of the mechanical working element of the control and protection system and holding it in fixed position.

2.28 single failure principle: Principle according to which a system shall fulfill specific functions with any demanding its work initial event and with independent from the initial event failure of one of the active or passive elements possessing moving mechanical parts.

2.29 principle of independence: Principle of reliability enhancement of a system by means of applying functional and (or) physical channeling (elements) where the failure of one channel (element) does not lead to the failure of the other channel (element).

2.30 diversity principle: Principle of reliability enhancement of a system by means of applying in different systems (or within the limits of one system in different channels) different means and (or) similar means based on different operating principles for realization of the specified function.

2.31 backing-up principle: Principle of reliability enhancement of a system by means of applying structural, functional, information and time redundancy in reference to minimally necessary and sufficient for the system scope for fulfilling the specified functions.

2.32 emergency protection actuating device: Means affecting reactivity employed

in the emergency protection.

2.33 working element of the control and protection system: Means affecting reactivity employed

in the control and protection system.

2.34 fuel element depressurization: Breakage of fuel element with breach of the fuel element casing integrity like gas leakiness or direct contact of fission material with the coolant.

2.35 project developers of nuclear power plant (reactor facility): Organizations developing the project of nuclear power plant (reactor facility) and providing its scientific maintenance (prime contractor, general designer, scientific adviser).

2.36 fuel element destruction: Breach of structural integrity of fuel element resulting in the fuel element losing geometry that provides its design cooling.

2.37 reactor facility: Complex of nuclear plant systems and components meant to transform nuclear energy into heat energy. It includes a reactor and systems directly connected to it and indispensable for its normal operation, emergency cooling, emergency protection and upkeep in safe condition providing that other systems of the nuclear power plant fulfill required accessorial and supporting functions. Boundaries of the reactor facility are determined for each nuclear power plant in the project.

2.38 emergency protection system: Signal formed in the emergency protection equipment set with design to initiate actuation of the emergency protection working elements and feeding into registration tools as well as into a unitized control center and a standby control center for warning the personnel.

2.39 alarm signal: Signal formed and registered by monitoring and control systems for initiating the function of alarm system and warning the personnel of possible breaches of normal operation.

2.40 reactor shutdown system: System for transition of the reactor into subcritical state and sustaining it in subcritical state via the instruments affecting reactivity.

2.41 control and protection system: Aggregate of means of technical, software and information support for providing safe flow of fission chain reaction.

Control and protection system is a system essential for safety combining functions of normal operation and safety. It consists of elements of the controlling systems of normal operation, protecting, controlling and supporting systems of safety.

2.42 instruments affecting reactivity: Technological tools realized in form of solid, liquid or gaseous absorbers (moderators, reflectors) by whose repositioning or change of state in the reactor core or reflector they provide the reactor core reactivity change.

2.43 fuel assembly: Mechanical-engineering product containing nuclear materials and meant to obtain heat energy in nuclear reactor by means of realization of controlled nuclear reaction.

2.44 fuel element: Individual assembly module containing fission materials and meant to obtain heat energy in the nuclear reactor by means of realizing controlled nuclear reaction and (or) to accumulate nuclides.

2.45 reactor core severe damage: beyond design basis accident with damage of the fuel elements harder than maximum design tolerance by which maximum permissible emergency value can be exceeded of radioactivity environment discharge.

2.46 control and protection system working element position indicator: Device for finding position of the control and protection system working element in the reactor core.

2.47 first criticality: Stage of NPP commissioning that includes core fueling with fission material, attaining the reactor core criticality and performing indispensable physical measurements at the capacity level by which heat abstraction from the reactor realizes on account of natural heat loss (dissipation).

2.48 equivalent oxidation degree of the enclosure: Related to the initial enclosure thickness, cumulative thickness of equivalent level that would have reacted with water vaper assuming that all of locally resorbed oxygen had gone to production of stoichiometric zirconium dioxide ZrO_2. In case of depressurization of the enclosure, they account for oxidation of both outward and inward surfaces of the enclosure.

2.49 nuclear plant operating organization: Organization that uses their own resources or involves other entities for realizing activities on location, construction, commissioning, operation, limiting performance characteristics, service life prolongation and decomissioning of the nuclear facility and (or) nuclear storage as well as activities on management of nuclear materials, spent fuel and (or) operational radioactive waste.

2.50 nuclear plant energy launch: NPP unit commissioning stage from completion of first criticality to beginning of electric energy production.

2.51 nuclear accident: Accident associated with damage of fuel elements exceeding the limits

of determined safe operation and (or) personnel exposure exceeding permitted limits caused by:

– breach of management and control of fission chain reaction in the reactor core;
– criticality rise during refueling, transport and storage of fuel elements;

– breach of heat sink from the fuel elements;
– other causes leading to damage of fuel elements.

2.52 nuclear safety: State of protection for citizens and environment against adverse effect of ionizing radiation of the nuclear facility and (or) nuclear storage provided by attaining appropriate conditions of their operation as well as appropriate handling nuclear materials, spent nuclear materials and (or) operational radioactive wastes.

3 Notations and abbreviations

This technical code uses the following notations and abbreviations:
EP – emergency protection;
NPP – nuclear power plant;
UCC – unitized control center;
WMWCPR – water-moderated water-cooled power reactor;
NLA – normative legal act;
SR NPP S – substantiation report on the nuclear power plant safety;
AS – alarm system;
RF – reactor facility;
RCP – reserve command post;
CPS – control and protection systems; FA – fuel assembly;
FE – fuel element;
TNLA – technical normative legal act;
CSS – controlling systems for safety;
CSNO – controlling systems for normal operation;
OO – nuclear plant operating organization.

4 General provisions

4.1 Nuclear safety of RF and NPP is determined by technical perfection of the projects, required quality of fabrication, assembling, adjusting and testing of elements and systems essential for safety, their reliability in operation, equipment technical condition diagnostics, equipment maintenance support and repair quality and timeliness, monitoring and control of technological processes during operation, organizational management, qualification and discipline of the personnel.

4.2 Nuclear safety of RF and NPP is provided by the system of technical and managerial procedures specified by in-depth concept of protection including those at the expense of:

– employing and developing properties of the inner self-protectiveness;

– employing safety systems built on the basis of principles of backup, spatial and functional independence, single failure;

– employing reliable, practice proven engineering solutions and sound techniques, design and experimental analyses;

– fulfilling requirements of normative legal acts on safety of RF and NPP as well as strict adherence to requirements underlying projects of RF and NPP;

– technological process stability;

– implementing quality assurance systems at every stage of NPP creation and operation;

– forming and implementing safety culture at all stages of NPP creation and operation.

4.3 Inner self-protection properties are provided by means of:

– the reactor core energy release self-containment owing to negative reactivity coefficients on temperature of fuel, coolant, capacity;

– the actuation of regulating elements in mode of emergency protection based on gravitational forces;

– the applications of passive elements, cutoff, limiting and discharge devices;

– utilizing inertial running-down of the reactor coolant pump for securing necessary drop of flow rate through the reactor core at a tripped power generating unit.

4.4 Supplements and changes enter into this technical code on the authority resolution of the bodies that approved and conformed this technical code.

5 Nuclear safety assurance requirements for the reactor and the reactor facility systems intrinsic for safety

5.1 General requirements

5.1.1 Project engineering, construction and operation of a unit of NPP as well as designing and fabricating elements for RF and NPP shall realize in compliance with requirements of NLA including TNPA on safety of NPP acting in the Republic of Belarus.

5.1.2 The construction of NPP shall be preceded by the project development of RF and the project of NPP. The projects of RF and NPP shall determine systems intrinsic for safety, their principal characteristics, reliability, service life as well as order of their functioning, operational conditions, instruments for monitoring and diagnostics of these systems.

5.1.3 Change in composition, structure and (or) characteristics of RF and its system intrinsic for safety as well as operational conditions of NPP cannot be fulfilled without entering corresponding changes into the projects of RF and NPP.

5.1.4 With RF project development and (or) the reactor core modernization with employment of new structures for FA, new compositions of fission material, perfecting CPS and other systems intrinsic for safety, the necessary bench tests and reactor researches shall be performed. They shall show adequacy of completed researches in RF project for substantiation of the safety criteria fulfilment.

5.1.5 For all stages of RF and NPP life cycle there shall be developed programs for quality assurance

5.1.6 With a view of sustention and confirmation of design characteristics system (elements) of RF and NPP intrinsic for safety shall pass inspection and tests in process of manufacturing, assembling and adjustment as well as periodic check on-stream.

The projects of RF and NPP shall specify devices, procedures and periodicity of checks of the safety intrinsic systems for their conformity with their design characteristics including comprehensive testing (sequence and signal transit time, which includes response of EP, switching to emergency power supply, ensuring safety functions etc.).

The projects of RF and NPP shall determine a list of systems and elements whose working capacity and characteristics they check on the operating or shutdown reactor with the indication of condition of RF and systems of RF and NPP intrinsic for safety.

Checking devices and procedures for systems of RF and NPP intrinsic for safety and their elements shall not affect NPP safety.

5.1.7 Fundamental document on substantiation of RF nuclear safety is ‘Report on Safety Substantiation of Nuclear Power Plant (RSS NPP)’ (corresponding sections of RSS NPP). Elaboration of RSS NPP is provided by the operating organization with conformity observance of RSS NPP to the projects of RF and NPP.

5.1.8 The projects of RF and NPP shall identify and present in RSS NPP a list of initiating events of design basis accidents and enumeration of beyond design basis accidents, classification of design and beyond design basis accidents by the rate of occurrence and severity of after-effects as well as analysis of design and beyond design basis accidents and their after-effects. Among beyond design basis accidents they shall investigate those with severe damage of the reactor core.

5.1.9 While designing RF they shall seek to achieve the combined rate value of severe reactor core damage estimated on the basis of a safety probability analysis less than 10^{-5} per year.

5.1.10 The projects of RF and NPP shall contain analysis of possible failures of the systems (elements) intrinsic for safety with highlighting dangerous for RF and NPP failures and estimation of their after-effects on the basis of a probabilistic and deterministic safety analysis.

5.1.11 The projects of RF and NPP shall specify and substantiate operational limits and conditions, limits and conditions of safe operation as well as design limits determined for design basis accidents.

5.1.12 The projects of RF and NPP shall assign to each design basis accident or group of accidents design limits for design basis accidents that shall not be exceeded factored in the safety system action.

5.1.13 The projects of RF and NPP shall show that design basis accidents with hardest consequences do not exceed maximum design damage tolerance for fuel elements.

For other design basis accidents, design damage tolerance for fuel elements shall be determined by RF project and have values lower than maximum design damage tolerance for fuel elements.

Fuel elements damage tolerance for NPP with RF of WMWCPR type is presented in Appendix A.

For projected NPP with RF of other types such tolerance shall be substantiated in RF and NPP projects

5.1.14 The projects of RF and NPP shall specify a list of nuclear-hazardous jobs.

5.1.15 In the course of operation of RF, control shall be provided of main coolant activity.

5.2 Requirements for the reactor core and its structural components

5.2.1 The reactor core shall be designed to the effect that any changes of reactivity during normal operation and during breaches of normal operation including design basis accidents shall not lead to a breach of corresponding damage tolerance of fuel elements.

Requirements to reactivity coefficients of RF of WMWCPR type presented in Appendix A.

5.2.2 The project of RF shall show that at any design basis accident associated with fast increase of reactivity, fuel pellet cross-section averaged (mean-radial) fuel enthalpy shall not exceed limiting value determined by project based on experimental data. Destruction of fuel elements and FA shall also be excluded. For beyond design basis accidents, conditions shall be specified by which part of fuel elements and FA could be destroyed.

5.2.4 The project of RF shall establish a correspondence between damage tolerances of fuel elements and cooling circuit activity of the main coolant circuit by reference radionuclides factored in effectiveness of the coolant purification systems.

5.2.5 For substantiation of meeting requirements for non-exceedance of normal operation limits on damage of fuel elements during normal operation breaches, the project of RF shall present heat-engineering safety analysis of the reactor core with sufficiency substantiation of the RF project specified margins.

5.2.6 Enclosure oxidation of fuel elements while in RF service shall not lead to their excessive embrittlement. The RF project shall substantiate (based on experimental data) and specify equivalent oxidation degree of the enclosure of fuel elements during normal service and at normal service breaching including design basis accidents.

5.2.7 Design and performance of the reactor core and its elements including FA and fuel elements during normal operation and at normal operation breaching including design basis accidents shall not allow exceeding the corresponding damage tolerances of fuel elements factored in:

– design basis conditions of RF operation, their number and designed flowing;

– force (mechanical), heat and radiation effects on the reactor core components;

– physicochemical interaction of the reactor core materials and the cooling circuit;

– limit deviation of design, processing characteristics and parameters of the processes;

– impactive and vibration actions, thermocyclic loading, irradiation and thermal creep as well as aging of materials;

– effect of fission products and admixtures in the coolant on durability and corrosion stability of fuel elements;

– other factors deteriorating mechanical performance of the reactor core materials and enclosure integrity of the fuel elements.

5.2.7 The RF and NPP projects shall substantiate and secure with the design technology a possibility to discharge damaged components of the active zone after a design basis accident.

5.2.8 The reactor core and CPS actuating devices shall be designed to exclude wedging up, surge of working elements or their unprompted disconnection from CPS drives.

5.2.9 The RF project shall show that an unwanted motion of one or a group of most effective CPS working elements does not damage fuel elements with breaching safe operation tolerance factored in trip of EP without one most effective EP working element.

5.2.10 During normal operation and at normal operation breaches including design basis accidents there shall be exclusion of unwanted motions and (or) deformations of the reactor core elements causing reactivity addition and heat-sink deterioration leading to damage of fuel elements above corresponding design limits.

5.2.11 The RF and NPP projects shall show and substantiate that by seismic loads characteristic of the NPP site, working elements of regulation and EP unhampered entry into the reactor core is secured as well as reliable heat sink from the reactor core.

5.2.12 The reactor core and instruments affecting radioactivity characteristics shall provide introduction of negative reactivity at any section of motion of the instruments affecting radioactivity entering the reactor core and (or) the reflector for any combination of their arrangement during normal operation and at normal operation breaches including design basis accidents.

5.2.13 The FA design shall tolerate form alteration of fuel elements and other elements of FA possible during normal operation and at normal operation breaches including design basis accidents, without causing shut-off of the FA passage opening, which leads to damage of fuel elements above corresponding limits and hindering the CPS working elements normal functioning.

5.2.14 The FA design shall have distinctive marks characterizing the nuclide composition and nuclear fuel enrichment in the fuel elements, that differentiate visually and (or) by means of reloading devices.

5.2.15 Fuel elements of various enrichment, with burnable poison in the fuel, with blended fuel etc., special burnable poisons as part of FA shall have distinctive marks that differentiate visually and (or) by means of industrial control instruments while assembling FA.

5.2.16 The RF and NPP projects shall provide technological tools and testing methods for enclosure hermeticity of the fuel elements on the shutdown and (or) operating reactor, that shall secure reliable and timely detection of nonhermetic FA (fuel elements). Criteria shall be specified for culling nonhermetic fuel elements (FA). The RF and NPP projects shall present and substantiate techniques applied for checking the enclosure hermeticity of fuel element on shut-down and operating reactor.

5.3 Requirements for the emergency protection system

5.3.1 General requirements

5.3.1.1 The RF structure includes CPS whose function is:

– the reactor core reactivity and RF capacity control;

– the control of neutron flux density (capacity), rate of its change, technological parameters essential for protection and the reactor core reactivity and RF capacity control;

– the transition of the reactor into subcritical state and sustaining it in subcritical state.

5.3.1.2 The composition, structure, characteristics and CPS working order shall be substantiated in the RF project.

The RF project shall contain numerical reliability analysis that shall represent the fact that CPS indices of reliability comply with NLA requirements including TNLA regulating these parameters.

5.3.1.3 The RF project shall contain analysis of CPS reactions to external and internal actions (fires, earthquakes, floods, magnetic pickups etc.), to possible defects and failures (short circuits, deterioration of isolation quality, voltage drops and parasitic voltages, false responses, losses of control etc.) proving absence of hazardous for the RF reactions.

In case of discovering in the process of operation hazardous for RF reactions of CPS, RF shall be shut down and steps taken on their exclusion. OO in established practice shall provide introduction of corresponding alterations into the RF project.

5.3.1.4 The RF project shall provide at least two systems of the reactor shut down. Each of them independently of the other shall be capable of securing the reactor transition to subcritical state and sustaining it in subcritical state factored in single mode principle or personnel error. These systems shall be designed subject to diversity, independence and backing up.

5.3.1.5 At least one of the reactor shut down systems (not performing EP function) under normal operation and abnormal operation including design basis accidents, shall possess:

– effectiveness sufficient for the transition of the reactor to subcritical state and sustaining subcritical state factored in possible release of reactivity;

– high-speed performance sufficient for the transition of the reactor to subcritical state without breaching design tolerance of fuel elements determined for design basis accidents (with account of the reactor core emergency cooling system action).

5.3.1.6 The RF project shall specify and substantiate quantity, efficiency, arrangement, group composition, operative position, succession and rate of motion of the CPS working elements (including EP working elements), as well as number of the drives.

5.3.1.7 The RF project shall specify and substantiate methods and conditions of the tests, replacement and removing for repairs the CPS actuating elements, their drives as well as other instruments affecting reactivity.

5.3.1.8 The CPS actuating mechanisms shall have markings of intermediate positions of their working elements, signaling devices of their terminal positions and limit switches actuating (where possible) directly from the working element. Other instruments for operational affecting reactivity shall have state indicators and (or) terminal positions.

5.3.1.9 If the RF project provides for the use of additional (to nominal) CPS during first criticality, this system shall suit the requirements of Section 5.3 in the part related the system of CPS.

5.3.2 Requirements for the emergency protection system

5.3.2.1 At least one of the stipulated reactor shutdown systems shall perform the EP function.

5.3.2.2 The RI draft has to be shown that the reactor shutdown system operating in EP function without one of the most effective operative part have:

- Speed, sufficient to transfer the reactor into subcritical state without derangements of safe operation limits for derangements of normal operation;

- Efficiency, sufficient to transfer the reactor into subcritical state and keeping subcritical reactor state in derangements of normal operation, including project breakdown.

If the efficiency of EP is insufficient for long-term maintenance of the reactor in a subcritical state, the RI project shall be provided to connection another system (s) of the reactor shutdown automatically, which has (have) efficiency, sufficient to maintain subcritical state of the reactor in view of the possible positive reactivity release.

5.3.2.3 EP must have at least two independent operative parts.

5.3.2.4 EP shall be designed in such a way that the beginning protective effect has been completed, taking into account the requirements of 5.3.2.2 and provided monitoring of EP function.

5.3.2.5 The RI project shall be given the procedure for determining and eliminating the causes of the operation of emergency protection, as well as the sequence of actions of operating personnel to restore the normal operation of RI after EP operation.

5.3.2.6 After EP signal EP operative parts shall be operable from any working or intermediate positions.

5.3.2.7 If the EP operative parts is not shown in the working position by means of influence on the reactivity of the provided RI project positive input of reaction shall be excluded. Operating position of EP operative parts and the procedure for their installation shall be defined in RI project.

5.3.2.8 In the case of combining means of influence the reactivity of normal use and EP in the RI project developed and justified the order of their functioning. Priority shall be ensured by EP functioning.

5.3.2.9 EP structure shall be chosen from the condition of compliance with established criteria (single failure, common cause failure) and reliability indexes.

5.3.2.10 EP equipment shall consist of at least two independent sets.

5.3.2.11 Each set of EP equipment shall be designed so that the range of change in neutron flux density from 10.7% to 120% of the nominal providing protection:

- On the neutron flux density - at least three independent channels;
- On the rate of rise of the neutron flux density - at least three independent channels.

5.3.2.12 If necessary to partition density measurement range of neutron flux into subbands overlapping of subbands measuring at least within the same decimal order in units of the neutron flux density and the automatic switching subbands must be provided.

It shall be possible to connect a recording device to each channel control of the neutron flux density.

5.3.2.13 Each set of EP devices must be designed so that emergency protection in the entire range of process parameters established in RI project, provides at least three independent channels for each process variable, demanding protection.

5.3.2.14 Alarm of each set of EP equipment shall be carried out on majority logic, which is selected by reliability analysis in RI project. Minimum of a majority is equal to 2 from 3.

Control commands for each set of EP actuating mechanisms shall be transmitted at least through two channels.

5.3.2.15 The acceptability of unification in each set of EP equipment measuring units the neutron flux density channels control with measuring units of speed control channel of the neutron flux increase must be justified in RI project.

5.3.2.16 EP shall in such an extent be separated from CSNO to the deactivating or failure of any element of the CSNO does not affect the ability of PE to perform its functions.

5.3.2.17 Failure of channel control of elements display, information registration and diagnosis shall not affect the channel's ability to perform the PE functions.

5.3.2.18 For each of the channels and in the whole set of EP equipment shall be provided for possibility to verify the formation and transit time of emergency protection signals without operation of EP operative parts.

5.3.2.19 The EP system shall be provided with automatic control and diagnostics of serviceability of sets and PE hardware channels of data on the control board on failures in the channels, as well as the formation of the PE signal by failure of channels or sets.

5.3.2.20 The RI project shall contain and substantiate methods of metrological certification and PE equipment calibration.

5.3.2.21 Admissibility and conditions of deactivating of one set or a channel in the PE set (duration, power switchgear, the state of the other sets, etc.) shall be substantiated in RI project.

5.3.2.22 While deactivating of one channel in one of the sets of PE equipment without deactivating this set alarm for this channel shall be generated automatically.

5.3.2.23 The list of parameters necessary for PE function, settings and PE operation conditions, as well as the passage of signals before the operation of PE operative parts must be substantiated in RI project. The settings and PE operation conditions shall be chosen in such a way as to prevent the breach of the limits of safe operation.

5.3.2.24 The RI project shall provide and substantiate a list of initiating events demanding PE alarm.

PE actuation shall be at least in the following cases:

- At PE setpoint of neutron flux density;

- At PE setpoint on the rate of rise of neutron flux density;

- Undervoltage in any active set of PE equipment and CPS power buses;

- In case of failure of any two of the three protection channels for neutron flux density or the rate of rise of the neutron flux in any active set of PE equipment;

- At PE setpoint process parameters for which protection shall be implemented;

- At the initiation of PE actuation from PUCS (RCS).

5.3.2.25 The admissibility of the application of preventative protection (protections), in breach of normal operation, not requiring PE actuation, and the conditions of its (their) application shall be substantiated in RI project.

5.3.2.26 PE shall be designed in such a way that with the help of technical means unforeseen by the project RI is excluded and technological regulations without safe NPP operation unit exposure to the elements of the input and output channels of PE operation and change of settings without personnel notification and PE actuation of operative parts.

5.3.2.27 The performance of the emergency protection function of the reactor shall not depend on the presence and condition of power sources.

5.3.3 Requirements for monitoring and control of the neutron flux and reactivity

5.3.3.1 To monitor the neutron flux reactor shall be equipped with control channels so that all values of the neutron flux density in the core from 7.10% to 120% of the rated value control carried out at least:

- By three mutually independent channels measuring the neutron flux density with indicating instruments;

- By three mutually independent channels measuring the rate of change of neutron flux density.

5.3.3.2 Admissibility of combining parts of the control of measuring the neutron flux density measurement channels with frequent monitoring the rate of change of the neutron flux density of the channel shall be substantiated in RI project.

5.3.3.3 At least two of the three neutron flux monitoring channels shall be equipped with recording devices with the possibility of connecting any channel to control the neutron flux density. Recording devices shall provide possibility of measure and record of readings throughout the range of the project of the neutron flux density.

5.3.3.4 The channels control the neutron flux density shall be calibrated throughout the project range of the thermal power of the reactor. The RI project shall substantiate and define the method and procedure for such calibration and its periodicity in power NPP operation.

5.3.3.5 If the partition of the neutron flux density measuring range into several sub-bands overlap at least subbands within one decimal exponent in neutron flux density measure and automatic switching of subbands shall be provided for.

5.3.3.6 If the density of the control channels of the neutron flux as defined in 5.3.3.1, does not provide monitoring of neutron flux at core feed (refueling), the reactor shall be equipped with additional control. Additional control can be removable and installable while loading and reloading of the reactor core, and shall compose of at least three independent control channels of neutron flux density with showing and recording devices.

5.3.3.7 To monitor changes in the reactivity of RI project measurable reactivity with sensors, operational display and registration devices with the automatically switching range of neutron flux density and reactivity shall be provided.

5.3.3.8 Methods of determining the accuracy and responsiveness (the number and placement of sensors, algorithms and constants for calculation errors and measuring range) shall be substantiated in RI project.

5.3.3.9 Channels of monitoring reactivity shall be provided with means of automatic test performance and safety alarm signaling.

5.3.3.10 The RI project shall substantiate and present the methods of metrological certification and calibration of reactivity control channels.

5.3.3.11 The RI project shall substantiate and establish characteristics of the system of RI automatic power regulation, which provide RI operating without breaking the exploitation.

Possible and reasonable time of RI operation without a system of automatic control, the cardinality of, in particularly, its breakdown and allowed RI power for such conditions shall be substantiated in RI project.

5.3.3.12 To switch a few measurement channels at the input of the automatic regulation power the device receiving the signal from the working measurement channels shall be provided to shutdown or failure of one of these channels do not cause a change in reactor power due to the impact of the automatic control system.

5.3.3.13 Technical or organizational measures shall exclude the possibility of inputting of positive reactivity simultaneously by two or more means provided for the impact on the reactivity, as well as the introduction of positive reactivity means of impact on the reactivity of fuel loading and unloading.

5.3.3.14 The rate of increase in reactivity by exposure means shall not exceed 0.07 β_{ef}/s. For the CPS operative parts with efficiency more than 0.7 β_{ef} positive reactivity input shall be step-by-step, not more than 0.3 β_{ef} from step performance (provided by technical measures). The RI project shall point out the size of a step, the interval between the steps and the rate of increase in reactivity.

5.3.3.15 Before starting the reactor PE operative parts shall be discharged into the operational position.

Subcritical reactor at any time after PE operative parts cocking campaign in the operational position with the core inserted into the other organs of CPS shall be not less than 0.01 in the core condition with maximum effective multiplication factor.

5.3.3.16 The failure of the density control channel and (or) the rate of change of the neutron flux density shall be accompanied by signaling to the operator and registration. The failure signal of such a channel shall form.

5.3.3.17 The RI project shall provide for the requirements of the means providing at operating rapid automated identification and registration of values of the current stock of the reactor core reactivity and its changes. The RI project shall substantiate the procedure for determining the overall effectiveness of the impact on the responsiveness, efficiency of PE operative parts, the effectiveness of the groups of CPS operative parts, reactivity coefficients in the parameters affecting reactivity (power, temperature of the cooling circuit, the moderator temperature, the concentration of dissolved absorber, etc.), as well as methods for determining these values and the bias of their determination.

5.3.3.18 The RI project shall provide the means and methods of control of the reactor subcritical.

5.3.3.19 The RI project shall be provide monitoring the uneven energy-release in the reactor core and means for calculating the operational reserve to the crisis of the heat.

5.3.3.20 For the reactor cores, which are not proved the absence of density fluctuations of neutron flux, the RI project shall provide controls and management swaying of neutron flux density and show the order of swaying control without swaying the operating limits for fuel rod damage.

5.4 Requirements for the normal service controlling systems and safety controlling systems

5.4.1 The RI project shall present and substantiate demands to the composition, structure, main characteristics, quantity and placement conditions CSNO, CSS, their components, as well as RI diagnostic systems.

5.4.2 The RI project shall substantiate and give lists of:

- Monitored parameters and signals about RI;
- Adjustable parameters and controlling signals;
- Settings and operation conditions of the DBE;
- Placement of the RI diagnostic sensors;
- Parameters defining the commissioning of security systems.

5.4.3 The RI project shall show that CSNO and CSS provide technical inspection and safety switchgear control during normal operation and in violation of normal operation, including design basis accidents.

5.4.4 The RI project shall give and substantiate a list of protection and locking RI equipment, as well as the technical requirements to the conditions of their operation.

5.4.5 CSNO and CSS shall provide to the formation of at least the following signals:

- Alert (siren, which has a distinctive signal tone) - in cases foreseen by the RI project;
- Alarm (light and sound) - while achieving parameters of settings and conditions of PE operation;
- Warning (light and sound) – while the breach of the normal operation systems and RI components and the achievement of the parameters of settings and DBE operation conditions;
- Guide - the presence of voltage in the power supply circuits, the status of the equipment.

5.4.6 CSNO and CSS diagnosis shall present.

5.4.7 CSNO and CSS shall be designed in such a way that makes it possible to identify the initiating events of accidents, to establish the actual algorithms of reactor systems important to safety, deviations from regular algorithms and actions of the operational personnel.

5.4.8 In order to implement the 5.4.7 requirements it shall be provided for registration of:

- Parameters and features of the RI systems (elements) allowing reliably determined the initial event;
- Actuating signal;
- Changes in the parameters that characterize the condition of the reactor systems that are important to safety;

- Parameters that actuate protection systems;
- The position of reinforcement of security systems;
- Parameters that characterize the radiation environment;
- Actions of operational personnel, including video information;
- Operational personnel talks via communication systems.

5.4.9 The RI project shall substantiate and give the information about volume and intensity of registration and storage of the information specified in 5.4.8.

5.4.10 Means of registration shall continue to operate and to ensure the preservation of information in terms of design and beyond design basis accidents (in the "black box" type unit).

5.4.11 The RI project shall establish:

- Permissible values of the reactor power depending on CSNO efficiency at partial loss of function;
- The conditions of output in repair of CSNO and CSS and their parts.

5.4.12 Ranges and the rate of change during normal operation and in violation of normal operation, including design basis accidents shall substantiate for regulated and controlled parameters.

5.4.13 CSNO and CSS elements shall pass metrological examination and certification.

5.4.14 THE RI project shall include an analysis of the CSNO and CSS reactions on external and internal impacts for possible failures and faults (short-circuits, the loss of insulation quality, drop and voltage inducing, false alarms, loss of signals, etc.) and on the main failures of equipment of proving the absence of hazardous RI reactions.

In case of in-service RI-threatening reactions CSNO and CSS RI shall stop and take measures to their exclusion. OO shall make appropriate changes in the RI project in the prescribed manner.

5.4.15 The programmable and software use in CSNO and CSS shall be substantiated and confirmed by tests. Programmable and software use shall be verified.

5.4.16 RI and its systems monitoring shall be made with PUCS and (if necessary) from local government posts.

5.4.17 In each block, in addition to PUCS RCS shall be provide for. It provides reactor transfer in subcritical state and emergency RI shut-down cooling as well as control that are necessary for the RI technology parameters safety, if for any reason (fire and so on) it can be done with PUCS.

5.4.18 Requirements to the PUCS equipment, RCS and local control stations shall be specified in the RI project.

5.4.19 The RPU shall display information about the status of systems and individual system elements, including at least:

- The neutron flux density in the core;
- The parameters of cooling circuit and systems involved in the emergency cooling;
- Indicators of intermediate and end positions of operative parts of the CPS;
- Signs that impact on reactivity (condition of pumps steel framework and elements that uniquely identifies the readiness of the means for reactivity to perform its functions and the fact of their operation, as well as the parameters of the state of the liquid absorber solution (if used) - temperature, pressure, concentration, etc.).;
- Indicators of steel framework position and condition of the systems that provide cooling.

5.4.20 The possibility for control circuits frying and PUCS and RCS control on common cause when taken into account the initial events, and excluding possibility to manage simultaneously with PUCS and RCS for each specific item by technical means.

5.4.21 In the reactor, the main circuit, the tanks of the emergency reserve of liquid absorbent and in all systems that are filled the RI project with liquid absorber solution shall be provided with a given RI project concentration liquid absorber solution. The method and frequency of measurement of the concentration of nuclide-absorber in liquid absorber shall be defined in the RI project.

5.4.22 Technical control facilities of the content of nuclide- neutron absorbers in liquid or gaseous absorbers (in case they are used) in RI and in the emergency reserve capacities of the absorber during RI operation and technical means to maintain a uniform concentration of the absorber solution containing its capacity shall be provided.

5.4.23 Technical means or organizational measures shall provide with the input control of the content of nuclides in the neutron absorber materials used in the means of impact on the reactivity, for compliance with design specifications.

5.4.24 Each container of emergency reserve liquid absorber solution shall be equipped by at least two channels of level control and (or) pressure measurement with the issuance of warning signal on PUCS and RCS.

5.4.25 Under normal or abnormal operation, including a design basis accident (including the blackout mode) CSNO and CSS shall be provided with reliable electricity and energy services to the extent reasonable in RI project.

5.4.26 The CSNO structure shall include a system of industrial television and CSS communications, RCS and local control stations (telephone, loudspeaker, radio and so on).

5.4.27 CSNO and CSS shall provide the information support system operator.

5.4.28 CSNO and CSS shall provide communication tools in internal and external emergency NPP control centers in conditions beyond design basis accidents to assess the situation and make decisions.

5.4.29 The RI project shall give organizational and (or) technical measures to prevent unauthorized access to CSNO and CSS.

5.5 Requirements for cooling circuit (heat transport main circuit)

5.5.1 The RI project shall define borders of the main circuit.

5.5.2 The RI project shall substantiate a reliable operation of the main circuit elements and systems during project durability, taking into account the physicochemical, thermal, power, and other possible impacts during normal operation and in breach of normal operation, including design basis accidents. The number and nature of the impacts considered when determining the project durability shall be given and substantiated in the RI project.

5.5.3 The RI project shall show that the strength of the reactor vessel during normal operation, and breach of normal operation, including design basis accidents, provided for the durability of NPP power unit.

5.5.4 The layout of the equipment and the geometry of the main circuit shall provide conditions for the development of natural circulation of cooling circuit in the main circuit for the loss or absence of forced circulation, including the design basis accidents.

5.5.5 Pipelines of main circuit shall be equipped with monitoring devices and averting prevention movements when exposed to reactive forces arising at breaks.

The RI project shall substantiate the strength and effectiveness of these devices during design basis accidents.

5.5.6 Heat-exchange equipment to transfer heat from the main circuit of RI shall have a supply of heat exchange surface to compensate for the deterioration in its heat transfer characteristics during operation.

5.5.7 In the case of forced circulation pumps, performing the circulation, the loss of their power and PE actuation at any level of reactor power shall have sufficient inertia, which would provide a main circuit cooling circuit forced flow until the natural circulation ensures removal of residual heat without exceeding the operational fuel elements damage limits.

5.5.8 The RI project shall provide:

- Automatic protection against unacceptable pressure increase in the main circuit with normal operation and in breach of normal operation, including design basis accidents;

- Compensation for changes in cooling circuit volume caused by temperature changes;
- Compensation for loss of cooling circuit at leaks (maximum leakage flow, compensated by these means, installed in RI project).

5.5.9 The RI project shall provide for setting of stops leaks in pipelines extending from the main circulation pipeline. Failure to install the leak limiters shall be substantiate in the RI project.

5.5.10 The elements of the main circuit shall be equipped with devices that reduce the impact of seismic effects. Failure of such equipment of main circuit elements shall be grounded in the RI project.

5.5.11 The RI and NPP projects shall install cooling circuit quality parameters, its chemical composition and permissible content of radionuclides in the operation, provided the technical means and organizational measures for their maintenance and control.

Technical solutions and organizational measures to ensure the quality of the cooling circuit, as well as methods and means of control shall be justified in the RI and NPP projects.

5.5.12 The RI project shall provide technical measures to protect the main circuit from unforeseen by the technological regulations of safe operation of NPP unit drained cooling circuit. Admissibility of partial drainage during repair and reloading shall be substantiated in the RI project.

5.5.13 The RI project shall provide for the means of locality identification and the value of the main circuit coolant leak with reasonable accuracy in the project.

5.5.14 Technical measures shall be avoided unintended entering of pure condensate and liquid absorber solution with a concentration less allowable by RI project (NPP) in the main circuit coolant system and the other that RI project (NPP) shall be filled with liquid absorber solution.

5.6 Requirements for the reactor core emergency cooling system

5.6.1 The RI and NPP projects shall provide emergency core cooling system zone.

The composition, structure and characteristics of the emergency cooling systems of the core shall be substantiated in the RI and NPP projects.

5.6.2 The emergency core cooling system shall be designed taking into account the independence and redundancy principles and be capable of taking into account the principle of single failure or human error to perform the function of preventing infringements design fuel elements damage limits during design basis accidents.

5.6.3 List of parameters, settings and operation conditions of the emergency cooling systems shall be substantiated in the RI project (NPP) based on the analysis of the design basis accidents.

5.6.4 Admissibility and conditions of withdrawal from the work of a single channel emergency core cooling system shall be substantiated in the RI project (NPP).

5.6.5 The RI project shall take into account all the possible impacts on the systems (elements), associated with the inclusion of the work and emergency cooling systems of the core.

5.6.6 Technical and organizational measures shall be provided in the project for the exclusion of an unauthorized RI access to systems of emergency core cooling.

5.6.7 The project must contain substantiation of indicators of reliability of systems of emergency core cooling.

5.6.8 When the reactor is subcritical the switching on and the work of the accident core cooling shall not move it from the subcritical state.

5.6.9 Emergency cooling systems shall provide for cooling and prolonged maintaining the reactor core for values of parameters of the cooling circuit, founded in the RI project.

5.7 Requirements for the refueling devices and the reactor core refueling procedure

5.7.1 Requirements for the refueling devices

5.7.1.1 The RI project shall substantiate and give the composition of the refueling devices, as well as their requirements, the implementation of which ensures the safety of the treatment of SCC and other core elements in case of refueling, including the failure modes refueling devices.

5.7.1.2 They shall provide for the heat removal from SCC reloaded without exceeding the temperature parameters of fuel rods established by the RI project for refueling operations at normal operation and failures.

5.7.1.3 Refueling devices shall be designed so that at their normal operation and failures are not breach the conditions of RI normal operation, and reactor storage of nuclear fuel.

5.7.1.4 The RI project shall be give the requirements for installation, operation, maintenance, repair, testing and periodic inspection of refueling devices, as well as requirements for their reliability.

5.7.1.5 Refueling devices shall be designed so that the access for inspection, repair, testing and maintenance is possible.

5.7.1.6 The design of the refueling device shall provide for measures aimed at the prevention of damage, deformation, breaking or falling of SCC and other core components, as well as the invalid loading when removing or installing. The maximum loading level shall be given in the RI project. The use of non-design things for reloading is prohibited.

5.7.1.7 In the construction of the refueling device they shall provide for the power supply cessation that does not lead to SCC drop and other core elements refueling.

5.7.1.8 The RI project shall substantiate and set the permissible SCC speed and other core elements of the transfer device.

5.7.1.9 The hardware (blocking, and so on) shall provide moving refueling devices the permissible limits.

5.7.1.10 In case of cancellation or breach of service conditions of refueling devices by the RI project equipment for safe SCC and other core components movement shall be provided in a safe place.

5.7.1.11 The refueling devices shall provide panels to display unit for presenting information on the state and the orientation of the fuel assembly, and other refueling core elements and grippers.

5.7.1.12 It shall not be possible to move the device refueling at the time of connection with the technological channel, or when SCC and other refueling elements entering the core (extracting from the core).

5.7.1.13 Blocking shall be provided to prevent movement of refueling devices when SCC and other core refueling elements are in non-project status.

5.7.1.14 Industrial television shall be provided for the refueling control system. The RI project shall determinate a list of operations in case of refueling, controlled by using a commercial TV systems.

5.7.2 Requirements for the procedure of refueling

5.7.2.1 The RI project shall substantiate:

- Methods of refueling;
- The frequency, volume and refueling regulations;
- Technical and organizational measures to ensure nuclear safety during refueling, including the monitoring of the neutron flux density;
- The concentration of the working liquid absorber solution (if used), the sampling

point, the means of control and maintenance.

5.7.2.2 The RI project as initiating events, in addition to the system equipment failures refueling shall consider errors when load (refueling) and their consequences, as well as activities designed error exception.

5.7.2.3 The order of the refueling of the core is determined by the program and (or) refueling instruction, work schedule and refueling cartogram composed by NPP personnel, approved by the NPP administration and agreed in the prescribed manner.

5.7.2.4 In carrying out transshipment and repair organizational measures by technical means if possible penetration of foreign objects into the interior of the equipment, fittings and pipelines of RI shall be prevented.

5.7.2.5 In reactors where the refueling is carried out with the CPS release operative parts, the refueling shall be carried out when introduced into the CPS core operative parts and other means of influence on the reactivity. Minimum subcritical reactor during transfer, taking into account the possible errors shall be at least 0.02.

5.7.2.6 In reactors where the refueling is carried out with the help of the CPS release operative parts and reactivity is compensated by the liquid absorber solution, a refueling shall be carried out when introduced into the core operative part of the CPS and other means of influence on the reactivity. The concentration of the liquid absorber solution shall be brought to a level at which (taking into account possible errors) provided subcritical reactor is not less than 0.02 (excluding introduced CPS operative parts).

5.7.2.7 In reactors in which in case of refueling required subcriticality is ensured by liquid absorber solution, the technical means and organizational measures to guarantee the supply in case of refueling the exception of pure condensate to the reactor and into the main circuit shall be provided.

5.7.2.8 The vessel type reactor with CPS overhead drives reactor construction and CPS actuators shall provide the disengaged state of CPS operative part when removing the upper block. Diagnostic tools shall record the disengaged state.

5.7.2.9 The RI project shall provide technical measures to avoid the "ascent" of CPS operative part in case of refueling or in the project shall substantiate the impossibility of operative parts "ascent" in case of refueling.

5.7.2.10 Tests (measurements) for confirmation of the main design and calculation of neutron-physical characteristics of the core shall be carried out for reactors with a partial refueling after the refueling. Test period (measurements) shall be substantiate in the RI project for reactors with continuous refueling.

During the tests the compliance of the experimental result measure parameters with the criteria set out in the RI project shall be checked.

6 Requirements for nuclear safety provision during the nuclear plant unit implementation

6.1 General requirements

The NPP unit implementation after finishing construction and assembly work includes:

– realization of adjusting and startup procedures, which includes testing of the systems providing the nuclear safety;

– preparation technical and operational documentation;

– recruiting and training the personnel;

– performing first criticality and energy launch (comprehension test of the NPP equipment);

– reactor launch and power operation.

6.2 First criticality

6.2.1 Prior to first criticality the following equipment shall be ready to use with drawing up of their acts of readiness:

– reactor facility;

– CPS (actuating elements, detectors, electronic equipment and means for controlling actuating elements including the systems of logic and emergency protection);

– standard start-up instrumentation;

– non-standard start-up instrumentation (optional);

– devices for transportation, loading and unloading fresh and exhausted fuel;

– spent fuel conditioning cooling ponds;

– radiological monitoring system;

– system of chemical and special preparation of the cooling circuit, including heating system (if specified by the project);

– blowing and exhaust ventilation system;

– liquid regulation system (if specified by the project);

– reliable power supply system;

– alarm signal system throughout all the premises;

– ground grid;

– phone and loudspeaker communication;

– decontamination centers;

– fire extinguishing system.

6.2.2 In the process of first criticality, they shall obtain experimental data on neutron-physical parameters of the reactor, effects of reactivity, efficiency of the regulating elements and EP etc.

6.2.3 The first criticality including loading the reactor with nuclear fuel is realized in accordance with first criticality program.

OO together with the RF and NPP project developers provide development of the first criticality program and its compliance in accordance with the established procedure.

6.2.4 The program of the reactor first criticality shall contain:

– the list of the systems and equipment, required for the reactor first criticality;

– the order of realizing the reactor loading with FA (fuel elements);

– the scheme of criticality attainment;

– the description of tests (measurements) and the order of their performing;

– expected values of critical loadings, critical positions (states) of the instruments affecting reactivity, their efficiency, estimates of the cooling circuit influence on FA (fuel elements);

– testing and measurement procedures;

– measures on securing nuclear safety during realization of the first criticality.

6.2.5 To achieve the first criticality, they shall prepare the following documentation:

– Techniques for realizing the tests and measurements in the course of first criticality. The techniques are developed by OO with participation of the RF and NPP developers;

– Service instruction on the reactor facility operation that determines the order of the RF NPP safe operation. The instruction is developed by OO based on project-designing documents and process regulations for safe operation. It is conformed with the RF and NPP project developers and approved by the NPP chief engineer;

– Instruction on the accident elimination, determining the actions of the reactor personnel and services of the nuclear power plant in case of the accident (in this number, nuclear accident). The instruction is developed by the operating organization, and it is agreed with the project developers of the switching equipment and the nuclear power plant and the Ministry of Emergency Situations of the Republic of Belarus;

– Instruction on the provision of nuclear safety at the first criticality realization;

– Instruction on the provision of nuclear safety at the transportation, reload and storage of the fresh and spent fuel;

– technical documentation, including description of the equipment and systems, providing nuclear safety;

– operative documentation (operative journals, collation maps journals, etc.);

– certificates and protocols of the testing of CPS as well as the control instrumentation of the reactor facility;

– order of the assignment of the first criticality research supervisor, his deputies and the first criticality implementation group;

– order of the head of the operating organization on the work authorization of the shift personnel, who have passed examinations for corresponding working positions;

– duty instructions of the operating personnel and statement on the controlling physicist, confirmed by the NPP administration;

– certificate of the working committee on the readiness of systems, equipment and personnel to the first criticality;

– permission of the State Acceptance Commission for the first criticality implementation.

6.2.6 Test of the readiness of NPP to the first criticality is carried out by:

– working committee assigned by OO;

– committee of the state regulatory agency in the field of the nuclear and radiation safety.

6.2.7 Working committee inspects:

– the correspondence of the fulfilled work to the RF and NPP projects;

– the operability of the equipment, presence of the equipment test certificates, and certificates on the completion of the prestart setup labors;

– the presence and issuance of the operational documentation;

– the availability of labor permits for operation personnel and passed examinations minutes for the controlling physicists.

Working committee prepares a certificate on the readiness of systems, equipment and personnel for first criticality. The certificate shall be approved by OO in the prescribed manner.

6.2.8 Committee of the state regulatory agency in the field of nuclear and radiation safety inspects:

– the technical readiness of NPP for the first criticality;

– project and operational documentation;

– the readiness of the personnel for first criticality implementation.

6.2.9 The first delivery of nuclear fuel to the site of NPP unit being commissioned could be realized upon availability of license of the state regulatory agency in the field of nuclear and radiation safety on operation of the NPP unit and subsequent to the results of inspections by the named agency on readiness for delivery of nuclear fuel.

6.2.10 The decision on the first criticality implementation is taken in the prescribed manner based on working committee certificate on readiness of the systems, equipment and personnel for first criticality implementation, as well as on the OO certificate on corrective measures according to results of the inspection of the state regulatory agency in the field of nuclear and radiation safety on the readiness of the NPP unit for first criticality implementation.

6.2.11 In case of the occurrence of pre-accident condition during the implementation of tests (measurements) during the first criticality implementation, tests (measurements) shall be stopped and the reactor shall be transitioned to its subcritical state.

6.2.12 The results of the reactor core load with FA (fuel elements) as well as testing results during first criticality implementation shall be documented as acts and reports to be presented to the state regulatory agency in the field of nuclear and radiation safety according to the established order.

6.3 Energy launch

6.3.1 Energy launch of the NPP unit includes stepwise and gradual increase of power, determination and specification of parameters of RF and the NPP unit, complex probation of systems and equipment of the NPP unit, realization of planned testing (measurements) on every stage, and analysis of the obtained results.

6.3.2 Energy launch of the NPP unit is realized according to the program of the energy launch of the NPP unit, corrected (if necessary) according to first criticality results. The program of energy launch is developed and approved by operating organization.

6.3.3 The program of energy launch shall contain the procedure of its implementation, expected values of the neutron-physical characteristics of the reactor (reactivity effects etc.), thermal performance of RF, testing implementation techniques, measures for providing nuclear safety at energy launch implementation, etc.

6.3.4 The program of energy launch shall provide for the testing and trial of the NPP energy unit operation modes, checking security systems in the scope and sequence that provides the reactor safe attainment of the nominal power level, including the development of safe and dynamically stable passage of transient modes on all the power consumption stages.

6.3.5 The examination of the readiness of the NPP unit for energy launch is carried out by working committee. Working committee checks the readiness of systems and equipment of the NPP unit for energy launch, the reactor attainment of the required power value, setting the turbo-generator in operation and the connection of the NPP unit to the power network, its level of equipment with the shift personnel, its preparation and work permit. The committee prepares the certificate of the readiness of the NPP unit to the energy launch. The certificate shall be confirmed by the operating organization according to the established order.

If required, the state regulatory agency in the field of nuclear and radiation safety shall send a special committee for examination of the readiness of the NPP unit for energy launch.

6.3.6 Energy launch of the NPP unit is performed after realizing corrective measures as indicated in the act of the working committee and in the certificate of the state regulatory agency in the field of nuclear and radiation safety (in the case of examination by the state regulatory agency in the field of nuclear and radiation safety).

6.3.7 The decision on energy launch implementation is taken in accordance with the established order on the basis of the certificate of working committee on the readiness of the NPP unit for energy launch, and on the basis of the OO certificate on corrective measures in accordance with results of the inspection of the committee of the state regulatory agency in the field of nuclear and radiation safety (in case of its realization) of the readiness of the NPP unit for energy launch.

6.3.8 A report shall be issued by the operating organization on the results of the first criticality and energy launch. SR NPP S shall be corrected (if required).

7 Requirements for nuclear accident prevention during the nuclear plant unit operation

7.1 The basic document determining the safe operation of the NPP unit is technological rules of safe operation of NPP unit containing the rules and the basic techniques of safe operation, the general order of operations related to security, as well as the limits and conditions of safe operation. The operating organization ensures development of technological rules of safe operation of the NPP unit.

7.2 The operation of the NPP unit shall be conducted in accordance with the instruction manual, based on the NPP administration on project documentation and technological rules of safe operation of NPP unit, adjusted according to the results of the physical and power starting.

7.3 Before the NPP unit operation operating organization shall be issued a passport on RI. The form of the passport and the amount of information are set by state regulation in the field of nuclear and radiation safety.

7.4 The operational organization on the base of RI and NPP projects and the AU with the requirements of technological rules of safe operation of NPP unit organizes the development and production of systems important to safety:

- Instructions for the inspections and tests;
- Carrying out maintenance schedules and preventive and complete overhauls of systems and components;
- Testing schedules and safety systems operation checks.

7.5 RI and its system conditions and the conditions under which permitted NPP unit operation shall be grounded in RI and NPP projects, and are listed in the regulations technological safe with NPP unit supervision.

7.6 In case of breach of operational limits by operating personnel the steps leading to the establishment in the RI (NPP) project and the technological regulations of the normal operation of the NPP unit shall be taken. In case they fail to the renew the normal operation they shall shut the NPP unit down.

7.7 In the case of pre-crash (accident) NPP unit shall be stopped, its causes shall be clarified and eliminated and the measures shall be taken to restore the normal operation of the NPP unit. Operation NPP unit can be continued only after the elimination of the causes of pre-crash (accident).

7.8 The operating organization shall investigate the accident at the plant in accordance with the NLA of the Republic of Belarus, as well as to transmit information on these breaches in accordance with established procedure.

7.9 In design basis accidents personnel actions shall be determined by the instruction to eliminate accidents in the NPP unit, developed by OO based on NPP SAR. The instructions shall consider design basis accidents and develop measures to eliminate its consequences.

7.10 To manage beyond design basis accidents in accordance with RI and NPP projects and NPP SAR operating organization shall be developed guidance on management of beyond design basis accidents.

7.11 The guidelines for the management of beyond design basis accidents shall specify the order of entry the action plan of measures to protect personnel and the population in case of a beyond design basis accident.

7.12 To prepare the NPP personnel to acts in pre-emergency situations and accidents emergency trainings shall be carried out. The frequency and the procedure for their implementation shall be approved by the OO.

7.13 Since the beginning of the accident and prior to the Commission that identifies the causes of the accident one shall not open instrumentation and devices, change settings, and emergency warning and protection. There shall be technical means and organizational measures that exclude the possibility of the loss of a registered information and unauthorized access to devices and components, databases and archives management system, which recorded the state of the equipment and systems before and after crash.

7.14 The RI project shall substantiate and technological regulation of safe NPP unit operation shall give the conditions of safe operation of the shutdown reactor with nuclear fuel in the core, including the boot modes and refueling. For these modes must be defined as a minimum:

- Volume control in accordance with the requirements of 5.3.3.1, 5.3.3.3 and 5.3.3.7 with the obligatory control of neutron flux density and concentration of the liquid absorber solution, if it is applied to this type of RI;

- Requirements to the availability of systems important for safety.

7.15 In reactors in which the loading and reloading of fuel are performed during filling liquid absorber solution reactor main circuit and associated systems, the concentration of the liquid absorbent solution during loading operations and reactor refueling, as well as testing of equipment, valves and pipes of the main circuit and during repairs shall not be below that defined in the RI project (NPP).

7.16 The operating organization based on project documentation and project list of nuclear hazardous work shall draw up a list of nuclear hazardous work of NPP unit.

7.17 Any work with systems (elements) that are important for safety, repair and operating as well as testing of systems (components) not covered by the technological regulations of safe operation of the NPP unit and instructions for use is nuclear hazardous.

7.18 Nuclear hazardous work shall be carried out by a special working program agreed by NPP administration.

Nuclear and hazardous work not covered by the technological regulations of safe NPP unit operation and the instruction manual shall be carried out by a special working program, approved by the OO at the coordination of the RI and NPP project developers.

The work program shall contain:

- The object of a nuclear-hazardous work;
- The list of nuclear hazardous work;
- Technical and organizational measures to ensure nuclear safety;
- Criteria and control of the correct completion of nuclear hazardous work;
- The appointment of a person responsible for carrying out nuclear and hazardous work. Nuclear and hazardous work shall be carried out, as a rule, on the shutdown reactor.

7.19 The subcriticality of the shutdown reactor during nuclear hazardous work shall be not less than 0.02 for the condition of the core with a maximum effective multiplication factor.

7.20 After completing the repair of equipment and systems important to safety these systems characteristics shall be audited to meet project specifications. Testing shall be conducted in accordance with the applicable regulations or developed in order, established by the NPP operating organization.

7.21 All tests of systems important to safety shall be verified with test results set out in the RI and NPP projects. The results shall be made out in act.

8 Supervision over this technical code requirements observation and responsibility for violating them

8.1 The operating organization shall continuously monitor the compliance with the requirements of this technical code.

8.2 The operating organization is responsible for the creation of the necessary organizational structures at the NPP, which would ensure compliance with the requirements of this technical code.

8.3 The operating organization organizes periodic (at least once every two years) to verify the compliance with the requirements of this technical code by NPP and establishes the procedure for inspections of nuclear plant safety by internal committees. The results of inspections carried out by the OO shall be submitted to the regulatory agencies in the field of nuclear and radiation safety.

8.4 The managers, engineers and technicians of enterprises and organizations responsible for breach of the requirements of this technical code are responsible in accordance with applicable law.

8.5 Heads of project designing, research, construction, installation, commissioning, repair companies and organizations, as well as enterprises – equipment manufacturers required to monitor compliance with the requirements of this standard in the design, construction, installation, commissioning, repair and manufacture of RI (NPP) equipment.

8.6 The operating organization is responsible for ensuring nuclear safety, organization and carrying out of works to ensure the RI (NPP) safe operation and personnel qualification.

Appendix A

(compulsory)

Additional safety requirements for the nuclear power plants with WMWCPR reactor facilities [1)]

A.1 The operational limit of the fuel elements damage:

– defects of the gas leakiness type – no more than 0.2 % of the overall number of fuel elements in the reactor core;

– direct contact of nuclear fuel with the heat-transfer agent – no more than 0.02 % of the overall number of fuel elements in the reactor core.

A.2 The limit for safe operation of the fuel elements damage:

– defects of the gas leakiness type – no more than 1 % of the overall number of fuel elements in the reactor core;

– direct contact of nuclear fuel with the heat-transfer agent – no more than 0.1 % of the overall number of fuel elements in the reactor core.

A.3 The maximal project limit of the fuel elements damage corresponds to a non-exceedance of the following boundary parameters:

– temperature of the fuel elements sheaths – not more than 1200 °C;

– equivalent degree of the fuel elements sheaths oxidation shall not exceed the threshold value, which is established in frames of the project on the base of experimental data;

– the fraction of reacted zirconium in the reactor active zone shall not exceed 1 % of its mass in the fuel elements sheaths;

– maximal temperature of the fuel shall not exceed the melting point.

A.4 The reactivity coefficients values by the specific volume of the heat-transfer agent and the fuel temperature, on the reactor power, and values of the summarized reactivity coefficient on the heat-transfer agent temperature and the fuel temperature shall not be positive in all the critical states, which occurrence is possible in the entire range of reactor parameters change at the normal operation conditions and at violations of the normal operation conditions, including design accidents.

[1)] The fuel element cladding damage limits are given as for the alloy of Zr+1%Nb

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