RUSSIAN MARITIME REGISTER OF SHIPPING

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RULES

FOR THE CLASSIFICATION AND CONSTRUCTION OF NUCLEAR-POWERED VESSELS AND FLOATING FACILITIES

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Saint-Peterburg
2012

Rules for the Classification and Construction of Nuclear-Powered Vessels and Floating Facilities developed by Russian Maritime Register of Shipping have been approved in accordance with the established approval procedure and come into force since 1 May 2012.

The present (seventh) edition of these Rules is based on the sixth edition (2008) taking into account additions and amendments developed immediately before publication.

These amendments and additions are induced by introduction of new Part XIII "Physical Security" based on research and development work. These Rules incorporate proposals of organizations contributing to construction and operation of nuclear-powered vessels as well as accumulated experience regarding application of the sixth edition of these Rules.

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As compared to the previous edition (2008), this seventh edition of these Rules contains the following additions and amendments:

PART I. GENERAL

1. Section 1: Para 1.5 has been introduced specifying Guidelines on Technical Supervision During Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Support Vessels and Manufacture of Materials and Products as a document supplementing requirements of these Rules.
2. Section 3: Para 3.1: new definition for the Montejus has been introduced.
3. Section 4: Abbreviations induced by introduction of the new Part XIII "Physical Security" into these Rules have been introduced.

PART II. CLASSIFICATION

1. Section 2: Table 2.2: item 6 "Physical Security" and item 6.1 "System of Physical Security Engineering Facilities" have been introduced.
2. Section 3: Para 3.2: clarification as regards reviewing design for the nuclear-powered vessel has been introduced.

Para 3.2.1.5 has been introduced.

Para 3.3: text of the first paragraph has been changed.

Para 3.4.3: document name has been clarified.

PART IV. HULL

1. Section 6: Para 6.12: definition for the leakage rate has been clarified.

PART XII. RADIATION SAFETY

1. Section 5: Para 5.17 has been amended.

Para 5.19: requirements to surface quality have been clarified.

2. Section 7: Para 7.1: the scope has been clarified.

Para 7.2 has been clarified.

Paras 7.3.1 to 7.3.5, 7.3.7 have been amended.

Paras 7.3.8 to 7.3.13 have been introduced.

Para 7.4: addition has been introduced.

The new Part **XIII "Physical Security"** is **introduced**.

APPENDIX 3 CONTAINMENT LEAK TIGHT CIRCUIT COMPONENTS FOR NUCLEAR-POWERED STEAM GENERATING PLANTS. PROCEDURE FOR CALCULATING LEAK TIGHTNESS STANDARD VALUES

1. Para 3.2 and 3.3: units of measurements for physical quantities have been introduced.

2. Para 4.4: Table 4.4 has been amended.

3. Para 4.5.5: clarifications have been introduced.

4. Para 4.5.6: clarifications have been introduced.

5. Para 5.5.5: clarifications have been introduced.

6. Para 6.3 has been deleted. Requirements of these Rules have been included into the Guidelines on Technical Supervision During Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Support Vessels and Manufacture of Materials and Products.

CONTENTS

PART I. GENERAL

PART II. CLASSIFICATION

PART III. SAFETY STANDARDS

PART IV. HULL

PART V. SUBDIVISION

PART VI. FIRE PROTECTION

PART VII. MACHINERY INSTALLATIONS

PART VIII. NUCLEAR-POWERED STEAM GENERATING PLANTS

PART IX. SPECIAL SYSTEMS

PART X. ELECTRIC EQUIPMENT

PART XI. AUTOMATION

PART XII. RADIATION SAFETY

PART XIII. PHYSICAL SECURITY

PART I. GENERAL

1 APPLICATION

1.1 These Rules for the Classification and Construction of Nuclear-Powered Vessels[1] and Floating Facilities developed by the Russian Maritime Register of Shipping[2] apply to nuclear-powered vessels and floating facilities (for definitions of nuclear-powered vessel and nuclear-powered floating facility, see Section **3**) fitted with two-circuit nuclear steam generating plants and pressurized water reactors.

1.2 Scope of requirements to nuclear-powered vessels and floating facilities using steam generating plants other than those specified in Para **1.1** is subject to special consideration by the Register.

1.3 These Rules represent the main normative document that regulates safety aspects related to special nature of nuclear-powered vessels/floating facilities having a harmful radiation effect on personnel, passengers, population and environment.

1.4 These Rules establish safely standards and criteria for nuclear-powered vessels/floating facilities, their classification principles and procedure as well as design and testing requirements to be met to ensure safely.

1.5 Requirements of these Rules are supplemented by provisions of Guidelines on Supervision During Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Support Vessels and Manufacture of Materials and Products.

2 APPLICATION

2.1 The requirements of the Rules for the Classification and Construction of Sea-Going Ships[3], Rules for the Equipment of Sea-Going Ships, Rules for the Cargo Handling Gear of Sea-Going Ships, Load Line Rules for Sea-Going Ships and Rules for the Prevention of Pollution from Ships apply in full to nuclear-powered vessels and floating facilities other than those to which the other provisions of these Rules apply.

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[1] Hereinafter referred to as the Rules.

[2] Hereinafter referred to as the Register.

[3] Hereinafter referred to as the Rules of the Register.

2.2 Requirements of these Rules apply to nuclear-powered vessels/floating facilities, their machinery and equipment with draft designs submitted to the Register for approval after the effective date of these Rules.

2.3 Nuclear-powered vessels/floating facilities under construction as well as their items with documents approved by the Register before the effective date of these Rules are to comply with requirements that were valid at the time of approval of those documents.

2.4 Nuclear-powered vessels/floating facilities to be refitted or repaired are to comply with requirements of these Rules as far as it is reasonable and technically grounded in each specific case.

3 DEDINITIONS AND EXPLANATIONS

3.1 For definitions and explanations concerning general terms of the Rules of the Register, see **Part I** "Classification" and appropriate parts hereof.

For the purpose of these Rules, the following definitions have been adopted.

Reactor emergency protection is a function of reactor control and protection system intended to prevent development of accidents on reactor by switching the reactor to the subcritical state.

Emergency state is a state of the vessel, plant or unit after abnormal short-term failure to perform their intended functions (post-accident state).

Reactor core is the portion of a nuclear reactor containing the nuclear fuel components where the controlled nuclear chain reaction takes place.

Nuclear steam generating plant is a component of the power unit designed for steam generation from nuclear energy.

Nuclear power unit is a main power unit designed to perform primary functions of nuclear-powered vessel or nuclear floating facility. Nuclear power unit comprises steam generating plant, main and auxiliary turbine-generating plants and electric power plant.

Nuclear-powered floating facility is a self-propelled/dumb floating facility fitted with nuclear power source (power plant, thermal plant or other facility) as a source designed to perform its primary functions.

These facilities are generally designed as rack-mounted facilities. Self-propelled floating facility may use organic fuel/nuclear power for its propulsion. Requirements to nuclear-powered self-propelled floating facilities and their steam generating plants are similar to those imposed to nuclear-powered vessels and their steam generating plants. Requirements to steam generating plants of organic fuel self-propelled floating facilities are the same as those imposed to steam generating plants of dumb floating facilities.

Nuclear-powered vessel is a vessel propelled by nuclear power.

Metal-water shielding tank is a multi-layer metal tank filled with water between the layers. This tank is to be used for attenuation of radioactive radiation emitted from nuclear reactor core.

Reactor biological shielding comprises reactor structural components and water layer for protecting personnel against radioactive radiation.

Leakage rate is an air mass/volume escaped from the controlled volume per unit time at given initial parameters (pressure, temperature).

Relative leakage rate is a ratio of leakage rate (by weight/volume) to air mass/volume in the controlled volume at given initial parameters (pressure, temperature) expressed as a percentage per unit time.

Gaseous radioactive waste is defined as a waste discharged in air as gaseous and aerosol emissions.

Leak tightness is the ability of structures, systems and their components to withstand gas/fluid exchange through them within design limits.

Date of construction of nuclear-powered vessel/floating facility is defined as an actual date of expiry of Classification Certificate and termination of technical supervision of vessel/floating facility under construction.

Single failure is defined as an accidental event which results in the loss of capability of a component/system to perform its intended safety/technical functions. Multiple failures which result from one event/operator's error are considered as parts of a single failure.

Liquid radioactive waste is defined as a radioactive fluid containing dissolved or suspended radionuclides in concentrations greater than values as specified in applicable standards and regulations. This radioactive fluid is not to be utilized.

Beyond design-basis (anticipated) accident is defined as an accident being analyzed in the design project as unlikely so no safety measures are taken to prevent it.

Containment is defined as a vessel structure with steam generating plant inside. The containment is designed to maintain the radioactive emissions released from steam generating plant within permissible limits.

Shielding barrier is a vessel's structural barrier surrounding the containment and major radioactive sources related to steam generating plant. This shielding is additionally provided to minimize release of radioactive materials from the containment to the environment and other compartments of the vessel or floating facility.

Safety class is defined as a class assigned to structures, systems and their components according to their significance for nuclear safety of the vessel or floating facility. This safety significance is defined with regard to effects of loss of functions being performed in various intended situations.

Design class is a class which establishes the specific design standards for equipment and systems of steam generating plant according to their impact on steam generating plant safety.

State class is a combination of states selected by its frequency of occurrence and assumed effects which can occur under normal operation or foreseen operating faults and accidents as well as when the vessel/floating facility is exposed to external or internal forces, natural and human-induced disasters.

Active component is defined as a component driven by external exposure (excitation, mechanical exposure or power supply)[1].

Passive component is defined as a component with no moving parts which is sensitive to variation in pressure, temperature and working fluid flow[2].

Collision, grounding and stranding protection comprises specific structures of the vessel/floating facility in the vicinity of reactor compartment and fuel assemblies storage facilities which protect steam generating plant, its safety systems and storage facilities for radioactive waste and nuclear fuel against actions in the event of collision or grounding and stranding.

Controlled area is defined as an area which comprises compartments of the vessel/floating facility with higher level of ionizing radiation and/or radioactive contamination under normal conditions, with controlled access and to which Standards for Protection Against Radiation are applicable.

Maximum design-basis accident is defined as an accident resulting in the highest radiation hazard for the crew and environment. In general, maximum design-basis accident is related to the rupture of the primary coolant pipeline.

Montejus is a special-purpose enclosed container for collecting and storing liquid radioactive waste with liquid pumped out by means of compressed air.

Supervised area comprises compartments of the vessel/floating facility where radioactive contamination and higher levels of ionizing radiation are likely to occur in case of abnormal operation of steam generating plant which requires continuous radiation monitoring.

Normal operating state and habitability conditions represent conditions when the vessel/floating facility, its machinery, systems and equipment which ensure its propulsion, intended operation, steerability, safe

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[1]For example, pumps, fans, relief valves, non-return valves, etc.

[2]For example, heat exchangers, pipelines, vessels, electric cables, etc.

navigation, buoyancy, internal signals and communication, means of escape, boat winches operation as well as minimum habitability conditions are working properly (i.e. capable of performing all functions within given design limits and conditions including startup, operation at power, deactivation, maintenance, testing and nuclear fuel handling).

S i m i l a r - t y p e f a i l u r e is defined as a failure in several devices or components due to particular event or for some reason.

M a i n d e s i g n - b a s i s a c c i d e n t is defined as the submitted accident which defines basic design requirements to the vessel/floating facility, steam generating plant and its safety systems.

O p e r a t o r ' s e r r o r is a single operator's erroneous action or inaction (when action is required) towards controls.

P r i m a r y c i r c u i t o f s t e a m g e n e r a t i n g p l a n t is the reactor-steam generator closed tight circuit with coolant circulating over it. The coolant removes heat from the nuclear reactor core and transfers the heat converted into steam in steam generators to the secondary circuit water.

P e r s o n n e l r e p r e s e n t crew members to be exposed to ionizing radiation according to their occupation.

E m e r g e n c y c o o l i n g c o n t r o l s t a t i o n is defined as an area or compartment of the vessel/floating facility fitted with equipment and devices for disabling the steam generating plant in case of failure in central control station.

P o t e n t i a l n u c l e a r - h a z a r d o u s o p e r a t i o n is defined as an operation which may result in pre-emergency or nuclear/radiation accident.

S i n g l e f a i l u r e c o n c e p t is defined as a capability of the system to perform its design functions in case of single failure.

D e s i g n - b a s i s a c c i d e n t is an accident considered and analyzed in the draft design of the steam generating plant and vessel/floating facility. This accident is prevented by means of appropriate arrangements and procedures and harmful effects are reduced to the applicable standards.

R e a c t o r c o n t r o l a n d p r o t e c t i o n s y s t e m a c t u a t o r is a device for changing reactor reactivity being moved by a single drive of the reactor control and protection system.

R a d i a t i o n s a f e t y i s d e f i n e d as capability of the facilities and measures used to protect crew, passengers and environment against harmful radioactive radiation and contamination within specified limits.

P r o c e s s r a d i a t i o n m o n i t o r i n g is defined as monitoring of the state of steam generating plant equipment and shielding barriers for all state classes by recording ionizing radiation by means of special-purpose instruments and procedures.

Radioactive waste is defined as equipment, items, materials, substances in any aggregative state which are not intended for further use, with radionuclide content exceeding the permissible values as specified in applicable standards and regulations.

Radioactive waste may be divided into solid, liquid and gaseous waste. Radioactive waste is classified by the degree of its radioactivity as per applicable Principal Sanitary Radiation Safely Rules.

Reactor plant is a component of the nuclear power unit. The reactor plant comprises the nuclear reactor, systems and equipment directly related to the reactor to provide its normal operation, prevent and control accidents as well as reduce their effects.

Reactor compartment is a watertight compartment of the vessel/floating facility restricted by its bottom, sideboards, bulkhead deck, forward and aft bulkheads or cofferdams with the reactor plant inside.

Decontamination station is a special-purpose compartment or cluster of compartments designed for checking for radioactive contamination of personnel, changing clothes and shoes as well as for sanitary treatment of personnel allowed to the controlled area.

Unrestricted area comprises all premises of the vessel/floating facility which do not form part of the controlled/supervised area.

Reactor control and protection system is a combination of technical, software and information facilities to provide appropriate conditions for safe chain reaction at a given power and its variation at startup, stop, reactor switchover, to check for chain reaction intensity, ensure fast termination of fission reaction in case of accident as well as to control power density fields.

Safety systems are defined as systems designed to disable the reactor reliably, remove heat from the reactor core and/or reduce effects of foreseen operating deviations and accidents.

Solid radioactive waste is defined as solid items, materials and substances contaminated with radioactive materials in concentrations exceeding the permissible levels as specified in applicable standards and regulations. This waste is no longer useful.

Physical security is defined as a combination of measures and engineering facilities to prevent sabotage/acts of terror and theft regarding to nuclear materials and nuclear plants installed on board nuclear-powered vessels/floating facilities.

Central control station is defined as a compartment of the vessel/floating facility designed for control and monitoring of nuclear power unit operation under normal conditions, in case of foreseen operating deviations and accidents.

Nuclear accident is an accident related to damage of fuel elements exceeding the specified safe operation limits.

Nuclear safety of the nuclear-powered vessel /floating facility is defined as a capability of the vessel/floating facility and crew to reduce harmful radiation impacts on the crew and environment down to specified limits under normal operation and in case of accidents.

Nuclear reactor is defined as a device used to initiate and maintain the controlled nuclear fission chain reaction.

ABBREVIATIONS

EEBDs — Emergency escape breathing devices - Emergency protection.
LRW — Liquid radioactive waste.
SRW — Solid radioactive waste.

PART II. CLASSIFICATION

1 CLASS NOTATION OF NUCLEAR-POWERED VESSEL AND FLOATING FACILITY

1.1 Where the nuclear-powered vessel/floating facility is fitted with nuclear power unit and complies with requirements of the RS Rules and these Rules, radioactivity symbol shall be added to the class notation specified in **Part I** "Classification" of the RS Rules. .

2 CLASSIFICATION SURVEYS OF NUCLEAR-POWERED VESSEL AND FLOATING FACILITIES IN SERVICE

2.1 Classification surveys of nuclear-powered vessels and floating facilities shall be performed as per requirements of this **Part** and Rules for the Classification Surveys of Ships in Service developed by the Register.

2.2 For additional scope of periodical surveys of nuclear-powered vessels and floating facilities, see Table **2.2**.

Surveys upon the **15**-year cycle expiry shall be repeated as per Table **2.2**. The scope of survey shall be established by the Surveyor depending on technical state and used service life of the vessel/floating facility.

The Register may assign the continuous survey of the nuclear-powered vessel/floating facility based on Owner's written statement. The continuous survey cycle shall cover the period not exceeding the prescribed period between appropriate periodical surveys with regard to delays as permitted by Rules for the Classification Surveys of Ships in Service.

Scheduled surveys of steam generating plant shall be typically combined with reactor core handling or other operations associated with opening of primary circuit, replacement, repair or preventive maintenance of equipment. In any case, during core handling and before further activation of the plant, the following surveys and tests shall be performed:

.1 surveying double bottom, structures and foundations within the reactor compartment;

.2 surveying biological shielding;

.3 surveying and testing pressure vessels, pipelines and fittings related to steam generating plant;

Table 2.2

ADDITIONAL SCOPE OF PERIODICAL SURVEYS OF NUCLEAR-POWERED VESSEL AND FLOATING FACILITY

See circular 846c

Symbols:

O — inspection with accessing, if required, and opening by means of remote inspection and non-destructive testing arrangements
C — external inspection
M — wear, clearance, insulation resistance measurements, etc
H — hydraulic, pneumatic tests
P — functional test of machinery, equipment and devices, their external inspection
E — checking for applicable documents and/or stamps on instruments calibration by competent authorities

No.	Item of survey	Vessel survey														
		First annual survey	Second annual survey	Third annual survey	Fourth annual survey	First annual survey	First annual survey	Second annual survey	Third annual survey	Fourth annual survey	Second scheduled survey	First annual survey	Second annual survey	Third annual survey	Fourth annual survey	Third scheduled survey
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	**Hull**															
1.1	Underwater hull[1]	C	C	C	C	O	C	C	C	C	OM	C	C	C	C	OM
1.2	Collision protection	C	C	C	C	O	C	C	C	C	OM	C	C	C	C	OM
1.3	Grounding and stranding protection	C	C	C	C	C	C	C	C	C	OM	C	C	C	C	OM
1.4	Supporting structures, platforms and foundations within reactor compartment					O					O					O

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1.5	Containment															
1.5.1	Hatch cover, covers, doors, windows, cable boxes, shut-off and safety valves and fittings of containment	C	C	C	C	CH	C	C	C	C	CH	C	C	C	C	CH
1.5.2	Containment bulkhead, protection units, lining	C	C	C	C	OH	C	C	C	C	OH	C	C	C	C	OH
1.6	Shielding barrier	C	C	C	C	CH[2]	C	C	C	C	CH[2]	C	C	C	C	CH[2]
2	**Steam generating plant equipment**															
2.1	Nuclear reactors	P	P	P	P	O[3]HP	P	P	P	P	O[3]HP	P	P	P	P	O[3]HP
2.1.1	Case and main connector pins					O[3]					O[3]					O[3]
2.1.2	Covers with their fasteners					C[3,4]					C[3,4]					C[3,4]
2.1.3	Internal removable and non-removable parts					O[4]P					O[4]P					C[4]P
2.1.4	Safety devices	P	P	P	P	CP	P	P	P	P	CP	P	P	P	P	CP
2.2	Control and protection systems (actuators)	P	P	P	P	OPM	P	P	P	P	OPM	P	P	P	P	OPM
2.3	Coms[4]					CE					CE					CE
2.4	Parameter testing arrangements, instruments	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE
2.5	Steam generating plant machinery															
2.5.1	Primary coolant circulating pumps	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM	CPM

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
2.5.2	Fresh water pumps for equipment cooling and protection	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.3	Core emergency cooling pumps	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.4	Sea water pumps for equipment cooling	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.5	Pumps and ejectors of primary water drainage, storage and discharge system	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.6	Pumps and ejectors for steam generating plant spaces drainage system	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.7	Primary make-up pumps	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.8	Automation hydraulic pumps	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.9	Pumps of residual heat removal system	P	P	P	P	P	P	P	P	P	OP	P	P	P	P	OP
2.5.10	Gas, air compressors for steam generating plant service	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.11	Controlled area fans	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
2.5.12	Pumps for pressure reduction system in containment	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
2.5.13	Spare parts					C					C					C
3	**Pressure vessels and units**															
3.1	Steam generators 5,6	P	P	P	P	HP	P	P	P	P	HP	P	P	P	P	HP
3.1.1	Case										O[6]					O[6]
3.1.2	Piping systems					O					O[6]					O

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
3.1.3	Supporting structures										O					O
3.1.4	Safety devices	P	P	P	P	OP	P	P	P	P	OP	P	P	P	P	OP
3.1.5	Fittings and valves	P	P	P	P	OHP	P	P	P	P	OHP	P	P	P	P	OHP
3.2	Pressure compensators[5]	P	P	P	P	HP	P	P	P	P	HP	P	P	P	P	HP
3.3	Primary filters with refrigerators[5]	P	P	P	P	HP	P	P	P	P	HP	P	P	P	P	HP
3.4	Hydraulic chambers[5]	P	P	P	P	HP	P	P	P	P	HP	P	P	P	P	HP
3.4.1	Case										O[8]					O[8]
3.4.2	Internal structures										O[8]					O[8]
3.4.3	Supporting structures										O[8]					O[8]
3.5	Heat exchangers for equipment fiesh water cooling circuit					C					CH					CH
3.6	Drainage and bilge containers					C					CH					C
3.7	Gas and air tanks	P	P	P	P	HP	P	P	P	P	HP	P	P	P	P	HP
3.8	Hydropneumatic cylinders	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
3.9	Metal-water shielding tanks	P	P	P	P	CP	P	P	P	CP	P	P	P	P	P	CP
4	**Steam generating plant systems**															
4.1	Primary coolant circulation system	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.2	Primary coolant purification system	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.3	Primary coolant make-up system	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.4	Residual heat removal system	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
4.5	Core emergency cooling system	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.6	Deaeration system	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.7	Primary water drainage system[5]	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.8	Primary volume compensation system[5]	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.9	Secondary system (to the secondary circuit)	P	P	P	P	HP	P	P	P	HP	P	P	P	P	P	HP
4.10	Fresh water system for equipment cooling and protection	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.11	Sea water system for equipment cooling	P	P	P	P	OP	P	P	P	OP	P	P	P	P	P	OP
4.12	Air ventilation and purification	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.12.1	Fittings and valves on the containment	P	P	P	P	CH	P	P	P	P	CH	P	P	P	P	P
4.13	Liquid radioactive waste collection, storage and discharge system					P					P					P
4.14	Steam generating plant spaces drainage system	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.15	Explosive mixture removal system					P					P					P
4.16	Automation hydraulic and fitting control system	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P
4.17	Containment pressure reduction system	P	P	P	P	P	P	P	P	P	P	P	P	P	P	P

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
5	**Radiation safety**															
5.1	Biological shielding[9]	OC	OC	OC	OC	OCM	OC	OC	OC	OC	OCM	OC	OC	OC	OC	OCM
5.2	Radiation monitoring systems and arrangements	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE	PE
5.3	Core handling equipment[10]	C	C	C	C	OHP	C	C	C	C	OHP	C	C	C	C	OHP
5.4	Fuel assemblies storage facilities	C	C	C	C	OH	C	C	C	C	OH	C	C	C	C	OH
5.5	Liquid radioactive waste treatment equipment	C	C	C	C	OHP	C	C	C	C	OHP	C	C	C	C	OHP
6	Physical security															
6.1	System of physical security engineering facilities	C	C	C	C	P	C	C	C	C	P	C	C	C	C	P

[1]Surveys in the vicinity of the reactor compartment shall be conducted annually.

[2]In case of containment depressurization, it shall be tested for leak tightness between scheduled surveys. Shielding barrier tests (H) may be omitted if pressure within the shielding barrier is maintained below atmospheric pressure.

[3]Survey shall be conducted in accessible places without dismantling after scheduled measurements.

[4]Survey shall be conducted before loading.

[5]Hydraulic tests shall be combined with the reactor (see item 2.1 of this Table).

[6]Survey shall be conducted during replacement of piping system.

[7]Hydraulic tests shall be combined with systems which they are serving.

[8]Survey shall be conducted in the course of primary pumps dismantling.

[9]Biological shielding shall be tested for efficiency by means of radiation monitoring system and portable devices.

[10]Surveys shall be conducted prior to being used on the reactor.

.4 surveying reactor including its assemblies to be dismantled (with the core unloaded) by means of remote inspection and non-destructive testing arrangements;

.5 surveying and testing the primary system;

.6 surveying machinery and systems serving the steam generating plant;

.7 integrated functional test of steam generating plant and safety systems;

.8 testing the containment for leak tightness;

.9 functional tests of radiation monitoring system;

.10 integrated tests of travel and motion forces of control rods group;

.11 surveying the set of fuel assemblies before loading.

The state of reactor body shall be estimated by stressed assemblies non-destructive testing system.

Survey shall be also conducted by means of remote control arrangements.

2.4 In the course of surveys and tests of steam generating plant equipment, the personnel involved shall be properly protected against radioactive radiation including additional biological shielding and decontamination, if required.

3 TECHNICAL DOCUMENTS

3.1 Design documents for the nuclear-powered vessel and floating facility being the most sophisticated items shall be considered in the process of developing of technical statement. Draft design materials shall be submitted for consideration in a scope as agreed upon with the Register.

3.2 Design For The Nuclear-Powered Vessel And Floating Facility Under Construction.

In addition to documents listed in Part I "Classification" of the RS Rules, the following documents for nuclear-powered vessels and floating facilities shall be submitted to the Register:

3.2.1 General:

.1 Information on safely of the vessel /floating facility[1] (see Appendix 1).

.2 Operating Manual for the nuclear power unit of the vessel /floating facility[2] (see Appendix 2).

.3 Layout of controlled and supervised areas

.4 Water- and gastight plan for the containment and shielding barrier

.5 List of equipment located within the controlled area.

[1]Hereinafter referred to as the Information on safety

[2]Operating Manual may be submitted at a later design stage.

3.2.2 Hull:

.1 Structural diagram for reactor compartment main girders;

.2 Structural diagram for biological shielding;

.3 Containment drawings;

.4 Collision protection diagram;

.5 Grounding and stranding diagram;

.6 Description of containment leak tightness test instruments and procedures;

.7 Calculation results on biological shielding and metal-water shielding tank attachment strength.

3.2.3 Fire Protection.

Structural fire protection diagram for reactor compartment (may be included into general fire protection diagram for the vessel).

3.2.4 Steam **Generating Plant.**

For documents on steam generating plant design, see Section **3**, **Part** VIII "Nuclear-**P**owered Steam Generating **P**lants".

3.2.5 Radiation Safety.

.1 Chart for radiation within the vessel and on its external surfaces;

.2 Memorandum for biological shielding calculation results;

.3 Radiation monitoring system for the vessel/floating facility (description, schematic diagram, layout on board the vessel/floating facility, calculation results and drawings for the system and its equipment, delivery specifications);

.4 Decontamination procedures for spaces and equipment subject to radioactive contamination.

3.2.6 Systems **and Piping:**

.1 Schematic diagrams for systems serving the steam generating plant;

.2 Calculation results on systems and piping.

3.2.7 Electric Equipment.

Schematic power supply diagram for steam generating plant consumers, automatic systems and radiation monitoring system

3.2.8 Automation Equipment:

.1 List of remotely-controlled fittings and valves with types, Manufacturers and Register's approval certificates;

.2 Control algorithms for steam generating plant and steam-turbine unit;

.3 Functional and schematic diagrams for automation and remote control of steam generating plant systems, safety systems and systems serving steam generating plant (components required for systems operations shall be specified: converters, manipulators, actuators, etc);

.4 Functional and schematic circuits for power air and water systems;

.5 Functional and schematic circuits for control management from emergency cooling station

3.3 Design **Documents For The Nuclear-Powered Vessel/ Floating Facility under Construction.**

Design documents for the nuclear-powered vessel/nuclear support vessel to be submitted for approval shall be established for each design as agreed upon with the Register. The following documents shall be submitted to the Register for approval.

3.3.1 Hull:

.1 Drawings for sections and assemblies of reactor compartments main girders;

.2 Containment drawings;

.3 Biological shielding drawings;

.4 Containment test procedure.

3.3.2 Piping.

Drawings for piping with layout and piping assemblies penetrating containment and biological shielding, bulkheads, decks and platforms shall be established for each design as agreed upon with the Register.

3.3.3 Steam Generating Plant:

.1 Layout and securing arrangement drawings for steam generating plant equipment;

.2 Operating Instructions for steam generating plant;

.3 Test procedure during mooring and sea trial;

3.3.4 **Radiation Safety.**

Layout and attachment plans for radiation monitoring system equipment.

3.3.5 Electric Equipment.

Drawings for cable routing within reactor compartment with assemblies penetrating containment and shielding barrier

3.3.6 Automation Equipment:

.1 Layout and securing arrangement drawings for safety system equipment and systems serving steam generating plant;

.2 Drawings for cable routing and pulse piping;

.3 Layout drawings for sensors required for operation of steam generating plant systems, safety systems and systems serving steam generating plant;

3.4 Report **Documents** for **Nuclear-Powered** Vessel/**Floating Facility.**

3.4.1 When the nuclear-powered vessel/floating facility is constructed, tested and commissioned, report documents for nuclear-powered vessels shall be forwarded to the Register Branch Office as per the procedure established in Section **11**, Part II "Technical Documents" of the Rules for Technical

Supervision During Construction of Vessels and Manufacture of Materials and Products for Vessels.

3.4.2 Report documents shall be submitted in a scope as specified in Appendix to this Section.

3.4.3 The onboard operating documents shall be updated when the vessel is submitted to the Register for survey.

REPORT DOCUMENTS
FOR NUCLEAR-POWERED VESSEL/FLOATING FACILITY

In addition to documents for nuclear-powered vessels/floating facilities mentioned in Appendix to Part II "Technical Documents" of the Rules for Technical Supervision During Construction of Vessels and Manufacture of Materials and Products for vessel, the following report documents shall be submitted to the Register.

1 General:

.1 Information on Safety;

.2 Instructions on containment testing during operation;

.3 Water- and gastight plan for the containment and shielding barrier;

.4 Layout of equipment within propulsion compartment spaces of nuclear-powered vessel;

.5 Layout of controlled and supervised areas.

2 Hull:

.1 Structural diagram for reactor compartment main girders;

.2 Structural diagram for biological shielding;

.3 Containment drawings;

.4 Protection diagram for reactor compartment.

3 Vessel's Arrangements:

.1 Drawing for instrument space hatch cover;

.2 Drawings for cargo transportation to solid waste storage facility.

4 Fire Protection:

.1 Structural fire protection diagram for reactor compartment.

5 Nuclear-Powered Steam Generating Plant:

.1 General layout of nuclear-powered steam generating plant equipment within the containment;

.2 Specifications for nuclear-powered steam generating plant;

.3 Primary and secondary circuits diagram.

6 Systems:

.1 Diagrams for special-purpose systems:

.1.1 Third circuit system;

.1.2 Primary deaeration system;

.1.3 Steam generator washing and storage system;

.1.4 Steam generator leak detection system;

.1.5 Emergency cooling system;

.1.6 Branching and cooling of nuclear-powered steam generating plant;
.1.7 Condensate-feeding system;
.1.8 Soluble poison injection system;
.1.9 Emergency overfilling system;
.1.10 Circuit water drainage system;
.1.11 Special-purpose drainage system;
.1.12 Emergency overfilling system;
.1.13 Containment drenching system;
.1.14 High pressure gas system;
.1.15 Fourth circuit system;
.1.16 Sorbent unloading system;
.1.17 Decontamination system;
.1.18 Controlled area ventilation system;
.1.19 Pressure suit air system;
.2 Control Instructions for nuclear-powered steam generating plant;
.3 Scheduled checks for nuclear safety systems and equipment;

7 Electric Equipment:

.1 Power supply and control circuit for primary circulating pump;
.2 Electric drives of nuclear-powered steam generating plant auxiliary machinery;
.3 Functional tests of power supply system and power supply circuits of nuclear-powered steam generating plant;
.4 Schematic main and emergency power supply circuit for nuclear-powered steam generating plant machinery;
.5 Drawings for cable routing within reactor compartment;
.6 Main and emergency lighting layout for nuclear-powered steam generating plant spaces;

8 Automation:

.1 Emergency parameter recorder (electric circuit diagram);
.2 Schematic diagram for control and monitoring system of nuclear-powered steam generating plant;
.3 Algorithms of nuclear-powered steam generating plant and steam-turbine unit;
.4 Parameters to be checked (nuclear-powered steam generating plant);
.5 Specifications for local control devices of nuclear-powered steam generating plant;
.6 Radiation monitoring system schematic diagram.

PART III. SAFETY STANDARDS

1 GENERAL

1.1 These requirements are mainly aimed at general safety of the nuclear-powered vessel/floating facility which includes safety of the nuclear power unit. Operation of the nuclear power unit may be required for safety of the nuclear-powered vessel/floating facility although in the context of the nuclear power unit safety, it shall be stopped or its power shall be reduced.

2 BASIC REQUIREMENTS

2.1 The following basic requirements shall be met to ensure safety of the nuclear-powered vessel/floating facility and to protect crew, people and environment against radioactive materials.

2.1.1 To reduce dissemination of ionizing radiation and radioactive materials into environment, radioactivity sources shall be surrounded by several sequential shielding barriers.

2.1.2 In addition to main normal operation systems, special-purpose safety systems which start automatically upon accident shall be provided.

2.2 To ensure protection against ionizing radiation the following is required:

.1 to provide proper biological shielding;

.2 to earmark radiation areas on board the vessel/floating facility;

.3 to reduce time of exposure;

.4 to prevent people from being in the vicinity of radiation sources unless it is necessary;

.5 in case of accidents personnel shall act in accordance with Operating Manual for the vessel/ floating facility;

.6 to provide individual protection means.

2.3 Steam generating plant control and protection systems, safely systems and other technical facilities specified in these Rules are to comply with single failure concept (see Section 7).

2.4 To confirm that safely of the nuclear-powered vessel /floating facility complies with requirements of these Rules, all required operating and emergency conditions are to be scrutinized in the detailed design with regard to purpose of the vessel/floating facility and their assumed frequency and effects are to be evaluated.

Based on this evaluation, safety design concepts are adopted assuming that the more severe effects may be permitted at less frequency.

2.5 Nuclear-powered vessel/floating facility is to be designed, constructed and operated to meet the requirements of quality assurance program to be approved by the Register as part of the Safely Information. The requirements to quality of structures, systems and equipment shall comply with their classification based on importance for the vessel's safely.

2.6 Under normal operation of the nuclear-powered vessel/floating facility and steam generating plant, all shielding barriers against radioactive materials are to be operational. Steam generating plant shall not be operated at power if design safety limits of shielding barriers or their safely arrangements are beyond those specified in the detailed design of the nuclear-powered vessel/floating facility according to safely operation conditions.

3 STATE CLASSES

3.1 States of the vessel/floating facility and its nuclear power unit shall be subdivided into four groups (SC1, SC2, SC3 and SC4) depending on their frequency and effects as per Table 3.1.

Table 3.1

State Class	State of the vessel and steam generating plant	Possible frequency	Effects
1	2	3	4
SC1	Normal operation	Permanent or often	Vessel/floating facility and its steam generating plant are in normal operating state Radiation environment on board the vessel is within standard limits.
SC2	Minor faults	Occasionally	Faults which do not result in significant damage of vessel/floating facility operation. Short-term stop of the reactor may be required. Minor deviations of radiation from standard limits which do not result in increase in exposure of personnel on board the vessel/floating facility beyond standard limits
SC3	Major damage	Rarely	Damages to vessel structures/nuclear power unit equipment which partially impair the use of the vessel/floating structure. Long-term shutdown of steam generating unit and containment isolation may be required. Possible deviations of radiation on board the vessel/floating facility from standard limits Exposure of personnel on board the vessel/floating facility is not beyond the specified limits.

1	2	3	4
SC4	Severe accidents	Very occasionally	Severe damages which require activation of emergency cooling system/containment operation but which do not result in unacceptable radioactive emissions into environment. Radiation on board the vessel/floating facility significantly deviates from permissible limits. Exposure of some persons on board the vessel/floating facility does not exceed the double value of permissible dose as specified by applicable Radiation Safety Standards for the Personnel.

N o t e s : Permanent/often means that the event occurs permanently or may occur often within service life of this vessel/floating facility.

Occasionally means that the event may occur more than once within service life of this vessel/floating facility.

Rarely means that the event is unlikely to occur within service life of this vessel/floating facility but may occur on some sister ships/floating facilities within their service life.

Very occasionally means that the event is unlikely to occur but anyway it may occur within the total service life of particular sister nuclear-powered vessels/floating facilities.

Class assignment of possible event shall be justified, approved by the Register and specified in the Safety Information. Evaluation results of frequency and effects of possible accidents shall be specified in the detailed design for the vessel/floating facility and its nuclear power unit.

3.2 The effects of events that are too unlikely to occur and followed by total loss of operability of all vessel's power sources (capsizing, flooding, grounding and stranding with heel more than 30°, etc) shall be evaluated. The effects of these events are not governed by these Rules.

The effects of the beyond design-basis accident shall be also accessed in the detailed design.

3.3 For class assignment of steam generating plant states, see Part VIII "Nuclear-Powered Steam Generating Plants".

4 SAFETY CLASSES

4.1 Systems and equipment of nuclear-powered vessel, floating facility and nuclear power unit are subdivided into four safety classes according to their importance for the vessel and floating facility. These systems and equipment are to comply with design requirements, requirements to materials, manufacture, testing and operation.

4.2 Division of systems and equipment into safety classes shall be justified in the detailed design according to impact of failure in systems and equipment on vessel's safety, as well as approved by the Register and included into Safely Information.

4.3 For division of steam generating plant into safely classes, see Part VIII "Nuclear-Powered Steam Generating Plants".

5 DIVIDING NUCLEAR-POWERED VESSEL/FLOATING FACILITY INTO RADIATION SAFETY AREAS

5.1 The nuclear-powered vessel/floating facility shall be divided into controlled, supervised and unrestricted areas according to existing or potential radiation hazard. Area borders may be established in the form of physical structures or administratively.

6 BASIC DESIGN CRITERIA AND SAFETY FUNCTIONS

6.1 To ensure safely in all operating conditions, the nuclear-powered vessel/ floating facility shall comply to following criteria:

.1 Criterion **A**: to provide means for proper shielding of ionizing radiation sources and means to reduce dissemination of radioactive materials to the least possible level to ensure that people's exposure to radiation and environmental contamination are low to the extent practicable;

.2 Criterion **B**: means for effective removal of residual heat from the reactor core shall be provided;

.3 Criterion **C**: means for safety control and reactor switching to subcritical state and its maintaining in this state within required time shall be provided.

6.2 To satisfy basic criteria specified in Para **6.1**, the following safety functions are to be met:

.1 Criterion **A**:

.1.1 Fuel claddings in the reactor core being the first shielding barrier between nuclear fuel and environment are to be properly maintained;

.1.2 The integrity of the primary heavy duty circuit being the second shielding barrier is to be maintained;

.1.3 Accidental release shall be excluded and leakage of radioactive materials from containment being the third barrier shall be reduced;

.1.4 The leakage of radioactive materials from shielding being the fourth barrier shall be additionally reduced.

.2 Criterion B:

.2.1 Residual heat shall be removed from the reactor core to the coolant;

.2.2 Coolant shall be supplied to the reactor core (core emergency cooling);

.2.3 power supply to safety facilities specified in Para 6.2.2.1 и 6.2.2.2;

.3 Criterion C:

.3.1 Reactivity shall be properly monitored;

.3.2 Reactor shall be switched to subcritical state without exceeding specified design limits for the core;

.3.3 Power shall be fed to facilities which perform safety specified in Para 6.2.3.1 and 6.2.3.2.

6.3 To ensure performance of safety functions specified in Para 6.2 for states SC2, SC3, SC4, special-purpose safety systems shall be provided in addition to operation support systems. Based on analyzed possible accidents and effects it is determined whether special-purpose safety systems are required.

7 SINGLE FAILURE CONCEPT

7.1 In the design process, safety systems shall properly respond to any initial event related to SC2, SC3 and SC4 despite of supposed single failure in any system component.

7.2 When analyzing the safety system for compliance with single failure concept, the single failure in each component is supposed, thereby:

.1 Safety system shall be evaluated assuming that the initial event (together with any other failures resulting directly from the initial event) is combined with an accident failure in one component of safety system.

Two or more simultaneously independent failures are not required to be considered;

.2 Operator's error shall be considered as a kind of single failure or initial event.

Failures in well-designed, manufactured and tested passive components (such as pipelines, vessels, heat exchangers, electric cables) are not required to be considered provided that there are no sufficient grounds;

7.3 To comply with single failure concept, the equipment and systems shall be reliable and appropriate redundancy methods shall be used (component wise or subsystem-based redundancy) supplemented by the following, if required:

.1 Components/subsystems shall be divided by bulkheads or located apart from each other;

.**2** Subsystems shall operate independently;

.**3** Components/subsystems shall be designed differently by operating principle, design, etc.

7.4 The single failure concept shall be applied rather to the safety system considered as the system of facilities intended for safety purpose than to components of this system even though these components/subsystem are capable of operation as a system.

7.5 The single failure concept is not required to be complied with in case of states which are more unlikely than SC4 followed by total loss of operability of vessel's power supply units (capsizing, flooding, etc).

8 ENVIRONMENTAL CONDITIONS

8.1 The states of the nuclear-powered vessel, floating facility and their nuclear power unit to be designed shall be considered under extreme environmental conditions in assumed operation area (hurricanes, tsunami, ice). The state of floating facilities exposed to seismic waves shall be also considered.

8.2 Inertial forces affecting the vessel/floating facility at a sea state shall be accepted with regard to equipment safety class. When calculating inertial forces, vessel/floating facility six degrees of freedom roll (6DoF roll) shall be considered over the sea spectrum within the navigation/docking area. In general case, sea spectrum based on statistical data for the North Atlantic region may be used.

8.3 Components and structures of 1 to 4 safety classes[1] shall be capable of withstanding inertial forces of accepted sea spectrum acting for a time specified in Table **8.3**.

Table 8.3

Safety class of components and structures	Time, days
1	15000
2 and 3	1500
4, as well as hull and machinery not subject to the requirements of international standards and rules	150

[1]For steam generating plant safety classes, see Section 5, Part VII "Nuclear-Powered Steam Generating Plants".

PART IV. HULL

1 GENERAL

1.1 The hull of nuclear-powered vessel and floating facility shall comply in full with requirements of Part II "Hull" of the RS Rules and requirements hereof.

2 SCOPE OF TECHNICAL SUPERVISION

2.1 In addition to Part II "Hull" of the RS Rules, the following structures of nuclear-powered vessels and floating facilities are subject to technical supervision by Register during their manufacture:

.1 Collision, grounding and stranding system;
.2 Containment;
.3 Shielding barrier;
.4 Metal-water shielding tanks.

3 MATERIALS

3.1 Materials to be used for manufacturing structural system components and containment shall be of category D (for thicknesses below **12**.5 mm) and category E (for thicknesses above **12**.5 mm). For steel grades, see Part XII "Materials" of the RS Rules.

4 TOTAL STRENGTH

4.1 Calculation results which confirm that longitudinal strength of the vessel/floating facility is sufficient when longitudinal girders are broken due to design collision shall be submitted to the Register.

4.2 The longitudinal strength of the grounded and stranded vessel/floating facility shall be evaluated.

4.3 The total strength of the vessel/floating facility under action of bending torque in the water plane due to design collision with the other vessel shall be evaluated.

4.4 The hull shall be designed in a way to avoid to the extent practicable the drastic change in cross-section drag torque in structural protection area (see Para 5.1).

There shall be a smooth transition between the structural protection area (see Para 5.1) and the reminder of the hull. This transition shall ensure continuous girders contributing to the total strength of the vessel/floating facility The transition area shall be designed so as to forces induced in way of the reactor compartment and structural protection are transferred to the other parts of the vessel/floating facility hull.

5 HULL LOCAL STRENGTH NEAR REACTOR COMPARTMENT

5.1 To prevent damage to shielding barriers, structural protection for absorbing power released due to collisions, grounding and stranding shall be provided in way of the reactor compartment.

If the vessel/floating facility is fitted with helicopter/helideck, structural protection against helicopter crash shall be provided in way of the reactor compartment and facilities for storing cores and fuel assemblies.

5.2 The design conditions for collision and procedure for calculating the structural onboard protection shall be approved by the **R**egister.

The **R**egister may require experimental check of calculation results by modeling, where necessary.

5.3 The length of structural protection forward and aft of traverse bulkheads of reactor compartment and used fuel storage compartment shall be justified by the Designer with regard to Para **4.4** hereof. The length shall be at least 0.2 of compartment length.

5.4 Double bottom and foundations in the reactor compartment shall be designed as to ensure protection of reactor, its safety systems and core storage facilities against damages due to grounding and stranding.

The bottom of the vessel/floating facility shall be 5/15 or 2 m (whichever is less) apart from the lower part of the shielding barrier.

5.5 The height of the double bottom in the vicinity of the engine compartment shall be sufficient to withstand the damage with dimensions specified in Para 2.1, Part V "Subdivision".

6 CONTAINMENT

6.1 The containment shall be designed as to reduce release of radioactive materials into environment for any states of the plant (SCI to SC4). For permissible leakage values, see Para **6**.9.

6.2 Containment may be designed as a reinforced leaktight structure if the vessel/floating facility hull or as independent reinforced leaktight containment which is no integral with the hull.

If the vessel/floating facility is fitted with several steam generating plants, there plants shall be enclosed in a separate containment.

6.3 Containment shall be designed and manufactured to meet requirements of safety class **2** structures (see Section 5, Part VII "Nuclear-Powered Steam Generating Plants").

6.4 Containment shall withstand the inner pressure due to emergency release of coolant caused by rupture of the primary circuit (see Para 7.9, Part VII "Nuclear-Powered Steam Generating Plants").

Safety valves to release gas-vapor mixture at SC4 are not permitted on the containment.

If there is an approved system for reducing pressure in case of emergency release, maximum pressure which may occur in the containment with regard to such a system shall be taken as a design pressure.

6.5 The containment shall be designed as to withstand the design pressure specified in Para **6**.4 with regard to inertial forces at sea states.

Thermal stresses in the structure in case of emergency shall be considered.

6.6 Containment shall be operational when compressed due to action of external pressure and in case of flooding of the vessel/floating structure (see Para 3.5, Part VII "Machinery installations").

6.7 All means of closure, doors, stop valves/shut-off valves, cable passages sealing arrangements and other components included into tight circuit of the containment shall be designed, manufactured and tested on benches (prior to mounting onto containment) under supervision of the Register and according to approved procedures.

Tightness standards for tight circuit components shall be determined as per Appendix 3 and included into design documents. These standards shall be specified in the delivery specifications.

6.8 The constructed containment along with its means of closure shall be subject to hydraulic test at 1.1 times the design pressure (see Para **6**.4). Test pressure (P_{test}) shall be calculated by the following formula:

$$P_{test} = (1{,}1\sigma_T/\sigma_t)P_{design}, \qquad (6.8)$$

where σ_T: yield strength of containment material at test temperature,

σ_t: yield strength of containment material at design temperature (maximum temperature in case of maximum design-basis accident),

P_{design}: pressure in containment in case of maximum design-basis accident.

6.9 If hydrostatic pressure in the course of containment test exceeds the test pressure and results in the risk of damage of structure, equipment or their foundations, hydraulic tests may be replaced with air pressure tests. The containment shall be subject to air pressure test when tight circuit of the containment is completely installed. Test pressure P_{test} shall be determined by formula as shown in Para (**6**.8).

6.10 Containment shall be subject to leak test at pressure equal to the design pressure. In case of air pressure tests, the leak tests may be combined with the pressure tests provided that the test pressure (P_{test}) is brought to design pressure (P_{design}).

Procedures for testing and calculating relative leakage rate as well as Calibration Certificate for the measurement procedure shall be approved by the Register.

In case of hydraulic tests, the air test pressure in the containment in the course of leak test may be reduced provided that relative leakage rate shall be measured three times at test pressures of 0.07 MPa, 0.05 MPa and 0.03 MPa. Test procedures, Calibration Certificate for the measurement procedure and extrapolation of test results at decreased pressures to the design pressure shall be approved by the Register. Permissible relative leakage rate at design pressure shall be justified by the Designer with regard to radiation safely conditions as per applicable Radiation Safely Standards for the Personnel and People. It should be noted that decrease in permissible relative leakage rate down to **1**%/day and less as compared to the design maximum permissible value results in decrease in potential radiation hazard for the personnel and population in case of maximum design-basis accident. Therefore, it shall be established with possibility of reaching and measuring this value.

6.11 When the core is loaded into reactor and installation works are complete, the containment shall be subject to leak tests (outside and inside) at excessive air pressure of 0.05 MPa.

6.12 When the containment is subject to leak test, compressed air parameters within the containment shall be recorded (pressure, temperature) at least every hour until the validation criterion is met at $a \geqslant 0.95$ to be calculated with regard to inequalities:

$$\begin{cases} L_M + \zeta_L \leqslant L_P \\ \zeta_L \leqslant 0{,}3L_P \end{cases}, \qquad \textbf{(6.12)}$$

where L_M: measured relative leakage rate based on directly measured P, T, $\tilde{\tau}$ obtained in tests, %/days;

ζ_L: design measurement error of relative leakage rate, %/day;

L_P: permissible relative leakage rate specified in the design of nuclear-powered vessel/floating facility, %/day;

a: confidential probability.

6.13 The containment shall be subject to leak tests during vessel/floating facility operation (in the course of periodical surveys and after reactor core reloading). In such a case, test pressure shall be 0.05 MPa, permissible relative leakage rate shall be equal to measured relative leakage rate at initial test pressure of 0.05 MPa.

The test results assessment criterion shall be the condition when the inequality holds true:

$$L_D^{EX} \leqslant 1{,}15(L_M + \zeta_L), \tag{6.13}$$

where L_D^{EX} — permissible relative leakage rate at excessive pressure of 0.05 MPa to be controlled during vessel operation, %/day

L_M: measured relative leakage rate at excessive pressure of 0.05 MPa obtained during vessel construction, %/day

ζ_L: design measurement error at excessive pressure of 0.05 MPa obtained during vessel construction, %/day

1.15 factor which accounts for operating life of the vessel.

The measured relative leakage rate at initial excessive test pressure of 0.05 MPa shall comply with inequalities (6.12).

7 SHIELDING BARRIER

7.1 The containment and significant radioactive sources related to steam generating plant shall be surrounded by shielding barrier. Boundaries of the containment and shielding barrier shall not be combined.

7.2 All bulkheads, decks and other structures forming the shielding barrier shall be steel and watertight as required by **RS** Register Rules for similar structures.

7.3 Boundary bulkheads of reactor compartment and other compartments of vessel/floating facility may be used as forward and aft traverse bulkheads of shielding barrier.

7.4 Longitudinal bulkheads forming side walls of shielding barrier shall be B/5 or 11.5 m (whichever is less) apart from boards unless the other penetration height is specified for collision protection.

The **R**egister shall be provided with reasonable evidence that the damage will not be exceeded at design collisions.

7.5 Shielding barrier shall be subject to watertight test as per hull watertight test set-up.

7.6 Upon completion of mounting operations, shielding barrier shall be subject to leak test. The test procedures and standards shall comply with requirements to vessel's spaces.

7.7 In the course of operation of the vessel/floating facility, shielding barrier spaces are not required to be leak tested provided that the design pressure in these spaces is maintained to be below the atmospheric pressure.

7.8 The structures of shielding barrier shall be capable of being decontaminated.

8 REACTOR FOUNDATIONS. FASTENERS OF CONTAINMENT AND BIOLOGICAL SHIELDING

8.1 Reactor foundations and fasteners of containment shall provide effective support under external conditions as specified in Section **8**, Part II "Safety Standards". Section **8**. Part III "Safety Standards".

The foundations shall be capable of keeping the reactor and primary systems as well as containment at place in case of inclinations of the vessel/ floating facility up to and including capsizing.

8.2 The foundations shall be capable of withstanding thermal stresses.

8.3 The foundation structures shall be accessible for inspection as far as possible.

8.4 The fasteners of biological shielding shall be designed with regard to inertial forces acting on it as specified for safety class **2** and **3** equipment and to deformation of vessel's hull and exposure to excessive pressure in the containment (see Para **6**.4).

8.5 The structures of foundations shall be capable of being decontaminated, if required.

9 WELDED STRUCTURES AND JOINTS

9.1 When selecting design thickness of fillet welds of collision, grounding and stranding protection structures as per Part II "**H**ull" of the **RS R**ules, weld efficiency factor shall be taken to be 0.45.

Components of protection structures jointed with shell plating are to be of full penetration type.

9.2 All welded joints of containment structure shall be subject to non-destructive testing during vessel construction.

9.3 20% of welded joints of hull structures in way of the reactor compartment and structural protection shall be subject to non-destructive testing during vessel construction.

9.4 No intermittent welds are allowed in the controlled area.

PART V. SUBDIVISION

1 GENERAL

1.1 Subdivision of nuclear-powered vessels and floating facilities shall comply in full with requirements of Part V "Subdivision" of RS Rules and requirements hereof.

1.2 Nuclear-powered vessels and floating facilities shall remain afloat and have sufficient stability in case of damage specified in Para **2.1** under operational loading conditions of the vessel/floating facilities.

When calculating emergency grounding and stability, one shall consider that such a damage may occur anywhere along the length of the vessel/floating facility.

The nuclear-powered vessel/floating facility shall have sufficient buoyancy in case of flooding of at least two adjacent compartments.

1.3 In case of probability evaluation of subdivision as per Part V "Subdivision" of the RS Register, the R index shall be specially defined by the Register. Formulas for calculating Sc and Sm are selected as agreed upon with the Register with regard to structural features and assumed operation of the vessel/floating facility.

2 DAMAGED STABILITY OF NUCLEAR-POWERED VESSEL/ FLOATING FACILITY

2.1 Damage Dimensions

2.1.1 When calculating damaged stability, the extent of damage shall be assumed to be as follows:

.1 Side damages:

Longitudinal extent:	**1**/3 (where $L_L^{2/3}$ — vessel length (see Part V "Subdivision" of the RS Rules) or **14**.5 m whichever is less.
Traverse extent:	B/5 or **11**.5 m (whichever is less) measured from inner side of shell plating at right angle to the centerline at the level of summer load line.
Vertical extent:	from the base line upwards without limit.

.2 Bottom damages:

Extents	0.3L from forward perpendicular	Remainder ofi the vessel
Longitudinal Traverse Vertical	$1/3L_L^{2/3}$ or 14,5 m[1] $B/6$ or 10 m[1] $B/15$ or 2 m[1]	$1/3L_L^{2/3}$ or 5 m[1] $B/6$ or 5 m[1] $B/15$ or 2 m[1]
[1]Whichever is the least.		

2.1.2 With regard to collision, grounding and stranding protection in way of the reactor compartment (see Section 5, Part IV "Hull"), the Register may accept less extents other than those specified in Para **2.1.1**.

2.2 Permeabilities

2.2.1 Permeabilities mentioned in Part V "Subdivision" of the RS Rules shall be applied in assessment of damaged stability.

Permeability for cargo holds is taken to be 0.8.

2.2.2 Permeabilities for the steam generating plant spaces shall be determined with regard to actual flooding of these spaces.

2.3 Requirements To Stability Elements Of Damaged Nuclear-Powered Vessel/Floating Facility.

2.3.1 Heel angle at a final stage of asymmetric flooding shall not exceed 15° before measures on righting are taken (actuation of valves fitted on crosspipes). This angle can be increased up to 17° provided that bulkhead deck is not submerged.

2.3.2 Stability at a final stage of flooding is considered to be sufficient if the righting lever curve (GZ curve) has a range of at least 20° at maximum righting lever of a least 0.2 m within the range specified. Area under the GZ curve within the same range shall be at least 3.5 cm*rad.

The vessel shall be capable of maintaining sufficient stability at the intermediate stage of flooding.

2.3.3 Crosspipes shall not be considered as arrangements to fulfill requirements of Para **2.3.1** and **2.3.2**.

2.3.4 The asymmetric flooding shall be kept to minimum by using effective heel stabilizing devices.

Spaces connected by means of large section ducts may be considered as common.

2.3.5 Systems to be applied for stabilizing heel angles, whenever reasonable and practicable, shall be capable of automatic operation.

The cross piping, if any, shall be operable from a position above the bulkhead deck.

3 INFORMATION ON EMERGENCY GROUNDING AND STABILITY

3.1 Information on emergency grounding and stability as required by Part V "Subdivision" of the RS Rules shall contain information for the Master as regards actions in case of damages greater than those specified in Para 2.1. Consequences of flooding caused by hull breach with depth to the centerline (for areas outside the reactor compartment) shall be considered.

PART VI. FIRE PROTECTION

1 GENERAL

1.1 Fire protection of the nuclear-powered vessel and floating facility shall comply with Part VI "Fire Protection" of the RS Rules for passenger ships carrying up to 36 passengers, and requirements hereof.

2 STRUCTURAL FIRE PROTECTION

2.1 Reactor compartment shall be separated from adjacent spaces by means of cofferdams or class A-60 bulkheads to ensure protection against external fires and explosions.

2.2 Only non-combustible materials shall be used in the reactor compartment and spaces where equipment for safe operation of steam generating plant is located.

Combustible materials may be permitted as an exception unless if they can not be replaced with non-combustible ones. Such an exception shall be specially considered by the Register on a case-by-case basis.

2.3 Spaces within the shielding barrier where combustible materials are used or containing installations which require the use of combustible materials (except for cables and paint materials for painting spaces) shall be enclosed with class A-60 structures.

Passages of pipelines and electric cables in the shielding barrier shall ensure gastightness and fire resistance equivalent to those for shielding barrier structure.

2.4 Trunks and vent ducts to the space bounded by containment/shielding barrier shall be insulated to A-60 standard lengthwise within these spaces and outside them for a length equal to the duct maximum section.

If trunks and vent ducts are fitted with fire dampers capable of automatic closing in case of fire and complying with Part VII "Systems and Piping" of the RS Rules, they may be insulated to A-0 standard.

2.5 Double bottom tanks located in the reactor compartments shall not contain fuel.

Where double bottom tanks containing fuel are provided forward or aft of the reactor compartment, they shall be separated from double bottom space of the reactor compartment by means of cofferdams with structural components to comply with Part II "Hull" of the RS Rules.

3 FIRE-FIGHTING EQUIPMENT AND SYSTEMS

3.1 Water shall not be used as a fire-extinguishing medium in spaces within the containment.

3.2 Nuclear power unit control stations shall be fitted with fire extinguishing systems as per Part VI "Fire Protection" of the RS Rules.

4 FIRE ALARM SYSTEM

4.1 In addition to requirements of Part VI "Fire Protection" of the RS Rules, nuclear-powered vessels and floating facilities shall be fitted with fire alarm system located in spaces of the containment, shielding barrier and control stations.

Fire detectors capable of actuating based on ionizing radiation are not allowed in high-level radiation spaces.

5 FIRE FIGHTING APPLIANCES

5.1 Containment spaces shall be equipped with CO_2 fire extinguishers as mentioned in Part VI "Fire Protection" of the RS Rules.

Spaces of central control station and shielding barrier shall be equipped with CO_2 fire extinguishers as mentioned in Part VI "Fire Protection" of the RS Rules.

5.2 The vessel shall carry emergency escape breathing devices in amount sufficient for members of damage control team + one emergency escape breathing device for training purposes.

PART VII. MACHINERY INSTALLATIONS

GENERAL

1.1 Nuclear power unit shall comply in full with all the requirements of Part VII "Machinery Installations" in the **RS** Rules and requirements of this Part.

1.2 The provision is to be made that at astern power of propulsion unit the vessel/floating facility doesn't exceed the distance braking at full speed forward as specified in technical assignment for the vessel/floating facility design. The same is to be checked during sea trails of the vessel/floating facility.

1.3 Nuclear power unit shall be capable of starting from power sources of the vessel/floating facility.

1.4 Nuclear-powered vessel and self-propelled floating facility equipped with one reactor shall have a stand-by power source to provide vessel/floating facility movement, steam generating plant cooling in case of its failure as well as to provide normal habitable conditions, steerability, buoyancy, fire safety, vessel signals and communication, escape routes and operation of boat winches. This stand-by power source shall provide the following:

.1 It shall be ready and provide sufficient power for safe operation of the vessel /floating facility in harbor and maintain steerability at sea equal to wind force of Beaufort scale 6 under any normal loading conditions;

.2 It shall be ready when the vessel/floating facility is in restricted waters or areas of intense navigation;

.3 It shall not depend on steam generating plant;

.4 It shall be placed outside the reactor compartment.

1.5 Non-self-propelled floating facility shall have a stand-by power source to cool steam generating plant and provide normal habitable conditions, fire safety, buoyancy, vessel signals and communication, escape routes.

2 OPERATION AT HEELS AND TRIMS

2.1 Main and auxiliary mechanisms shall remain operational under conditions specified in Table **2.1**. For particular type of nuclear-powered floating facility, operability conditions of mechanisms may be set as agreed upon with the **R**egister.

Table 2.1

No.	Conditions	Machinery and systems providing operation of steam generating plant	Main and auxiliary machinery	Emergency machinery and equipment
1	Long-term heel (degrees)	30	15	22,5
2	Roll (degrees)	45	22,5	22,5
3	Long-term trim (degrees)	10	5	10
4	Pitch (degrees)	15	7	10
Note: If proper justification is provided the Register may reduced requirements specified in column 3. In this case the reduced requirements shall be mentioned in Information on safety.				

3 STEAM GENERATING PLANT COMPARTMENT

3.1 Steam generating plant compartment shall be located in a manner to reduce to minimum the probability of damage to steam generating plant in case of nuclear-powered vessel/floating facility collision with another vessel or in case of grounding and stranding.

It is recommended that steam generating plant be located in the middle part of the vessel/floating facility.

Transverse distance from shell plating to shielding barrier of steam generating plant is specified in Para 7.4 and height of double bottom in area of the reactor compartment is specified in Para 5.5 Part IV "Hull'.

3.2 Steam generating plant and its components with radioactive substances shall be enclosed in containment (see Section 6 Part IV "Hull").

3.3 Passages of pipelines and electric cables through containment shall be minimized. These passages shall withstand conditions resulting in containment under state classes SC1 to SC4.

Layout and structure of these passages shall allow their surveys and local leakage tests.

3.4 All pipelines connecting internal volume of the containment with shielding barrier compartments or with atmosphere shall be provided with shut-off valves. Valves shall be located outside containment as close to it as possible. They shall automatically cut off the containment and be provided with remote control.

Containment cut-off means as safety system shall comply with single failure criterion.

3.5 Containment shall be provided with facilities for automatic external and internal pressure balancing in case of flooding of the vessel/floating facility. Structure of these facilities shall be approved by the Register.

3.6 Special facilities shall be provided for periodic inspections and tests of containment in service to determine integral leakage.

3.7 In addition to hatch for fuel loading, special hatch shall be provided for personnel access to equipment in containment. This hatch shall maintain gas tightness of containment at state classes SC1 to SC4.

Containment shall be also provided with escape manhole.

3.8 For pressure reducing system in containment at emergency release, see Section 5 Part IX "Special Systems".

3.9 For containment ventilation, see Section 6 Part IX "Special Systems".

4 ARRANGEMENT OF MACHINERY AND EQUIPMENT OF STEAM GENERATING PLANT

4.1 For allocation of steam generating plant machinery and equipment important for safety, their protection at internal and external emergencies shall be considered.

Components and systems of safety classes 1 and 2 as well as systems and storage facilities with radioactive environments and waste shall be located within protection against collision.

4.2 It is required to provide shielding Machinery and equipment which may be dangerous for steam generating plant in case they are damaged and broken into fragments.

5 STEAM GENERATING PLANT CONTROL STATIONS

5.1 Central control station for reactor shall be located in less vulnerable place (against fires, explosions, flying fragments, radioactivity, etc.) but as close to the reactor and machinery installation as possible in order to reduce length of control circuits. Central control stations shall be provided with at least two exits for people escape into lifeboats or fire safe places.

5.2 Emergency cooling control station shall be located at sufficient distance from central control station in order to avoid damage in case of fire or any other emergency in central control station.

Emergency cooling control station may be functionally connected to the bridge (see Para 19.17 Part VIII "Nuclear-Powered Steam Generating Plants").

6 SPECIAL REQUIREMENTS TO FUEL SYSTEM OF STAND-BY AND EMERGENCY DIESEL GENERATORS

6.1 Fuel system shall be designed so as similar-type failure shall not cause failure of all generator sets.

6.2 Daily service fuel tanks shall be placed as close to diesel generators as possible.

6.3 Stand-by and emergency diesel generators shall use the same fuel. Fuel storage tanks shall allow its mutual transfer.

6.4 Stand-by diesel generators shall have enough fuel to provide operation at full load considering expected length of vessel/floating facility voyages.

6.5 Fuel in emergency diesel generators shall provide operation for at least 30 days after any emergency state including **SC4**.

PART VIII. NUCLEAR-POWERED STEAM GENERATING PLANTS

1 APPLICATION

1.1 This Part covers requirements for vessel's two-circuit nuclear steam generating plants with pressurized water reactors.

Requirements to vessel's nuclear steam generating plants with other reactors on board shall be specially defined by the Register.

1.2 The Register may also apply the requirements of this Part in accordance with provision in force to equipment other than that specified in Para **2.2**.

2 SCOPE OF TECHNICAL SUPERVISION

2.1 For general provisions of classification and surveys of steam generating plant, see Part II "Classification".

2.2 Machinery, equipment and systems of the steam generating plant subject to technical supervision are given below:

.1 Reactors (cases, covers with their fasteners, piping attachments, removable and non-removable parts, safety devices and valves, supporting structures);

.2 Cores (fuel elements, burnable poisons, displacers, working and permanent neutron sources and their assemblies);

.3 Arrangements for control, testing and suppression of chain reactor (rods, protective liners, drives and actuators, ionization chambers with suspensions, thermocouples and resistance thermometers, level gauges);

.4 Machinery (pumps, compressors, fans);

.5 Safety valves and devices, valves and fittings of equipment, machinery and systems;

.6 Pressure vessels and units (metal-water shielding tanks), steam generators, pressure compensators, hydraulic chambers, ion-exchange and electromagnetic filters, heat exchangers and refrigerators, drainage containers, gas and air tanks, hydropneumatic cylinders);

.7 Systems:

Primary coolant circulation system;

Primary coolant purification system, primary coolant make-up system;

Residual heat removal system;

Core emergency cooling system;

Primary coolant sampling system;

Deaeration system;

Primary water drainage, storage and distribution system, pressure compensation system;

High pressure gas system;

Secondary coolant (from steam generator to the secondary circuit);

Fresh water cooling system (equipment and protection system);

Sea water cooling system (equipment);

Ventilation system for steam generating plant spaces and controlled area spaces;

Sorbent storage and handling system;

Explosive mixture removal and hydrogen content monitoring system;

Automatic equipment operating water and fitting control system;

.8 Reactor control and protection systems and arrangements;

.9 Reactor control and alarm systems and arrangements;

.10 Control, protection, monitoring and alarm arrangements for steam generating plant systems and devices;

.11 Survey facilities;

.12 Facilities for handling and repair of steam generating plant machinery.

3 TECHNICAL DOCUMENTS

3.1 The steam generating plant design documents to be submitted to the Register for approval shall include the following:

.1 Description with basic specifications, technical assignment and delivery specifications for steam generating plant;

.2 Memorandum;

.3 General arrangement drawings for steam generating plant;

.4 Operation modes for steam generating plant.

.5 Emergency modes of steam generating plant including the following:

Reactivity change accident analysis;

Analysis of heat removal accidents followed by coolant loss;

Safety systems reliability design analysis;

Safety analysis by strength conditions;

.6 Quality assurance program for steam generating plant;

.7 List of steam generating plant equipment comprising information regarding its approval by the Register;

.8 Schematic diagrams for steam generating plant systems;

.9 Feasibility study on steam generating plant safety;

.10 List of facilities for survey of steam generating plant equipment;

.11 Ways of handling of fuel assemblies and cores and handling equipment;

.12 In case of steam generating plant equipment specified in Para 2.2.1 to 2.2.6 which is not previously approved by the Register, along with the steam generating plant design the following technical documents shall be submitted to the Register:

General arrangement drawings with sections and drawings for major parts;

Memorandum or description;

Strength calculation results;

Delivery specifications/draft delivery specifications;

Delivery- Acceptance Trials programs for prototype and serial equipment.

.13 Memorandum for core physical and thermo hydraulic calculations.

Prior to starting manufacture of steam generating plant equipment specified in Section 2 detailed design documents shall be submitted to the Register for approval.

4 DESIGN CRITERIA

4.1 To ensure safety of steam generating plant at its operating and emergency states, basic design criteria mentioned in Section 6, Part III "Safety Standards" shall be observed.

5 SAFETY AND DESIGN CLASSES

5.1 As stated in Table According to Section 3, Part III "Safety Standards", equipment, machinery, systems and devices of steam generating plant shall be divided into four safety classes depending on their importance for the vessel's safety.

Classification below is given for indicative purposes.

5.2 The following components of steam generating plant are of Safety Class 1:

.1 Reactors, core supporting structures, fuel assemblies, pressure vessels and other primary components including systems and piping which failure may result in emergency states SC 3 and SC4 (see Section 6).

Equipment and piping associated with reactor cooling system and forming part of the primary circuit of reactor cooling are not required to meet requirements of safety class 1. In this case there should be provision for disconnecting and cooling them conventionally by making up leakages using primary make-up system only in case of design basis failure in equipment or piping under normal operation of the reactor or there should be provision for disconnecting equipment/piping from reactor cooling system by means of two valves. Each open valve shall be ready for automatic closure. The time of closure for the valve shall be as to ensure its operability and possibility of disconnection and conventional cooling of reactor in case of design-basis failure in equipment/piping at normal operation of the reactor.

.2 Steam generator and secondary piping including shut-off valves fitted on the main steam line and feed-water piping.

.3 Reactor emergency protection system including reactor control and protection system drives and monitoring system sensors which generate emergency protection signal and also produce and implement the steam generating plant control algorithm according to emergency protection signals.

.4 Primary circulating pump and its cooling pipelines including shut-off valves.

5.3 The following components are of Safety Class 2:

.1 primary circuit components which are not part of Safety class 1;

.2 equipment and systems required for the following:

Residual heat removal from the reactor core in case of SC2, SC3 and SC4

Monitoring the release of radioactive materials within the containment.

Suppression of excessive hydrogen content within the containment after the accident followed by major leakage/loss of primary coolant

Reactor core cooling and/or decompression in the event of accident (residual heat removal system and core emergency cooling system including emergency power supply, hydro pneumatic cylinders, coolant tanks, etc)

Reactor core cooling and/or decompression in the event of accident followed by coolant loss

Making up leakages of primary coolant (make-up system)

Performing any other functions which may be of similar importance for safety

.3 Steam generating plant control and monitoring system;

.4 Power supply systems and equipment for steam generating plant control systems and reactor control and protection system;

.5 Containment air purification system from the primary circuit to the containment;

.6 Overpressure protection means and system for removing primary coolant from safety valves not related to Safety Class 1.

5.4 The following components belong to Safety Class 3:

.1 Any safety systems of steam generating plant or their components not related to Safety Classes 1 and 2;

.2 Auxiliary systems for maintaining safety systems: Lubricating oil systems, hydraulic systems, sea water cooling systems (equipment), compressed air systems, emergency power supply fuel system for core emergency cooling system;

.3 Sea water cooling system performing safety functions to meet the basic design criterion B;

.4 Systems which are not directly associated with safely but failure of these systems may result in release of radioactive materials into environment and normally requiring appropriate waiting time to reduce radioactivity;

5.5 Safely Class 4 components are given below:

.1 Feed-water and secondary vapor system downstream of the second shut-off valves not related to Safely Classes 2 and 3;

.2 Turbines, condensers and turbogenerators not related to Safely Classes 1, 2 and 3;

.3 Other equipment which failure may directly lead to SC2;

5.6 The system and its components of appropriate safety class shall be related to corresponding design class (DC 1 to DC4).

Every design class provides specific design, manufacture and test standards based on consequences of failure for vessel's safely.

Numbers of design classes are not required to correspond to safely classes.

5.7 Higher level design standards and tougher requirements for quality control shall be applied to equipment of design class 1. The following basic provisions shall be observed.

5.7.1 Strength shall be calculated in accordance with standards approved by the Register. The following shall be taken into account for calculations: Stable pressure loads including test pressure loads

Variations in pressure during startup, operation and deactivation, pressure fluctuations caused by vessel roll in a seaway, constant and variable thermal loads

Dynamic loads in accidents followed by coolant loss acting on supporting structures and internal components of the reactor

Dynamic forces caused by pipe rupture with two-sided coolant escape

Dynamic forces caused by any design-basis accidents of SC 3 and SC4

Vessel's vibration effect

Loads at vessel/floating facility static heel of max. 30°, roll angles of max. 45° and trim of max. 10°

Support reaction forces at roll of the vessel/floating facility due to accidents followed by pipe rupture, actions of quick-closing valves and vibration under way shall be determined in the course of strength calculations.

If resonance oscillations due to vibration under way are excluded, these calculations are not required. In this case, appropriate evidence shall be submitted to the Register.

Static strength, brittle fracture resistance, low-cycle and radiation service durability of equipment components shall be evaluated.

5.7.2 When calculating strength, impact on material subject to radiation exposure and ageing processes (deformation and thermal) during operation of installation shall be taken into account.

When selecting materials, the following shall be taken into account:

Their physical, chemical and mechanical properties (ductility, strength, brittle-to-ductile transition temperature, intergranular corrosion susceptibility, weldability, radiation resistance, etc)

Force actions under operating conditions (alternating loads, shocks, vibrations, pressure, temperature, radiation exposure, working fluids corrosive action, etc)

The materials to be applied shall be approved by the Register.

Design requirements are give below:

Pressure vessels shall be welded with full penetration, holes and flanges shall be fitted with stiffeners to prevent unacceptable stress concentrations

5.7.4 Manufacturing process and quality control:

All components of Safety Class **1** shall be manufactured as per the approved procedure.

During manufacture all welds shall be subject to non-destructive testing. All components shall be also subject to non-destructive testing in the required scope in order to detect surface and internal defects and cracks. Test results shall be recorded into logbooks and operating documents and shall be further used for assessing the equipment state during non-destructive testing when surveying ships in service.

All pressure vessels as well as pressurized bodies of pumps and engines shall be subject to hydraulic test upon completion of manufacture;

Cavities and surfaces shall be clean and be checked for hygiene as per approved standards.

5.8 Higher level design standards and tougher requirements for quality control shall be applied to equipment of design class **2**. The following provisions shall be observed

5.8.1 Strength shall be calculated in accordance with the RS Rules or documents approved by the Register.

Structures and their supports shall be capable of withstanding static and dynamic loads due to variations in operating parameters and vessel motion in a seaway.

Pipelines with working fluid temperature of **120**°C and above shall be capable of withstanding static pressure and temperature loads with factors accounting for dynamic loads due to vessel rolls under different loading conditions. Pipelines of small diameter shall meet requirements of Part VIII "Systems and Piping" of the RS Rules.

5.8.2 Materials shall be selected, tested and surveyed to meet requirements of these Rules and the RS Rules.

5.8.3 Pressure vessels and pipelines shall be designed, manufactured and tested as per requirements of the RS Rules or approved provisions for high-temperature steam piping.

5.9 Equipment of design class **3** shall meet requirements of the RS Rules applicable to boilers, heat exchangers and pressure vessels.

5.10 Equipment of design class 4 shall be in compliance with design, manufacture and test standards approved by the Register with regard to inertial forces affecting this equipment.

5.11 Cyclic loads shall be taken into account in the design process of steam generating plant equipment.

Assessment of effect of every accident and every test shall be conducted to identify the remaining safely service life for the primary equipment with respect to cyclic loads.

6 STATE CLASSES

6.1 When designing steam generating plant, arrangements to ensure its safety and reliability shall be provided at regulated state of steam generating plant ans vessel/floating facility as well as according to weather and other environmental effects.

6.2 Four classes (see Section **3**, Part III "Safely Standards") are established to assess the state of steam generating plant (including emergency one) depending on frequency and consequences of events/faults and failures for equipment to be considered in the steam generating plant design.

6.3 SCI: normal state when steam generating plant may be operated in any prescribed mode. In such a state, failures in some equipment components may occur. These failures do not affect plant safely operation and do not impose any restrictions to operation of the installation.

The following operating modes of steam generating plant are included into SCI:

.1 Startup;

.2 Operation at prescribed power;

.3 Mooring trials and sea trials;

.4 Routine preventive inspections and maintenance;

.5 Variable modes;

.6 Exposure to bad weather conditions;

.7 Sorbent handling;

.8 Stoppage;

.9 Neutronic and thermohydraulic measurements;

.10 Recharging of reactor core.

6.4 SC**2**: state of steam generating plant in any prescribed mode. In such a state, there may be failures or malfunctions of equipment due to some faults or operator's errors imposing timing constraints on steam generating plant operation (power reduction or short-term deactivation).

SC**2** includes steam generating plant operating modes in case of occasional failures in equipment or scheduled actions during such abnormal operating conditions including:

.1 Failure or malfunction of machinery or device which results in variation in primary coolant parameters/maneuverability of vessel/self-propelled floating facility, for example: Shutdown of power generator, turbine, condenser, fresh water heat exchanger, termination of sea water supply, closure of valves on the main pipeline, failure in the main electric system, shutdown of feed-water pump;

.2 Unintended startup of feed-water pump/primary circulating pump;

.3 Change in core reactivity as a result of cold water supply;

.4 Sticking of one or more control valves of reactor control and protection system or failure in emergency protection drive;

.5 Reactor emergency protection actuation;

.6 Shutdown/failure in primary circulating pump when other pumps are operational;

.7 Failure in control (turbine, feed-water, water flow regulators, etc);

.8 Minor leakage in primary coolant circulating system;

.9 Actuation of secondary safely valve.

6.5 SC**3** is defined as an emergency state of steam generating plant which may require its deactivation. The following rare accidents are included into SC**3**:

.1 Failure in tightness of primary coolant system which results in pressure drop in the system and requires such measures as containment isolation, primary circuit make-up and reactor deactivation;

.2 Termination of forced circulation of primary coolant;

.3 Failure in secondary feed-water supply;

.4 Grounding and stranding of vessel/floating facility with no failure in heat removal from reactor in case of intact vessel/floating facility;

.5 vessel/floating facility collisions followed by flooding of two adjacent watertight compartments;

.6 Fire/explosion on board the vessel/floating facility which does not result in damaged reactor compartment;

.7 Fire in engine compartment/central control station;

.8 Accidents due to rare bad weather conditions and natural disasters in the planned navigation area/docking area of the vessel/floating facility. These accidents are too unlikely to occur;

.9 Temporary blackout of the main electric system.

6.6 SC4 is defined as a very occasional severe emergency state of steam generating plant requiring its urgent deactivation SC4 includes very occasional accidents where some power sources are capable of operation on board the vessel/floating facility:

.1 Accident followed by integrity failure/depressurization of fuel element cladding, failure in heat removal and primary coolant loss;

.2 Grounding and stranding of vessel/floating facility followed by periodic loss of capability of heat removal to coolant;

.3 Extremely occasional severe weather conditions and natural disasters;

.4 Grounding or stranding of vessel/floating facility with local damage to double bottom over its height or with the long-length damage to the bottom;

.5 vessel/floating facility collisions followed by fire and/or explosion on board the vessel;

.6 Rupture of the main steam line/steam line within the shielding barrier;

.7 Flooding of vessel/floating facility in shallow waters (up to the upper deck);

.8 Helicopter crash in the area of the reactor compartment and/or nuclear fuel storage facilities.

6.7 According to Para 2.2, Part III "Safety Standards", the effects of extremely occasional events followed by total loss of operability of all vessel's power sources (capsizing, flooding, grounding and stranding with the heel above **30°**) shall be considered in the design. The effects of these events are not governed by these Rules.

7 ACCIDENT ANALYSIS

7.1 Analysis of possible accidents shall be performed for SC2 to SC4. The analysis results shall be specified in designs of steam generating unit and vessel/ floating facility and presented in Information on Safely.

7.2 Analysis of possible accidents shall be approved by the Register and shall include the following:

.1 Conditions at the beginning of an accident, initial data for analysis;

.2 Preventive measures, guidelines on systems and equipment being activated by steam generating plant protection systems including reactor control and protection systems and other measures to be taken by personnel;

.3 Data on analysis procedures, physical and mathematical models, experimental data and computer codes;

.4 Assumptions and theory of calculated radiation effects (for example, increase in primary coolant specific activity in case of failure in fuel assemblies cladding, efficiency of coolant purification, its leakage, radioactivity propagation factor, doses);

.5 Data for assessing propagation range of radioactive materials into surrounding air (radioactive materials emission height above the upper deck, weather conditions);

.6 Description of accident development including predicted representation of radiation and other effects;

.7 Measures to prevent failures in safety systems due to one reason;

.8 Measures to protect personnel on board the vessel/floating facility during accident.

7.3 For making assumptions on accident occurrence and sequence of events it is required to take into account provisions of Section 6 Part III "Safety Standards". These assumptions shall be based on the following:

7.3.1 Systems and arrangements specified in Para **10.7** of this Part shall remain operational in the event of single failure.

7.3.2 The stand-by subsystem of safety system shall not be considered as operational in the event of single failure in case it may be repaired during reactor operation as per its Operating Manual.

7.3.3 Protective arrangements shall automatically actuate upon start of reactor accident. If operator's actions are required, it should be considered that they are not possible within the first **30** minutes. Operator's actions shall not obstruct normal operation of protection systems. The steam generating plant shall be shown to be in safe condition when no operator's actions are taken for at least **30** minutes after accident.

7.3.4 If the results of event being considered cannot be envisaged in a definite manner, the appropriate safety factors shall be adopted in assessment of possible accidents.

7.4 When assessing effects of possible accidents, their long-term effects shall be also considered. They shall be specified in the design documents.

7.5 When analyzing possible accidents in steam generating plants, it is required to consider the events coming as result of accidents on board the vessel/floating facility. Despite of collision, grounding and stranding protection as required by Part IV "Hull", the following concepts shall be accepted in analyzing specific accidents in steam generating plant related to accidents on board the vessel/floating facility.

7.5.1 In case of collisions or grounding and stranding, the vessel gets damaged to the maximum extent as accepted in Para 2.1, Part V "Subdivision".

All equipment located within the damage reaches including equipment located in flooded spaces shall be considered as non-operational. Equipment designed specifically for underwater operations may be considered as operational if its power supply units are shown as remaining operational.

7.5.2 It is assumed that the vessel/floating facility is sunk with reactor disabled and is flooded up to the level above the upper continuous deck (flooding in shallow water). Shielding barrier and containment remain unflooded unless special-purpose equipment is provided for flooding these spaces at such a depth. Hydrostatic pressure stabilizing devices fitted on the containment, if any, may remain non operational and the vessel/floating facility may remain at inclinations as defined in Para 2.1, Part VI "Machinery Installations".

7.5.3 When the vessel is flooded at deep water, at least criterion A specified in Para 6.1.1, Part III "Safety Standards" shall be met.

Radioactivity shall be efficiently retained for a long period to ensure minimum possible release of radioactive materials by keeping at least one significant structural barrier of sufficient leak tightness and corrosion resistance around highly radioactive sources.

7.5.4 The development of flooding process in terms of timing shall be considered regarding that the reactor got plugged before immersion of the vessel/floating facility.

7.5.5 Horizontal components of shock loads due to collisions, grounding and stranding shall be determined based on analysis. Conclusions shall be given in the design document (see Para 5.2., Part IV "Hull").

7.5.6 According to Para 3.2, Part III "Safety Standards", capsizing of the vessel/floating facility shall be considered. The conditions of heat removal from the reactor core of capsized vessel/floating facility shall be analyzed and the results shall be given in Information on Safety.

7.5.7 Vessel grounding and stranding with heel as specified in Para 2.4, Part VII "Machinery Installations" shall be evaluated with regard to the following:

cLoss of ability to intake sea water through board and bottom openings;

.2 Grounding and stranding in tidal waters with regular interruption of sea water supply;

.3 Grounding and stranding of vessel/floating facility with heel above **30°** shall be considered in terms of possible effects not regulated by these Rules.

Fires and explosions on board the vessel shall be analyzed with regard to the following:

.1 It may be accepted that fire originates from a single source in any compartment with combustibles.

.2 The analysis shall reveal that appropriate structural fire protection, fire alarm and extinguishing systems which ensure sufficient protection of reactor safety system are provided.

.3 If cargo holds/tanks may be subject to fire/explosion hazards, these cases shall be analyzed and based on analysis results it shall be proven that reactor safety is not impaired.

.4 Collisions followed by fire and/or explosion shall be analyzed as well as impact of long-term fires on radiation safety shall be addressed;

.5 If the vessel/floating facility is fitted with helicopter, the effects of helicopter crash on board the vessel/floating facility shall be analyzed. It is to be proved that this accident followed by fire will not impair the safety of vessel/ floating facility.

7.6 Accidents in steam generating plant which may result in hazardous situation for people/environment on board the vessel/floating facility shall be classified by states and marked as main design-basis accidents

7.7 Accidents in equipment, machinery, systems and devices of steam generating plants included into SC**2**, SC**3** and SC**4** shall be analyzed. Particularly, the following cases shall be analyzed and results shall be specified in Information on Safely:

.1 Deactivation of any single reactor control or parts of reactor control and protection system from the core which are being driven by common drive or controlled from common control device with maximum possible speed at any initial state (cold/hot), in any conditions of subcritical or critical core regardless of its power;

.2 Leakage of primary coolant into second circuit via loose joints of piping of steam generator with regard to possible isolation of steam and feed-water lines after increase in activity in the second circuit Predicted dose rates in the engine compartment shall be specified in Information on Safely and in Operating Manual for the steam generating plant;

.3 Sticking of reactor control and protection system control valves in any position by height in the core and under the worst conditions for nuclear fuel burn-up or failure in core control rod drive;

.4 Unintended startup of primary circulating pump with injection of cold water into the reactor;

.5 Cold water supply to the reactor from make-up systems, feed-water systems or other sources with maximum possible water flow;

.6 Pressure increase in primary coolant system due to stoppage of vapor removal;

.7 Unintended decrease in neutron poison concentration in the primary coolant;

.8 Possible failures in reactor power control system;

.9 Loss of ability for sea water heat removal;

.10 Accidents followed by loss of primary coolant.

.11 Leakage of primary coolant out of storage for primary water drainage

7.8 When analyzing the loss of ability for heat removal, the following shall be taken into account:

.1 Stoppage of main turbine;

.2 Failure in main condenser without using auxiliary condenser unless it is operating/or in stand-by mode;

.3 Failure in the feed-water pump, closure of feed-water line or other failure in secondary feed-water line;

.4 Failure to use one of reactor cooling ducts when the vessel/floating facility is berthed.

7.9 Accidents followed by primary coolant loss shall be analyzed with regard to the following:

.1 Possible rupture of any primary pipe except for reactor body branches;

.2 Coolant loss rate through the assumed damaged pipe shall correspond to the two-sided coolant escape rate unless it may be proven that there is a sufficient restriction of broken pipe ends motion or other two-side coolant escape means are provided;

.3 Accident followed by primary coolant loss shall be considered as maximum design-basis accident with regard to the following:

Stresses in containment structure and its systems shall be within the specified limits and design pressure shall be taken with appropriate margin of estimated pressure.

Radiation effects shall be within those as specified in Part XII "Radiation Safety".

The reactor core and its fuel elements shall be capable of withstanding thermal and mechanical loads and possible deformations shall not exclude heat removal by circulating coolant.

Variation in position of the vessel/floating facility due to wind and sea states as accepted for SC1 and SC2 in the design process shall not impair actuation of coolant removal tanks from safety valves and pressure drop tanks.

.4 When analyzing the accident followed by coolant loss, the following initial or boundary conditions shall be considered:

One subsystem of emergency cooling system supplies the coolant to the damaged piping rather than to reactor body.

The second subsystem is being repaired (if core emergency cooling system may be maintained in service according to design).

Single failure may occur in the operating system.

The reactor is switched off and maintained in safe condition for **30** minutes after the event is originated.

With steam generating plant automatic and remote control system the operator may activate safety systems.

Chemical reactions (for example, hydrogen and zirconium reactions) proceed.

Only those systems remain operational which are specially designed for operation in case of accidents with coolant loss.

7.10 Whenever necessary, loss of secondary vapor/feed-water after main steam line/feed-water line is completely ruptured shall be considered as the main design-basis accident. In any case, impact on such an accident on the reactor shall be evaluated and described in Information on Safety.

7.11 Failure in active component or control error of radioactive waste treatment system shall be addressed. This failure/error shall not impair safety functions of the system for SC3 and SC**4**.

7.12 Analysis of impact of any failure in critical element of electric installation on steam generating plant shall be performed based on single failure criterion.

Total blackout of main electric installation shall be considered as the main design-basis accident.

8 POWER SOURCES FOR STEAM GENERATING PLANT

8.1 For requirements to power sources for steam generating plant, see Section X "Electric Equipment".

9 ENVIRONMENTAL EFFECTS

When designing steam generating plant, different environmental effects specified in Section **8**. Part III "Safety Standards" shall be analyzed.

10 GENERAL REQUIREMENTS

10.1 If the vessel/floating facility is fitted with two steam generating plants, they shall operate independently and ensure operation of nuclear power unit regardless of the other steam generating plant.

10.2 The equipment of steam generating plant shall be secured to prevent its displacement in case of variation in vessel's position up to and including capsizing.

10.3 The core emergency cooling system, residual heat removal system and reactor protection system shall be tested for capability of performing their intended functions. The operating reactor shall be tested without temporary deactivation of safety functions and failure in system operation.

10.4 Liquid and gas systems as well as pressure vessels shall be provided with arrangements required for the following purposes:

.1 Filling systems and vessels after initial installation, alteration or repairs;

.2 Initial pressure test;

.3 Overpressure protection;

.4 Regular inspections and pressure tests;

.5 System isolation;

.6 Survey procedure;

.7 Checking thermo dynamical parameters.

10.5 Automatically controlled system critical for operation and safety of steam generating plant shall be also equipped with manual local or remote controls.

10.6 Reactor safety systems shall be capable of automatic activation as soon as events requiring quick response start.

Automatically activated systems shall be capable of keeping reactor plant in safe condition for at least **30** minutes without operator's assistance.

Safety systems may be controlled manually provided that operator's error does not impair normal operation of these systems and does not impede proper actuation of protection means.

10.7 All safety systems shall meet the single failure concept. These systems are given below:

.1 Steam generating plant automatic and remote control, protection, monitoring and alarm system (in terms of safety functions)

.2 Residual heat removal system;

.3 core emergency cooling system;

.4 Containment isolation arrangements;

.5 Primary pressure rise prevention system;

.6 Containment pressure reduction system.

10.8 Activation time for stand-by equipment shall eliminate probability of accident in the installation.

The adequacy of the accepted redundancy for equipment shall be explained in the steam generating plant design.

10.9 Systems and piping of steam generating plant shall be properly secured under normal and emergency conditions. Structure of piping fasteners shall be as such to allow their thermal expansion, where necessary. The distance between piping, systems and fastening surface shall be as such to ensure their proper maintenance and repair.

10.10 The steam generating plant shall be capable of operating at decreased power in case of deactivation of parts of steam generators/steam generating sections, pumps and other equipment of steam generating plant as well as parts of pumps, heat exchangers and other equipment of steam-turbine unit.

10.11 Primary coolant shall properly circulate to ensure proper reactor cooling at any power as stipulated by operating conditions.

10.12 Steam generating plant equipment shall meet requirements regarding hygiene of its cavities and surfaces approved by the Register. Prior to assembly, in the course of assembly and workshop tests, onboard installation, testing and operation, parts, assemblies and items of steam generating plant shall be properly cleaned.

10.13 Equipment to maintain coolant hygiene and quality at a required level during operation of steam generating plant as per design standards shall be provided.

10.14 Filtering elements/substances in filters with radioactive working fluids shall be replaced by reliable shutdown of filters by means of double valves from the system under operating pressure.

10.15 The vessel/floating facility shall be fitted with water preparation equipment for steam generating plant. The quality of water shall meet the standards as specified for this steam generating plant.

10.16 The vessel shall be provided with equipment to maintain the primary pressure at a required level and its make-up as well as other auxiliary equipment for safe normal operation of steam generating plant in all operation modes.

10.17 It is required to envisage effective means for piping leak tightness continuous monitoring for each steam generator and effective means for deactivation of steam generators/steam generating sections in case of vapor and feed-water.

10.18 Equipment of steam generating plant shall be designed to withstand vibrations as per standards approved by the Register.

10.19 The list and amount of spare parts for machinery and equipment of steam generating plant shall be based on delivery specifications or supplier's specifications approved by the Register.

10.20 The list and justification for selecting emergency parameters of the installation at which reactor is stopped, shall be given in the steam generating plant design.

10.21 Systems and devices with possible formation of explosive mixture in hazardous concentrations shall be fitted with effective removal system or concentration reduction system.

10.22 Regulatory documents on welding of structures and equipment of steam generating plant and welded joint quality check shall be approved by the Register.

10.23 Steam generating plant equipment intended for technical supervision by the register shall be subjected to load test on Manufacturer's test bench as per procedures approved by the Register upon completion of manufacturing, assembling, adjustment and running-in. After these the equipment may be fitted on board the vessel/floating facility.

10.24 Prototypes of equipment shall be tested as per procedures which ensure testing for reliability, long-term operability and compliance with operating conditions.

10.25 The steam generating plant and its equipment installed on board the vessel/floating facility shall be subject to mooring and sea trials as per procedures approved by the Register.

10.26 Cases of machinery, equipment, devices, vessels and units shall be stamped as per Instructions on Stamping the Items Under Technical Supervision of the Register given in Part I "General Regulations for Technical Supervision" of the Rules for Technical Supervision During Construction of Ships and Manufacture of Materials and Products for Ships.

11 REACTOR CORE

11.1 Reactor core shall ensure continuous effective reactor operation in the specified operating and transient modes as well as interrupted operation when number of startups is not less than permissible value per core life.

11.2 Core components and structure shall be designed as to prevent reactor uncontrolled runaway and nuclear accident in all operating and emergency states of installation and vessel/floating facility.

11.3 The core shall be designed as to ensure effective displacement of devices required for its operation at permissible powers in case of reactor startups and stops.

11.4 When designing the core, permissible limits for damages to core components shall be specified and justified.

The core shall be designed as to prevent release of radioactive materials from core components in concentrations exceeding the specified limits during manufacture, testing, storage and operation in the reactor till total exhaust of energy.

11.5 Fuel assemblies as well as core control and protection components shall be designed as to account for material properties, exposure effects, physical and chemical processes, static and dynamic loads for all states of the plant, roll effects of the vessel/floating facility, manufacture tolerances and uncertainties in calculations, effect of deposits on heat-emitting surfaces on heat removal efficiency.

11.6 The structure, shape and dimensions of the core and its components shall allow for their effective cooling for SC1 to SC4.

11.7 There shall be safety margins for abnormal conditions of coolant consumption due to loss of power of circulating pumps or for other reasons.

11.8 To detect damages of core components, means for permanent monitoring of primary coolant radioactivity shall be provided.

11.9 When estimating core thermal loads, the appropriate inaccuracy in calculations shall be taken into account. Thermal margins shall be selected as operating restrictions. Estimated heat transfer values at limiting transient processes shall be verified experimentally.

11.10 The calculated distributions of coolant flow through fuel assemblies shall be provided. The calculations shall account for variation in coolant flow and heat transfer with the roll of the vessel/floating facility. Safety factors envisaged in calculations shall account for inaccuracy of similar calculations.

11.11 Calculation results/test data confirming availability or non - availability of vibration in the core and its supports due to coolant hydraulic flows shall be presented.

12 REACTOR

12.1 The reactor shall ensure effective and stable operation under operating conditions stipulated by the design for all design loads.

12.2 The reactor load shall be increased and decreased at a rate to ensure sufficient maneuverability of the vessel/self-propelled nuclear-powered floating facility.

12.3 The reactor, actuating controls, adjusters and protective elements shall be designed to prevent unintended variation in reactivity in the event of roll, heel, capsizing, vibrations, shocks and other prescribed dynamic loads.

12.4 The reactor shall be capable of being switched to subcritical state from any power for all positions of the vessel/floating facility including capsizing.

12.5 The reactor shall be designed to prevent free drainage of the coolant: All branches on reactor body shall be arranged at the level above the upper cut of the core.

12.6 The reactor shall meet applicable nuclear safety requirements in respect of marine reactors as agreed upon with Register.

12.7 The reactor shall be designed to allow for safety handling of the core.

12.8 The reactor shall be designed to allow for visual internal survey and survey by means of remote/non-destructive testing.

13 PRIMARY COOLANT SYSTEM

13.1 Equipment, pipelines, valves and fittings forming the primary coolant system shall comply in full with requirements as applied to safety class **1** and **2** equipment.

13.2 Means for detection of primary coolant leakage shall be provided.

13.3 The primary circuit shall be designed with sufficient safely factor to ensure ductility of walls under stresses due to operation, maintenance, testing and emergency conditions adopted in the design. The safely factor shall account for impact of operating temperature on walls and radiation effects on material properties as well as other effects available in these conditions.

13.4 When selecting materials and manufacturing procedures, the following shall be taken into account:

.1 Compatibility with working fluids;

.2 Corrosive and erosive action of coolant, washing and decontamination fluids;

.3 Forming of components with large half-life period;

.4 Impact of neutron exposure on material properties.

13.5 The primary circuit shall be fitted with automatic arrangements which prevent its overpressure. The choice of these arrangements is justified in the design. At least two of these shall be provided for installation of safely valves/devices. Fluid from actuation of these valves/devices shall be drained into overpressure protected container as agreed upon with the Register.

13.5.1 Safety valves shall have sufficient capacity to prevent pressure increase by more than **10%** as compared to the design pressure for all main design-basis accidents if one valve fails to actuate.

13.5.2 Usage of burst diaphragms instead of valves is not permitted.

13.5.3 No shutdown devices of safety valves opening and closing are allowed:

.1 Unless effective locking device capable of automatic opening of auxiliary discharge valve of appropriate capacity is provided;

.2 Unless the nuclear reactor protection system is fitted with shutdown devices upon the pressure increase signal.

13.5.4 Other equivalent safety valves may be used if the following criteria are met:

.1 Such valves are at least of equal efficiency as compared to safety valves;

.2 Risk is not increasing in quantity;

.3 Primary circuit remains intact for SCI to SC**4** and maximum stresses in the reactor body and in the whole primary circuit are restricted i.e possible stress will not exceed permissible stresses due to safety factor;

.4 State classes which include release of coolant into environment (fourth circuit) are taken into account;

.5 It is proved that Criteria A, B, C specified in Part III "Safety Standards" are met;

.6 Similar replacement is approved by the Register.

13.5.5 Evidence confirming that requirements of Para1**3**.5.**4**.**1** — **13**.5.**4**.5 are met shall be given as a part of Information on Safety.

13.5.6 Expansion tanks may be located beyond the containment if the spaces where they are located comply with Para **13**.5.**4**.**1** — **13**.5.**4**.5, s well as requirements to containment structure (see Section **6**, Part IV "Hull").

14 SECONDARY COOLANT SYSTEM

14.1 The secondary coolant system shall comply with Part VIII "Systems and Piping" of the RS Rules in addition to provisions specified in these Rules.

14.2 Steam generators with pipelines and fittings up to the second shut-off stop valve included shall comply with the same design standards and be of the same reliability as the primary equipment.

14.3 Steam generators with fittings under internal pressure shall be tested as per Table **2**.**2**, Part II "Classification".

14.4 Each reactor shall be provided with at least two steam generators or one steam generator with two separate sections capable of being switched off.

14.5 Secondary steam lines and feed-water pipelines shall be fitted with two shut-off stop devices with draining of water in-between into the protected contained within the controlled area. Stop devices shall be fitted as close as possible to the steam generators. At least one stop device fitted on steam and feed-water pipelines shall be capable of being remotely and locally controlled, the other devices may be locally controlled only. Steam and feed-water pipelines shall be fitted with one automatic stop device which actuates upon the signal of failure in leak tightness of steam generator piping.

14.6 Provision for washing steam generators shall be envisaged.

14.7 In case of multi-section steam generators provision for isolation and disconnection of the non-tight sections shall be envisaged.

14.8 Means for overpressure protection of secondary coolant system shall be provided.

14.9 Each steam generator (or assembly of steam generating units connected as to prevent isolation from each other), unless rated for the primary pressure, shall be fitted with at least two safety valves located upstream of the first shut-off valve. Safety valves shall comply with Part X "Boilers, Heat Exchangers and Pressure Vessels" of the RS Rules, as far as applicable.

14.9.1 Safety valves shall have sufficient capacity to prevent pressure increase by more than 10% as compared to the design pressure for all main design-basis accidents if at least one valve fails to actuate.

14.9.2 If leakage of coolant from primary to secondary circuit may result in actuation of safety valves of the secondary circuit, water through these valves shall be drained in a container located within the containment/shielding barrier.

15 REMOVING RESIDUAL HEAT FROM REACTOR

15.1 The equipment for residual heat removal from the core during normal/ emergency shutdown of reactor as well as during core handling and repair shall be provided.

The residual heat removal system shall comply with single failure criterion.

15.2 The residual heat removal system shall remain operational in the course and after all accidents on board the vessel except for the following:

.1 Capsizing of the vessel/floating facility;

.2 Flooding at a depth where it may be proven than heat may be removed by containment flooding.

15.3 The residual heat removal system shall remain operational for a period as specified during analysis of operating and emergency situations.

15.4 The residual heat removal system shall be effective and have sufficient capacity and safety margin for the following purposes:

.1 To ensure integrity of fuel elements cladding through core cooling in case of SC1 and SC2;

.2 To ensure core cooling in case of SC3 and SC4 and to prevent exceeded permissible exposure of persons and environmental pollution due to damaged claddings of fuel elements.

16 CORE EMERGENCY COOLING

16.1 Core emergency cooling system is a safety system.

16.1.1 This system shall comply with single failure criterion.

16.1.2 This system shall maintain, as far as practicable, integrity of fuel elements after maximum design-basis accident followed by reactor stoppage. Core coolant supply means shall ensure effective operation until residual heat removal means appear to be capable of removing the remaining continuous heat release of the core.

16.1.3 In case hydro pneumatic cylinders are used for core emergency cooling, they shall be fitted with safety valves, gas pressure and water level indicators. Sources for maintaining gas cushion in such vessels shall be provided.

16.1.4 All switches of core emergency cooling system other than the main switch, shall be mechanically locked in a position required for system operation.

16.1.5 The core emergency cooling systems shall be controlled from central control station.

16.1.6 Machinery, devices and fittings of core emergency cooling systems shall be accessible for testing and operability checks.

16.2 Emergency cooling systems in multiple-reactor steam generating plants shall be completely separated unless it is shown in the design that using their assemblies in different reactors does not impair the capability of this system to perform its intended functions.

17 SYSTEMS AND PIPING

17.1 Systems and piping of safety classes **1** and **2** shall comply with the following requirements.

17.1.1 Strength of systems and pipelines shall be estimated based on their safety classes as per procedures approved by the Register.

17.1.2 Design pressure and temperature shall be selected based on analysis of operating modes of steam generating plant.

17.1.3 Pipes and fittings shall be made of easy-to-weld, corrosion- and erosion resistant materials which are non-susceptible to intergranular corrosion and which maintain their strength and ductility when exposed to radioactive radiation during operation on board the vessel/floating facility. The materials shall be capable of being decontaminated.

17.1.4 System piping shall be made of seamless pipes.

See circular 846c **17.1.5** All pipelines shall be connected by welding. Flanged and union joints shall be specially agreed upon with the Register provided that welding is not possible.

17.1.6 Structure, welding and test of piping welded joints as well as welded joints of branches shall be performed as per welding provisions and rules for testing welded joints approved by the Register.

17.1.7 If non-radioactive fluid is to be supplied to the piping with radioactive fluid, the intake pipe shall be fitted with non-return and stop valves.

17.1.8 The system fittings and valves shall be fitted with welded neck flanges and typically with bellows sealing.

17.1.9 Piping shall be thermally insulated with regard to possibility of decontamination.

17.1.10 Materials and structure of thermal insulation shall be approved by the Register. Thermal insulation shall be of non-combustible materials.

17.1.11 Upon final treatment at workshop and installation on board the vessel/floating facility, pipes and fittings shall be subject to hydraulic test at the test pressure and to leak tightness test. Standards for hydraulic tests shall be agreed upon with the Register.

17.2 Systems and piping of safely classes 3 and 4 shall comply with Part VIII "Systems and Piping" of the RS Rules.

17.3 All pipelines penetrating the containment shall comply with Para 3.3 and 3.4, Part VII "Machinery Installations".

17.4 Fittings and valves of equipment, systems and pipelines of steam generating plant shall be fitted with local position indicators and legible nameplates. Remotely controlled fittings and valves shall be additionally fitted with devices for its local control from the point of its location. Fittings and valves to be controlled from central control station shall be additionally marked similarly to the console marking.

18 HEAT EXCHANGERS AND PRESSURE VESSELS

18.1 Heat exchangers and pressure vessels shall be designed with regard to their safely class.

18.2 Heat exchangers and pressure vessels of safely classes **1** and **2** except for the reactor shall meet the following requirements.

18.2.1 Strength of heat exchangers and pressure vessels shall be estimated as per procedures approved by the Register.

18.2.2 Sealing procedures for main connectors shall be approved by the Register.

18.2.3 Cases of heat exchangers and pressure vessels shall be adapted for hydrostatic tests.

18.2.4 Heat exchangers and pressure vessels except for the primary circuit, shall be protected against unacceptable pressure increase by means of safety devices, where necessary.

18.2.5 It is allowed not to install safety devices on heat exchangers and pressure vessels of safety classes **1** and **2** if they are connected to the vessel fitted with safety devices through non-isolated pipes.

18.2.6 Heat exchangers and pressure vessels shall be made of easy-to-weld, corrosion- and erosion resistant materials which are non-susceptible to intergranular corrosion and which maintain their strength and ductility when exposed to radioactive radiation during operation on board the vessel/floating facility. The materials shall be capable of being decontaminated and approved by the Register.

18.2.7 Structure, welding and test of welds of heat exchangers and pressure vessels shall be performed as per welding provisions and rules for testing welds approved by the Register. All welds shall be subject to non-destructive testing.

18.2.8 Upon manufacture and installation on board the vessel/floating facility, heat exchangers and pressure vessels shall be subject to hydraulic tests at test pressure and leak tightness test as per standards for hydrostatic test approved by the Register.

18.2.9 Prior to applying insulation/protective coating to heat exchangers and pressure vessels, they shall be subject to hydrostatic tests and other mechanical strength tests.

18.3 Heat exchangers and pressure vessels of safety classes **1** to **4** shall also comply with Part X "Boilers, Heat Exchangers and Pressure Vessels" of the RS Rules to the extent to which these Rules are complied with.

19 CONTROL AND PROTECTION SYSTEM

19.1 The control and protection system shall be provided for the following purposes:

.1 Continuous monitoring of the reactor operating state;

.2 Automatic and remote control of steam generating plant which prevents exceeding of reactor design specifications critical for safety;

.3 Automatic and remote control of reactor at a given power;

.4 Perception of emergency state signals and activation of systems and equipment critical for safety.

19.2 The control and protection system shall be redundant in terms of safety functions and be able to perform its intended functions assuming the single failure.

19.3 The control and protection system shall ensure proper control of reactor power as per operating demands of the vessel/floating facility for all operating maneuvers under normal and emergency situations and appropriate sea states. The control and protection system shall as far as practicable prevent operating restrictions for the nuclear-powered vessel/floating facility not applied to ships of similar dimensions with conventional propulsion plant of the same power.

19.4 The control and protection system shall receive signals from sensors of parameters measured via different channels, including neutron flow.

Parameters critical for reactor control shall not be measured via one channel.

19.5 The control and protection system shall be capable of being in-service tested without impairing safety.

19.6 The reactor adjusters shall be designed to ensure automatic and remote control of the reactor.

19.7 Means for testing operability of each channel of the reactor control and protection system and means for detecting faulty components shall be provided.

19.8 To detect faults/accidents in the reactor, at least two naturally different parameters characterizing the operating process shall be tested. In case it is unreasonable or impracticable, additional redundancy shall be envisaged for variable parameters in the test channel.

19.9 Devices of reactor protection system required for monitoring in SC3 and SC4 shall remain operational in such conditions.

19.10 The light signal shall be supplied in case of failure or damage to the channel of the reactor control and protection system.

19.11 Basic design provisions for reactivity control to be taken into account for design process.

19.11.1 Events which may result in unintended increase in reactivity shall be very occasional as specified in Section **3**, Part III "Safety Standards" and shall not result in situations more hazardous for the crew, population and environment than mentioned in Part XII "Radiation Safety".

19.11.2 The anticipated accidents of reactivity change shall not result in spontaneous chain reaction or depressurization of primary coolant system and impede shutdown of reactor.

19.11.3 When reactor is operating at power corresponding to running modes of the vessel/floating facility, reactivity factor shall be negative with regard to design roll and acceleration of the vessel/floating facility.

19.11.4 Control and protection system shall be capable of shutting down the reactor automatically when the vessel/floating facility is inclined up to angle of vanishing stability and maintain the reactor in such state at all angles. In addition, the control and protection system shall actuate automatically if the vessel/floating facility is sinking, or its static heel is **45°**, or its trim is **10°**.

In case of less static heel and trim angles, automatic actuation of control and protection system for reactor shutdown is not required.

19.12 The control and protection system shall meet the following requirements:

19.12.1 The system shall comprise at least two independent reliable reactivity control subsystems of different design.

19.12.2 One subsystem shall be mechanical and have the following features:

.1 Be capable of automatically switching the core to the subcritical state and maintaining it in a cold subcritical state without neutron poison within the core life considering that the most effective control of reactor control and protection system is extracted from the core and may not be inserted again;

.2 Be capable of effectively controlling variations in reactivity and preventing exceeding design restrictions on fuel specifications of the core for any operating and emergency design state;

.3 Contain devices for preventing unintended motions of any control of reactor protection and control system;

.4 Properly operate in case of failure in one stand-by channel which generate signals for actuation of emergency protection including measurements

.5 Reduce reactor power at a rate preventing exceeding any design restrictions upon receipt of the emergency signal;

.6 Display position readings of each neutron poison element on reactor control console;

.7 Be designed to reduce the possibility of unplanned continuous removal of control of reactor control and protection system from the core down to the acceptable level;

.8 Issue sequence of commands to actuators of control and protection system to minimize the possibility of operator's error;

.9 Be fitted with devices for preventing removing controls of control and protection system from the core by abnormal groups or in abnormal order.

19.12.3 The other reactivity control subsystem shall be capable of switching and maintaining the reactor core in a subcritical state.

19.12.4 Reactivity control subsystems shall remain completely operational for all design inclinations of the vessel/floating facility and ensure functional checks, regular calibrations of device within the measuring power range and tests for proper operation of devices.

19.12.5 Controls of the reactor control and protection system being inserted into the core shall be capable of maintaining the core in a subcritical state with sufficient margin within its entire life and after total exhaust of energy including periods of maintenance, fuel handling, emergency states of reactor and vessel/ floating facility including capsizing and flooding.

19.12.6 The reactor shall remain operational at powers sufficient for steerability of the vessel/self-propelled floating facility under SC1 within the specified energy of the core in case of sticking of the most effective control in the core at energy level of power and its failure to be removed from the core except for cases when the core is poisoned with xenon.

19.12.7 The reactivity control means shall be designed to ensure control from central control station and possibility of switching and maintaining the core in the subcritical state from the emergency cooling station.

19.13 To prevent unplanned variations in reactivity due to moderator density variation, means for estimation and control of power arbitrary fluctuations and variations within the reactor core shall be provided unless calculation results prove that such fluctuations are minimum and along with acceptable margins do not result in conditions where limited estimated specifications may be exceeded.

19.14 For SC2, control and protection system shall switch on the reactor after its short-term shutdown at the specified time to ensure maneuverability of the vessel/self-propelled floating facility without impairing safety.

19.15 Failure in any control shall not impede safe stoppage of the reactor.

19.16 The control and protection system shall be located as to ensure total monitoring and control of reactor for SC1, SC2, SC3 from central control station as well as reactor shutdown and control its state from the navigation bridge or from emergency cooling control station.

19.17 Reactor emergency cooling control station located away from the central control station shall be provided for the following purposes:

.1 Independent shutdown of the reactor The reactor may also be shutdown from the other continuously manned station;

.2 Possibility of further independent reactor cooling;

.3 Monitoring reactor state and primary circuit and maintaining the reactor in a cold state as well as indication on reactivity control position.

19.18 Measures to prevent effects caused by incorrect operator's actions shall be envisaged in the control system.

19.19 Where locking devices for emergency protection actuation are permitted by design, these locking devices shall be clearly marked on the reactor control station. Generally, locking devices for emergency protection actuation are not required in control and protection system.

19.20 In addition to automatic and remote control, drives of control and protection system shall be manually controlled directly from the point of their location. The direction of manual handle rotation and appropriate direction of motion of controls (control and protection system) shall be clearly marked.

20 INSTRUMENTS

20.1 The thermal technical control and alarm system including pressure, temperature, fluid level and consumption test shall be provided to ensure continuous remote and local control of system and equipment operation.

The system operation shall be checked from local and central stations by means of visual and audible alarm devices and arrangements. For the scope of alarms, indications and protection means, see Table **3.1**, **S**ection XI "Automation".

20.2 The steam generating plant shall be provided with instruments which ensure effective measurement of operating parameters and testing its operation.

20.3 The most critical parameters of steam generating plant required for its operation or giving possible causes of faults (such as reactor power, primary pressure and temperature, level in pressure compensators) shall be automatically recorded by appropriate instruments with time indication.

20.4 Instruments of control and protection system shall ensure continuous measurement of neutron flow (including reactor startup period) from minimum controlled power to the maximum design power of the reactor.

20.5 As far as reasonable and practicable, instruments of control and protection system shall be redundant and separated from the instruments designed for measuring parameters and testing operation of systems.

20.6 The instruments shall remain operational under extreme conditions of spaces and working fluid.

20.7 Instruments, cables and equipment shall be located and backup so as to ensure operability of instruments of reactor protection system in case of accidents on board the vessel/floating facility.

20.8 Indicating gauges shall be marked with limiting values and standard operating range.

20.9 **M**easuring channels for measuring critical parameters and safety-related equipment shall be fitted with built-in automatic self-test system.

20.10 Where necessary, visual and audible alarms actuating both in case of exceeding threshold parameters of operating steam generating plant and faults in instrument measuring channel.

20.11 The instruments shall be designed to ensure prompt and unified estimation of the plant state. Means for generalized data representation and prediction as well as sound and light signals shall be used provided that may contribute to improvement of operation and safely.

21 SURVEYS

21.1 Steam generating plant and its equipment shall be subject to technical supervision by the Register in the process of design, manufacturing and testing as well as during construction and testing, operation and repair of the vessel/ floating facility.

21.2 For the scope, intervals of surveys to be conducted by the Register as well as procedure for issuing documents by the Register, see Part I "Classification" of the RS Rules, Rules for the Classification Surveys of Ships in Service and these Rules as well as Guidelines on Technical Supervision of Ships in Service, Guidelines on Technical Supervision During Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Support Vessels and Manufacture of Materials and Products.

PART IX. SPECIAL SYSTEMS

1 SCOPE OF TECHNICAL SUPERVISION

1.1 This part contains the requirements for special systems of nuclear-powered vessels and floating facilities. The systems must meet the requirements stated in the **RS** Rules Part VIII Systems and Piping, unless otherwise specified herein.

1.2 The following systems are subject to the Register technical supervision;

.1 Controlled area drain system;

.2 Compressed air and hydraulic steam generating plant service systems;

.3 Containment pressure reducing system;

.4 Special ventilation.

2 GENERAL REQUIREMENTS

2.1 The pipelines outside the containment which contain or might contain radioactive substances shall be fitted with double shut-off valves and leakage detectors. In the pipelines with diameter more than 15 mm provision is to be made that one of shut-off valves is remotely operated and actuated automatically in when necessary.

2.2 As a rule there shall not be connections between vessel's general systems and systems which contain or might contain radioactive substances. If such connections are inevitable, they shall be fitted with double shut-off valves, and drainage is to be envisaged for the pipeline section between the valves.

2.3 The systems carrying radioactive media shall be fitted with glandless instruments and bellows sealed fittings.

3 SPECIAL BILGE SYSTEM FOR CONTROLLED AREA

3.1 The controlled area special bilge system is to be provided with means capable to prevent emission of radioactive fluids.

3.2 The controlled area special bilge system shall be independent of the vessel's bilge system. The system shall be designed for operation under state classes SCI to SC4. Compartments shall be drained into special containers, named montejus tanks.

3.3 The special bilge system pipelines within the controlled area shall be made of seamless pipes. All the connections of such pipelines shall be welded. Flange and union connections are permitted in case welding is not possible upon the Register's special approval.

3.4 The pipelines shall be made of corrosion-resistant materials.

3.5 The pumps and pipelines shall be provided with biological protection, if necessary.

3.6 The scuppers within the controlled area shall be fitted with mesh (lattice).

4 COMPRESSED AIR AND HYDRAULIC SYSTEMS FOR REACTOR SERVICE

4.1 The compressed air systems for service of critical auxiliary equipment of the steam generating plant or to be used for control purposes shall be supplied with air from two independent compressors, each compressor shall be capable of keeping the system operational.

4.2 Every compressed air system operating as a part of the reactor safety system shall include at least two separate air cylinders having capacity sufficient for system requirements.

4.3 Compressed air shall be cleaned and dried and its temperature shall be maintained at a specified level.

4.4 Requirements **4.1** и **4.2** are also applied in general to pumps and hydro pneumatic accumulators and to hydraulic systems to be used for service of critical auxiliary equipment.

5 CONTAINMENT PRESSURE REDUCING SYSTEM

5.1 If the project envisages the containment pressure reducing system to be actuated in case of emergency release of coolant out of the primary circuit, it shall be capable of maintaining operability in case the main electric generators fail.

5.2 The system shall remain in permanent readiness and be capable of automatic actuation if increase in containment pressure is above the specified limit.

If proper justification is provided remote actuation of the system may be permitted.

5.3 If sprinkler systems are used for pressure reducing, they shall arranged on the principle of hydrophore, except for the case, when it is proven that the system is actuated in time during which pressure in the containment does not reach critical values taking into account a single-failure criterion.

5.4 When installation on board the vessel/floating facility is completed the pressure reducing system and its components shall be tested in operation.

5.5 Provision is to be made for periodical surveys and trials for structure of pressure reducing system in operation on board the vessel/floating facility in service.

5.6 The pressure reducing system components (expansion tanks, bubbling chambers, etc.) may be located in compartments connected to the containment provided those compartments are similar to the containment structure as related to protection against emission of radioactive substances.

5.7 The pressure reducing system shall also meet the requirements specified in Para 5.3 and **10.7**, Part **VII** Nuclear-Powered Steam Generating Plants.

6 SPECIAL VENTILATION SYSTEM

6.1 The ventilation systems of controlled and supervised areas shall be isolated from each other and the rest of ventilation systems installed on board the vessel/floating facility.

6.2 Reduced pressure shall be maintained in the controlled area spaces where radioactive contaminations exist under state classes SC1 and SC2 even with one entrance opened.

Directed air flow shall be provided from spaces where probability of contamination is less into those spaces where probability of contamination is higher.

6.3 The containment ventilation system structure shall provide for closed-circuit and open-circuit running.

6.4 The containment ventilation system shall be fitted with automatic shut-off valves for quick closing air channels under emergency states (SC2, SC3 and SC4).

6.5 Air from the containment shall be discharged via channels equipped with radioactivity monitoring instruments and warning devices.

6.6 The containment is vented into atmosphere after SC3 and SC4 through special filters providing required rate of air cleaning.

6.7 Air from the controlled area spaces shall be discharged through a special mast.

6.8 Layout for ventilation air intake of vessel spaces shall be selected so as to prevent intake of discharged radioactive gases.

6.9 Exhaust and intake ventilation units of the spaces where radioactive contaminations occur or might occur shall be located in isolated enclosures.

6.10 The provision is to be made for redundancy of ventilation equipment for the controlled area. One of the backup fans shall be started automatically once the running fans fail.

PART X. ELECTRIC EQUIPMENT

1 GENERAL

1.1 Application

The electric equipment of nuclear-powered vessels/floating facilities shall comply in full with the requirements stated in the RS Rules Part XI, Electric Equipment and requirements herein.

1.2 Definitions and explanations

In this part the following definitions have been adopted.

The emergency electric system of vessel /floating facility is an electric system consisting of emergency generators and emergency switchboards independent of the main electric system and intended to supply electric energy to consumers important for safety of the steam generating plant and the vessel /floating facility as a whole, when the main and stand-by electric energy sources are not available.

The emergency electric energy sources are electric generators intended to supply electric energy to vessel's critical consumers when voltage at the main switchboards is not available.

The integrated electric energy system is a system consisting of propulsion, main and stand-by generators with associated driving motors, transformers, converters, distributors with power lines intended to supply electric energy to the propeller drives and all the vessel consumers.

The main electric system is a system consisting of main and stand-by electric energy sources and main switchboards intended to supply electric energy to both the steam generating plant consumers and all the vessel consumers.

The main electric energy sources are sources of electric energy required to maintain the vessel /floating facility under normal operating condition and normal habitability conditions with the steam generating plant running without engagement of the stand-by or emergency generators.

The uninterrupted power supply units are sources providing uninterrupted supply of electric energy to certain consumers when all the other electric energy sources do not operate.

The stand-by electric energy sources are electric generators independent of the steam generating plant to be used in cases of steam generating plant failure or in other abnormal situations instead of faulty main electric energy sources. These energy sources shall supply electric energy to the consumers ensuring safety of the vessel/floating facility and restore normal operational state for minimal habitability conditions, as well as for scheduled launch and cooling of the steam generating plant without engagement of the emergency generators.

The marine electric power plant is a set of primary motors and electric generators with the main switchboard intended to supply electric energy to all the vessel consumers in any operation mode of the marine vessel's energy system.

1.3 Scope of **technical supervision**

In addition to the equipment listed in the **RS R**ules Part XI, Electric Equipment, the equipment of installations and service systems of the steam generating plant is subject to the technical supervision.

1.4 Technical documentation

For requirements to technical documentation, see Part II "Classification".

2 GENERAL REQUIREMENTS

2.1 The electric installation of the vessel/floating facility shall consist of the main and emergency electric systems.

2.2 The electric installation with generators off shall be capable of supplying electric energy to the systems required for disabling the reactor and keeping it in safe state for at least during **30** days under any state class, including **SC4** and taking into account a single failure of the electric system in addition to an initial event which caused the state class.

2.3 When starting up the reactor and shutting down the reactor, the safety control systems and security systems of the reactor shall be supplied with electric energy from at least two independent sources.

2.4 The stand-by and emergency generators, in case on of them fails, shall supply electric energy to the consumers required for starting up the steam generating plant from cooled (or hot stand-by) state and maintaining minimal habitability conditions. The emergency generators may be used for starting up the steam generating plant, if they produce enough power, and for supplying electric energy to the consumers important for safely of the vessel or floating facility.

2.5 The main electric system shall be capable of providing reliable electric energy supply for the steam generating plant consumers and for all vessel's critical consumers from two electric stations at least in all operational and transient modes.

2.6 The provision is to be made for periodic inspections and trials of electric installation structure equipment which is critical for safety of the steam generating plant and the vessel/floating facility.

2.7 In accordance with Table **2.1**, Part VII "**M**achinery Installations", electric equipment of machinery and systems important for safety of the steam generating plant shall be capable of faultless operation under continuous heel up to **30°**, roll up to **45°** and trim up to **10°**.

3 MAIN ELECTRIC SYSTEM

3.1 The following shall be envisaged for the main electric system.

3.1.1 Failure of a single component within any main generator, drive motor of the latter and associated auxiliary machinery shall not cause shutdown of the reactor and loss of the vessel or self-propelling floating facility maneuverability. Simultaneously provision is to be made for fast recovery of required electric power needed for maintaining the vessel/floating facility in a normal operational state and under normal habitability conditions.

3.1.2 Failure of a single component within distribution devices of the main electric system shall not cause shutdown of the reactor and loss of the vessel/ floating facility maneuverability.

3.2 The main electric system shall include at least as follows:

Two main generators;

Two stand-by generators;

Two main switchboards.

3.3 The main electric system shall include at least two separate electric stations implemented so as not to effect each other operation in case of failure in any station under state class **SC1** or **SC2**.

3.4 Every electric station within the main electric system shall include main generator (generators), stand-by generator (generators) and main switchboard.

3.5 Electric energy supply to the machinery and systems of running steam generating plant shall be done from at least two electric stations.

3.6 Total power of operating main generators within every electric station of the main electric system shall be sufficient for full supply of electric energy to all the consumers required for maintaining the vessel /floating facility in a normal operational state and normal habitability conditions.

3.7 Loss of voltage at the buses of any main switchboard shall automatically actuate stand-by generators to take up load for a time necessary for safe operation of the steam generating plant.

3.8 The project shall envisage parallel operation of the stand-by generators with the main generators at least for a time necessary for transferring load.

3.9 Total power of the stand-by and main generators which remain operational shall be sufficient to supply electric energy to the consumers required for maintaining the vessel/floating facility in normal operational state and normal habitability conditions. In this case it is permitted that consumers which are not critical for the safety of the vessel/floating facility be disconnected.

3.10 Power of the stand-by generators actuated under abnormal conditions shall be sufficient to supply electric energy to the consumers providing safety of the vessel /floating facility, to return the latter into a normal operational state at minimal habitability conditions, as well as to perform scheduled startup and cooling of the steam generating plant.

3.11 It is permitted that the jumpers between the buses of main switchboards with appropriate switching devices be used for vessel's electric energy system.

3.12 Controls and instruments located in the central control station shall be arranged in consoles and panels so as to prevent failure of the remote control and monitoring more than one electric station if some of them fails.

3.13 The critical consumers of electric energy if they are two or more in number (provided they are mutually redundant and stand-by consumers are engaged automatically once running equipment fail) shall be separately connected to the different main switchboards both as related to power supply and control.

3.14 The steam generating plant consumers shall be powered from special switchboards of the steam generating plant fed from the main switchboards and emergency switchboard. The steam generating plant consumers may also be powered directly from the main switchboards.

3.15 Each of the main switchboards within the main electric system shall be located in a separate compartments.

The separate compartments are those isolated from each other with watertight fire structures.

3.16 The main generators of electric stations may be located in a common engine compartment, provided that the requirements stated in Para **3.15** are met

3.17 If the main generators are located in one common engine compartment, the stand-by generators shall be located in other separate compartments.

4 EMERGENCY ELECTRIC SYSTEM

4.1 The emergency electric system and generators independent of the steam generating plant engaged in emergency power supply, as well as the emergency distribution systems shall perform their safety functions taking into account the principle of single failure under the state classes SC1 to SC4 (see Para 2.2 also).

4.2 In addition to the requirements specified in RS Rules Section 9, Part XI Electric Equipment, power of the emergency electric system shall be sufficient to shut down the reactor, subsequently switch over into the cold subcritical state, and supply the consumers intended for performing the reactor safety functions.

4.3 The emergency electric system shall include not less than two emergency generators and two emergency electric energy distribution system independent on each other. It is permitted that separate emergency distribution systems with associated emergency generators be envisaged for the consumers of steam generating plant and those consumers be fed as per the requirements specified in the RS Rules Section 9, Part XI Electric Equipment. In this case consumers of the steam generating plant shall be supplied from at least two emergency generators with distribution systems and one emergency generator with an independent distribution system for power supply of the consumers as per the requirements specified in the RS Rules Section 9, Part XI Electric Equipment.

4.4 Each emergency generator shall be connected only to an associated emergency switchboard.

4.5 The emergency switchboards shall be powered from every main switchboard. If the emergency switchboards are used for power supply of the consumers in emergency modes only (from the emergency generators), they may not be connected to the main switchboards.

4.6 The consumers in charge of the safety systems shall be powered from the emergency switchboards via two feeders. If the system features full functional redundancy of machinery, power may be supplied via one feeder, provided power to the redundant machinery is supplied from the other emergency switchboard and requirements of Para **4.1** are met.

4.7 Each emergency generator shall start automatically by the signal of voltage loss on the associated main switchboard bus and by the actuation of signal of reactor protection system. In case of separate emergency power supply systems for the consumers of steam generating plant and for consumers listed in the RS Rules Section 9, Part XI "Electric Equipment" (see Para **4.3**), every emergency generator shall start automatically by the signal of voltage loss at the associated main switchboard buses and by actuation of the signal of reactor protection system.

Emergency power supply shall envisage independent actuation from the central control station, emergency control station and emergency generator location.

Breakdowns in any of those spaces (except for the emergency generator compartment) shall not prevent startup and control of the emergency generator from the emergency switchboard.

4.8 Power of the steam generating plant emergency cooling console, when supply from the main and emergency sources fails, shall be supplied from a transient electric energy source. Switch over from main supply to emergency and further to transient source of electric energy shall be performed automatically.

4.9 The emergency electric system shall take up load in a short time determined by the reactor safety conditions.

4.10 The emergency electric system shall be designed so as to exclude direct synchronization of electric energy sources in emergency.

4.11 Measuring instruments for every emergency generator installed in the emergency switchboard shall be redundant in the central control station.

5 TRANSIENT POWER SUPPLY SOURCES

5.1 The provision is to be made for at least two independent transient power supply sources.

5.2 The devices measuring parameters of the steam generating plant, radiation monitoring and other instruments and indicators critical for safety of the vessel /floating facility shall be powered from each transient power supply source for **30** minutes as agreed with the RS.

5.3 The transient power supply sources may not be required if justification is provided that the consumers specified in **Para** 5.**2**, have uninterrupted power supply where principle of single failure under any state, including SC**4** is envisaged

5.4 The transient power supply sources shall be distributed and installed so as to that not more than one transient power supply source fails under states SC1 to SC**4**.

5.5 Batteries to be used as transient sources for the steam generating plant only may be located below the bulkhead deck.

5.6 The project shall include a charger of sufficient power for charging the battery from completely discharged state to full charge during **6** hours as maximum.

5.7 The central control station shall be fitted with a common light indication (noncritical) of low battery and digital indication of battery discharge current controlled by an operator.

6 LIGHTING

6.1 Main Lighting

6.1.1 Each space within the controlled area which is important for safety of the steam generating plant shall be fitted with at least two main lights.

6.1.2 The main lights within the controlled area shall be powered from special switchboards earmarked for the controlled area only.

6.1.3 The main lighting switchboards listed in Para **6.1.2** shall be powered from different main switchboards via separate feeders.

6.1.4 The main lighting network of the controlled area spaces shall feature switchboard remote enable/ disable system with appropriate indication in the central control station.

6.1.5 Switches of the main lights of separate compartments or groups of compartments within the controlled area shall be installed outside these compartments.

6.1.6 All the lighting fittings within controlled area spaces shall have protection degree not less than IP55.

6.2 Emergency Lighting

6.2.1 Emergency lights shall be fitted in spaces as follows:

.1 Central control station;

.2 Reactor emergency cooling station;

.3 Spaces to be attended by personnel within the controlled area and compartments important for safety of the steam generating plant;

.4 Radiation monitoring station (if located separately);

.5 Special switchboards of the steam generating plant (if available);

.6 Storages places of new and wasted fuel assemblies.

7 ELECTRIC EQUIPMENT OF STEAM GENERATING PLANT AUTOMATION AND MONITORING SYSTEM AND RADIATION MONITORING SYSTEM

7.1 The automation and monitoring systems supporting operation of the safety systems and radiation monitoring system shall be powered from the main and emergency switchboards. Power supply shall be switched over to the emergency sources automatically.

The list of automation and monitoring devices powered from the transient electric energy sources shall be approved by the **RS**.

8 SUPPLY FROM EXTERNAL POWER SOURCE

8.1 Provision is to be made for a power supply board from an external electric energy source.

8.2 Provision is to be made for power supply from switchboard mentioned in Para **8.1**, to every main switchboard.

8.3 Construction of the external power supply board shall comply with the requirements specified in the **RS** Rules, Part **XI** Electric Equipment.

9 CONTROLLED AREA CABLING

9.1 Number of cables passing through the containment and shielding barrier shall be minimized.

9.2 Requirements for cable glands shall not be lower than the requirements to tightness of the spaces as regard to leakages and fire resistance of bulkheads. These requirements shall not be a hurdle for conducting inspections and tests.

9.3 The cables shall be led in through glands fitted from outside these compartments. Free spaces of these glands not filled with the cables from inside these compartments shall be properly packed with compound all over the protection thickness.

9.4 Application of electric cables with outer metal screen is not allowed.

9.5 Perforated panels and bridges are not allowed for cable installation.

9.6 Cables shall be laid in the shortest routes possible.

9.7 Cables shall be laid at distances from the planes of bulkheads, decks, framing and other hull structures so as to facilitate decontamination when necessary.

9.8 Cables running through the containment shall feature longitudinal tightness otherwise shall be led in with the aid of special bulkhead connectors or other fixtures ensuring tightness of the containment. Methods and standards for testing of cables for longitudinal tightness and emergency factors shall meet the requirements of normative documents approved by the **RS**.

Laying cables through the containment may be admitted in exceptional cases when a particular compartment cannot be bypassed, provided they are laid in steel tight pipes. All such cases are subject to special consideration by the **RS**.

9.9 Design of the cable glands passing through the containment shall allow inspecting its tightness in the course of installation and operation and guarantee tightness of the containment under conditions stated by the design-basis accidents. The air leakage rate shall not exceed 0.5 l/h through one bundled cable entry under absolute test pressure of 0.5 MPa after operational effect, emergency states and fire.

For getting the RS approval for application of cables and glands passing through the containment, their samples shall be tested for longitudinal tightness under circumstances of the design basis accident as per the procedures specified in the Guidelines for Technical Supervision over Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Fleet Support Vessels, and Manufacture of Materials and Products.

9.10 The cables for the redundant safety systems, switchboards and consumers shall be laid separately from main lines and protected properly.

9.11 A single wire system can not be applied for one-phase alternate current with the vessel hull used as a return wire.

9.12 The cables and electric equipment that shall be kept operational also after design accidents shall withstand environmental factors (pressure, temperature, humidity, etc.) associated with those accidents.

9.13 All the cables running from the transient power supply sources (if available) to the designated switchboards and going from the switchboards to consumers shall be distant from each other and from cable routes of the main and emergency distribution systems as far as possible.

9.14 The local cables connected to the equipment to be dismantled in reloading the core shall be marked.

9.15 The control equipment for the electric motors located within the controlled area shall be installed outside the latter. Start buttons are allowed in that case.

10 INTERNAL COMMUNICATION

10.1 Reliable communication between the central control station and spaces as mentioned below shall be provided even at total lack of power supply on the vessel /floating facility:

.1 Bridge;

.2 Reactor emergency cooling control station;

.3 Main engine;

.4 Main generators;

.5 stand-by generators;

.6 Emergency generators;

.7 Compartments to be attended within the controlled area important for safety of the steam generating plant;

.8 Storages of fuel assemblies.

11 ELECTRIC EQUIPMENT INSPECTIONS AND TESTS

11.1 The project shall envisage possibility for testing the stand-by and emergency generators. The tests shall include checking automatic, remote and local startup as well as checking start-up time and 100% load take-up. Speed regulators of the primary motors shall be tested in action as well.

11.2 A procedure for periodical testing a transient power supply source (if available) shall be approved by the **RS**.

PART XI. AUTOMATION

1 GENERAL

1.1 Application

This part comprises requirements to automation equipment of nuclear-powered vessels and floating facilities. The requirements in Part XV "Automation" of the RS Rules shall apply in full to automation equipment, unless otherwise specified herein.

1.2 Scope of technical supervision

In addition to automation systems specified in Part XV "Automation" of the RS Rules, the following shall be subject to technical supervision on nuclear-powered vessel and floating facility: control, protection, alarm and indication systems required for steam generating plant and safely systems operation.

1.3 Technical documentation

For requirements to technical documentation, see Part II "Classification".

2 GENERAL REQUIREMENTS

2.1 In addition to the requirements for components and appliances of automation systems specified in Part XV "Automation" of the RS Rules, control, monitoring and protection systems of nuclear power unit shall also comply with requirements of Sections 19 and 20 Part VIII "Nuclear-Powered Steam Generating Plants" as far as applicable.

2.2 For automation systems with redundancy according to single failure concept it is allowed to use common sensors in channels of protection, control, monitoring, alarm and indication if failure in channels of control, monitoring, alarm and indication does not affect operability of protection system.

2.3 Short power loss (up to 1 s) in the systems shall not affect operation of protection and control channels and shall not result in false actuation.

2.4 List of steam generating plant equipment subject to control and monitoring from central control station as well as level of automation and monitored parameters shall be justified in the design.

2.5 Automation systems required for operation of systems specified in Para 10.7 Part VIIi "Nuclear-Powered Steam Generating Plants" shall be reserved and shall comply with single failure criterion (see Section 7 Part III "Safely Standards").

2.6 In multi-channel automation systems the channels shall be galvanically independent.

2.7 Systems, specified in Para 2.5, shall be provided with sound and light alarm system to indicate completeness failure, if necessary.

2.**8** Control, protection and monitoring systems of steam generating plant shall allow remote actuation of safety systems.

2.**9** For control channels of safety systems, priority of automatic control over remote control shall be set by signals on protection actuation.

The scope of control channels for fulfillment of requirements specified in this Para will be specially examined by the Register for every specific case.

2.**10** Failures in systems specified in Para 2.5, shall be examined taking into account Para 7.4 Part III «Safety principles» in accordance with the following emergency situations:

.1 Failure of functional components within system (for example, fuse, card, module, etc.);

.2 Failure of structural components (for example, device, console, panel, etc.);

.**3** Failure of structural components group (for example, those located in common compartment).

3 ALARM, INDICATION AND PROTECTION SYSTEMS

3.1 For the list of alarm, indication and protection parameters of steam generating plant, see Table **3**.1.

3.2 Emergency parameters recorder providing record of pre-emergency and emergency values of steam generating plant parameters shall be installed on board nuclear-powered vessel/floating facility. Scope of parameters to be stored in emergency parameters recorder for each design shall be specially considered by the Register.

3.3 Emergency parameters recorder shall functionally meet requirements specified in Paras 5.21.2, 5.21.5, 5.21.7, 5.21.**8** and 5.21.1**0** Part V "Navigational Equipment" of Rules for the Equipment of Sea-Going Ships.

Table 3.1

No.	Parameter to be checked	Measurement point	Tolerance for alarm parameter	Protection, stop or change of mode	Parameter indication		Record on emergency parameters recorder
					Central control station	Emergency cooling control station	
1	2	3	4	5	6	7	8
1	Neutron power[1]	Feed-water valve	↑	×	●		+
2	Reactor power doubling period	Feed-water valve	↓	×	●		+
3	Position of regulating rods	Compensating group and emergency protection drive	↓↑	×	●	●	+
4	Pressure inside reactor	Primary circuit	↓↑	■ ×	●	●	+
5	Level in volume compensator	Volume compensator	↓↑	▼	●		+
6	Pressure in safety systems cylinders and tanks	On container	↓↑	▼■	◗		+
7	Water temperature at reactor output	Nuclear reactor	↑	▼	●	●	+
8	Water temperature at reactor input	Nuclear reactor	↑	▼	●		+
9	Coolant activity as per radiation monitoring standard sensors	Primary circuit	↑	▼	◗		+
10	Feed-water flow	Behind feed-water valve	↓	▼ ×	●		+
11	Feed-water pressure	Behind feed-water pump	↓	▼■	◗		+

1	2	3	4	5	6	7	8
12	Feed-water temperature	At steam generator input	↓↑	▼	◗		×
13	Feed-water salinity	Before feed-water pump	↑	▼	◗		×
14	Steam pressure	Behind steam generator	↓↑	▼	●		×
15	Steam temperature	In main steam line	↓	▼	●		×
16	Steam and steam-water mixture activity	Behind steam generator and main condenser	↑	▼	◗		×
17	Primary circulating pump revplutions	In primary circulating pump	↓	▼	◗	◗	×
18	Primary circulating pump load current	On nuclear steam generating plant switchbpard	↑	▼	◗		×
19	Pressure in containment	In containment	↑	▼■	◗		×
20	Water activity in III cpntour	Behind equipment	↑	▼	◗		×
21	State pf pumps and position of fittings in I to IV contours and safety systems	On pumps and fittings			○	○	×
22	Pressure difference on reactor plant and safety systems pumps	On the pump	↓	▼■	◗		×
23	Water levels in reactor plant tanks, safety systems, deaerating plant and ice boxes	On container	↓	▼■	◗		×

1	2	3	4	5	6	7	8
24	Pressure in pneumatic control system of steam generating plant systems	Within the system	↓	▼■	◗		×
25	Indication of power supply availability on steam generating plant panels and contactors position	On steam generating plant panels	↓		○		×
26	Vacuum in main condenser	On main condenser	↓	▼	◗		×

Notes: 1. Record is made after processing in nuclear reactor control and protection system
2. Parameters in Paras 1 to 26 are subject to cyclic recording at normal reactor operation at power.

Symbols::
● — remote indication (constant);
◗ — remote indication (on call);
↑ — alarm signal when parameter reaches upper limit value;
↓ — alarm signal when parameter reaches lower limit value;
○ — alarm signal;
■ — automatic start of stand-by pumps;
▼ — mode change, load decrease;
× — nuclear reactor stop.

PART XII. RADIATION SAFETY

1 SCOPE OF TECHNICAL SUPERVISION

1.1 Facilities for protection against radioactive radiation and emissions of radioactive materials, radiation monitoring systems, systems for collecting, storage, treatment and removal of radioactive waste from the vessel/floating facility, decontamination and sanitary treatment systems shall be subject to technical supervision by the Register.

1.2 Equipment and radiation safety systems shall be subject to technical supervision by the Register in the process of design, development of design drawings, manufacture, installation and testing during construction and testing on board the vessel/floating facility and in operation.

1.3 For documents on radiation safety equipment and systems subject to technical supervision by the Register, see Section 3, Part II "Classification".

1.4 For the scope of technical supervision on radiation safety equipment and systems during manufacture and testing as well as installation and testing on board the vessel/floating facility, see Guidelines on Technical Supervision During Construction of Nuclear-Powered Vessels and Floating Facilities, Nuclear Support Vessels and Manufacture of Materials and Products.

1.5 For the scope of technical supervision on radiation safety equipment and systems in service, see Table 2.2, Part II "Classification".

2 DEDINITIONS AND EXPLANATIONS

2.1 In addition to definitions given in Part I "General", the following definitions and explanations have been introduced in this Part:

Biological shielding comprises special structures and structural components designed to protect biological organisms and environment against radioactive emissions reduced to applicable standards.

Biological shielding may be made of steel alloys, concrete, lead, polyethylene and others.

Unacceptable risk is defined as a design minimum probability when the crew, passengers, population and environment are exposed to excess ionizing radiation and radioactive contaminations.

Limited part of population is defined as population being in area of possible radioactive emissions in case of severe accidents in steam generating plants or cores being stored on board vessel/floating facilities of SC4.

3 RADIOLOGICAL PROTECTION

3.1 To ensure radiation safety for all states of steam generating plant and vessel/floating facility, along with shielding barriers (see Para 6.2, Part III "Safely Standards") steam generating plant, storage facilities for radioactive waste and core fuel assemblies and other radioactive sources shall be provided with biological shielding.

To reduce radiation exposure, along with biological shielding it is required to use time of exposure, distance to radiation source as well as individual protection means.

3.2 Crew members other than personnel and any other people on board and near the vessel/floating facility shall be exposed to dose equivalents as specified by applicable Radiation Safely Standards for the Limited Part of Population.

3.3 The radiation protection facilities shall prevent exposure of people on board or near the vessel/floating facility due to penetrating radiation/radioactive contamination in amounts above the appropriate radiation dose limits as specified by applicable Radiation Safety Standards under SC1, SC2, SC3 and in case of reactor shutdown.

3.4 The basic design radiation dose limit for people on board the vessel / floating facility and limited part of population in case of SC4 shall be less than double maximum permissible dose as specified by applicable Radiation Safely Standards for the Personnel.

3.5 Biological shielding directed towards the bottom of the nuclear-powered vessel/floating facility shall prevent adverse effects on sea water when reactor plants is operating at a rated power. Radiation levels below the bottom of the vessel/floating facility shall allow for required docking operations with the reactor stopped.

3.6 The biological shielding design shall envisage repair works, reactor core handling, replacing steam generating plant equipment with shielding dismantled to the minimum level as well a survey of steam generating plant equipment.

3.7 The controlled and supervised areas shall be enclosed on board the vessel and floating facility according to actual and potential radiation hazards. To prevent contamination transfer into unrestricted area decontamination station shall be positioned between the controlled area and adjacent compartments. Decontamination station shall be provided with clothes changing facilities, dose control facilities for people and overalls and washing equipment. Access to supervised area spaces is to be allowed through special purpose sanitary space in case of radioactive contamination.

Warning sign shall be placed near the entrance to the controlled area and supervised area, if required.

3.8 All controlled area spaces where radioactive contamination may occur under normal operation of the vessel/floating facility shall be located inside the shielding barrier.

3.9 Systems shall be provided to supply fresh air to pressure suits and helmets. Air shall be supplied via two independent ventilation units including the stand-by one. The redundant ventilation unit shall be capable of automatic activation in case of failure in the main ventilation unit.

3.10 Decontamination facilities shall be provided for removing radioactive contaminations.

3.11 Material of structures as well as paint coatings of controlled area spaces and equipment where radioactive contaminations occur under SC1 and SC2 shall allow multiple decontamination procedures.

3.12 Controlled area spaces where decontamination solutions and washing water may stagnate shall be of simple configuration without recesses and projecting parts, if possible. Bulkhead stiffeners shall be fitted from the side of less likely contaminated spaces. Corners of hull structures shall be rounded, if possible.

3.13 Foundations, machinery and equipment attachments in the controlled area spaces where radioactive contaminations occur under SC1 and SC2 shall be designed to ensure access to all surfaces of foundations/their attachments for decontamination.

Foundation spaces inaccessible for decontamination shall be sealed.

3.14 Machinery and equipment not suitable for decontamination shall be easily replaceable. Arrangement for covering these machinery and equipment during operation or general decontamination of spaces shall be envisaged.

3.15 Controlled area spaces shall have emergency escape route to the open deck.

3.16 Controlled area spaces where radioactive contaminations may occur shall be tightly arranged within the collision protection in a single block, if possible to facilitate maintenance of machinery and equipment inside as well as to provide the shortest possible routes for people and transportation of equipment, materials and radioactive waste.

3.17 Controlled area spaces on decks shall have an exit to cargo lift/trunk. Spaces where more fittings are located and lift/trunk is likely to be used shall have direct exit/entrance from lift/trunk, if possible.

3.18 The controlled area spaces shall be free of equipment, machinery and device which require continuous supervision and maintenance.

3.19 The scuppers within the controlled area spaces shall be fitted with shut-off valves and allow the water to be completely drained from spaces. Decks within compartments shall be deflected/inclined towards the scuppers.

3.20 Equipment and machinery shall be secured within the controlled area spaces where radioactive contaminations occur under SC1 and SC2 to allow for decontamination when mounted on foundation and secured on the bulkhead.

3.21 Compartments intended for handling contaminated radioactive substances, fluids, machinery and materials according to the design shall be equipped with local exhaust ventilation in the vicinity of workplaces.

3.22 Through pipelines and cable routes not related to controlled area shall be laid in special-purpose sealed corridors/linings within this area. Penetrations of these routes and pipelines in bulkheads enclosing the controlled area shall be sealed.

3.23 Layout of equipment, fittings and valves, laying of pipelines cable routes shall be arranged within the controlled area with regard to their accessibility for maintenance, repair, inspection, decontamination and survey as well as application of protective coatings and covering.

3.24 Application of lattice structures is not allowed for ladders, flooring and catwalks.

4 RADIATION MONITORING

4.1 The special-purpose radiation monitoring system complying with these requirements and Part XI "Electric Equipment" and Part XV "Automation" of the RS Rules shall be provided to record levels of air and surface radiation, contamination and radioactivity of liquids on board the vessel/floating facility.

4.2 The radiation monitoring systems shall be designed for radiation process and radiation dose monitoring on board the vessel/floating facility for all states.

The part of radiation monitoring system intended for process monitoring purposes shall ensure the following:

.1 Monitoring leak tightness of fuel element claddings;

.2 Monitoring radioactivity of primary coolant;

.3 Monitoring radioactivity of the secondary and third fluids;

.4 Monitoring radioactivity of fluids in radioactive waste storage facilities

.5 Monitoring leakages flowing from primary to secondary and third circuits and to spaces;

.6 Measuring intensity of alpha-, beta- and neutron radiation, volumetric activity of gases and aerosols in corresponding spaces of controlled area;

.7 Radiometric analysis of radioactive samples;

.8 Indication on high ionizing radiation, contamination and fluid radioactivity;

.9 Indication on open access doors to controlled area spaces and open emergency escape doors;

.10 Output of signal for isolating the faulty steam generator.

See circular 846c **4.3** The ionizing radiation detecting units shall be redundant within the controlled area spaces, where necessary.

All sensors of radiation monitoring system devices shall be at least IP68 protected, the other equipment shall be IP23 protected.

4.4 Recording system shall record and store the following parameters:

.1 Radiation doses for people involved in operations within the controlled area and supervised area, if required;

.2 Ionizing radiation levels on board the vessel/floating facility;

.3 Radioactive contamination levels within attended areas on board the vessel/floating facility;

.4 Amounts and activity of radioactive waste being stored on board the vessel /floating facility;

.5 Activity of waste being discharged to shore facilities/special-purpose vessels;

.6 Volumetric radioactivity of primary coolant;

.7 Data on pre-emergency situation change in radiation situation in case of accident.

4.5 Data on radiation levels within controlled and supervised areas, air radioactivity within the containment as well as concentrations of radioactive gas and aerosols being released into environment shall be displayed on the console of radiation monitoring system. The console shall be equipped with indicators for monitoring any increase in radiation level.

4.6 The vessel/floating facility shall be equipped with sufficient portable means of radiation dose monitoring for operation under normal and emergency conditions. This equipment shall include dosimeters for alpha-, beta- and neutron radiation, air sample activity and contamination meters.

4.7 The vessel/floating facility shall be provided with sufficient amount of individual dosimeters for all people on board and for all conditions as specified by SCI to SC4.

4.8 In addition to devices specified in Para **4.6** and **4.7**, the vessel (floating facility) may be equipped with laboratory instruments for analyzing radioactive samples if automated radiation monitoring system is used for other purposes.

4.9 Radiation situation shall be monitored on board the vessel/floating facility under all states.

5 HANDLING RADIOACTIVE WASTE GENERAL

5.1 The design for steam generating plant and vessel/floating facility shall envisage safety of crew and passengers and environmentally friendly collection, storage and treatment of radioactive waste before this radioactive waste is further discharged from the vessel/floating facility.

5.2 The design of steam generating plant shall envisage the minimum formation of radioactive waste to the extent practicable.

5.3 The designs for steam generating plant and vessel/floating facility shall include appropriate arrangements for monitoring and handling of solid, liquid and gaseous radioactive waste being formed during normal operation to minimize its harmful effects on crew members, passengers, environment and vessel/floating facility.

5.4 When designing and operating the radioactive waste treatment and storage arrangements, the following shall be taken into account:

.1 Permissible radioactive levels;

.2 Requirement for biological shielding and usage of cooling system;

.3 Possible corrosive effects of some radioactive gases and liquids on materials of containers, pipelines, equipment and fittings;

.4 Requirement for radioactive leakage detection;

.5 Possible formation of radioactive gases and measures to be taken to reduce effects and prevent combustible gas explosions.

5.5 Capacity of radioactive waste storage facilities shall comply with operating conditions for the vessel/floating facility.

5.6 The design shall envisage preventive measures for radioactive waste discharge from storage facilities into environment and spaces of the vessel/ floating facility.

5.7 Storage and transportation facilities as well as pipelines for radioactive waste discharge from the vessel/floating facility shall be designed to prevent any discharge of radioactive substances into environment and other compartments of the vessel /floating facility.

5.8 Documents and Information on Safety shall contain criteria for design, manufacture, operation and testing intended for radioactive waste treatment and storage equipment. These criteria shall envisage subdivision of waste by their composition and volumetric radioactivity.

5.9 Radioactive materials with major impact on individual radiation doses shall be arranged within the shielding barrier.

5.10 The amount of radioactive gas being released into the atmosphere under SC1, SC2 and SC3 shall not result in radiation dose for passengers, crew or limited part of population above the limits as specified in the Radiation Safety Standards.

5.11 Solid and liquid radioactive waste shall be discharged ashore in accordance with radiation and sanitary requirements.

5.12 Radiological protection of people on board or in the vicinity of vessel/ floating facility during radioactive waste treatment and discharge shall comply with Para 3.2 and 3.3 hereof.

5.13 Containers and pipelines with fittings shall be made of corrosion-resistant materials and alloys intended for multiple decontamination. These materials shall be approved by the Register.

5.14 Pipelines of radioactive fluid transfer systems shall be made of seamless electro polished pipes. Pipelines shall be connected by welding as per regulatory documents approved by the Register. Flange/union connections are not allowed unless special approval is given by the Register.

5.15 Pumps, pipelines and fittings shall have biological shielding, if necessary.

5.16 The foundations and fasteners of radiation safely system equipment shall prevent its displacement in case of variation in vessel/floating facility position up to and including capsizing.

5.17 The interior surface of radiation safety system containers exposed to radioactive fluid and not to be painted shall have roughness not more than R_a = 6.3 μm.

5.18 The distance between piping and systems shall be as such to ensure their proper maintenance and survey.

5.19 Requirements to quality of external surfaces of structures and equipment located in the controlled and supervised areas shall be developed by the Designer and approved by the Register.

5.20 Strength of equipment and radiation safety system shall be estimated according to standards specified for safety class 3 equipment.

6 HANDLING SOLID RADIOACTIVE WASTE

6.1 The spent ion-exchange resins and filters as well as different parts (dirty tools, overalls, laboratory kits, etc) shall be considered as typical solid radioactive waste.

6.2 Solid radioactive waste shall be stored and transported in special-purpose containers. Storage of solid radioactive waste shall envisage possible concentration/formation of gases and liquids.

7 HANDLING LIQUID RADIOACTIVE WASTE

7.1 Liquid radioactive waste forming in case of SC1 and SC2 shall be collected on board into enclosed containers (montejus, tanks) located in the controlled area spaces.

7.2 Liquid radioactive waste treatment and storage facilities shall transfer this waste ashore or on board the special-purpose vessel through two separate pipelines. One pipeline shall be used for medium-radioactivity liquid radioactive waste, the other one — for low-radioactivity liquid radioactive waste.

7.3 The following shall be taken into account for designing liquid radioactive waste storage facilities.

7.3.1 Waste shall be subdivided by their activity and with regard to physical and chemical properties, if required. Liquid radioactive waste shall be subdivided into low-, medium- and high-radioactivity waste as per applicable Sanitary Radiation Safety Rules.

7.3.2 Containers/tanks shall be protected against spontaneous emptying in case of damage to pipelines due to water ejection by siphon effect or by gravity.

7.3.3 Liquid radioactive waste discharge pipelines shall be remotely isolated from central control station and from discharge station.

7.3.4 Liquid radioactive waste collection and storage containers shall be designed as free-standing, externally framed and inclined towards the drain hole. The roughness of containers interior surface exposed to radioactive fluid and not subject to painting, shall not exceed R_a = **6.3** μm.

Containers shall be designed to meet Para 7.1.1.5 of the Rules for the Classification and Construction of Nuclear Support Vessels. Means for remote measurements of liquid radioactive waste levels shall be provided.

Medium- and low-radioactivity liquid radioactive waste shall be stored in separate spaces.

Containers for storing medium-radioactivity liquid radioactive waste shall be made of corrosion-resistant materials suitable for multiple decontamination and washing. Such containers shall have the appropriate biological shielding. The vessel shall be provided with at least two containers.

Containers for storing low-radioactivity liquid radioactive waste may be made of ordinary structural materials with anti-corrosion coatings applied. Vessel's structures and spaces may be used as a biological shielding.

7.3.5 Overflow of high-radioactivity liquid radioactive waste to containers for low-radioactivity liquid radioactive waste is not allowed.

7.3.6 Liquid radioactive waste containers shall allow for regular removal of contamination.

7.3.7 When discharging liquid radioactive waste overboard, contamination of the vessel and environment shall be excluded.

Arrangements for automatic discharge stop shall be provided for urgent piping shut-off or in case of spontaneous disconnection of removable pipelines. These arrangements shall be capable of automatic actuation upon the low pressure signal.

Prior to starting operations, removable pipelines shall be subject to leak tests.

Trays with water draining into liquid radioactive waste collection system shall be provided at connection points of removable pipelines. Branches for joining removable pipelines shall be arranged in a special-purpose station/ enclosure near sides. Requirements to liquid radioactive waste discharge station shall comply with Para 7.1.2.5 of the Rules for the Classification and Construction of Nuclear Support Vessels.

Removable pipelines shall be capable of being decontaminated, washed and completely drained without being disconnected from the liquid radioactive waste discharge pipeline.

7.3.8 Containers for storing liquid radioactive waste shall be equipped with air pipes made of corrosion resistant materials. Air pipes from liquid radioactive waste storage containers/tanks under hydrostatic pressure shall be led from the top of container/tanks to spaces where they are located. Air pipes from low-radioactivity liquid radioactive waste storage containers/tanks may be led to the ventilation mast through special-purpose ventilation system. Water injection from liquid radioactive waste containers to vent ducts shall be excluded. Air pipes shall be connected to each other and to containers/tanks by welding.

7.3.9 Liquid radioactive waste storage containers under hydrostatic pressure shall be made and tested as per Part II "Hull" of the RS Rules.

In addition to air pipes, liquid radioactive waste storage containers under hydrostatic pressure only shall be fitted with overflow system for collection and discharge of liquid radioactive waste when the main containers/tanks are overfilled.

7.3.10 Containers continuously or regularly operating under internal pressure shall be made and tested as per Part X "Boilers, Heat Exchangers and Pressure Vessels" of the RS Rules.

7.3.11 Fittings of liquid radioactive waste storage and discharge systems shall be of bellows type with branches to be welded and fitted with local position indicators and alarm with extreme position indication.

7.3.12 Electric pumps for liquid radioactive waste transfer shall be corrosion resistant and leak tight. At least two pumps shall be provided on board the vessel. Liquid radioactive waste discharge system shall be provided with arrangements for preventing pressure increase above design values.

7.3.13 Liquid radioactive waste pipelines and fittings shall have biological shielding (where control is provided from its location).

7.4 Spaces which are likely to be contaminated with liquid radioactive substances shall be fitted with bilge wells and bilge alarms.

8 HANDLING GASEOUS RADIOACTIVE WASTE

8.1 All escape routes for gaseous radioactive waste shall be monitored.

8.2 Radioactive gases and aerosols shall be discharged into environment through pipelines and vent ducts meeting tightness requirements and fitted with radioactivity filtering and monitoring equipment.

8.3 Gaseous radioactive waste may be compressed and stored provided that pressure vessels and appropriate pipelines meet the requirements of these Rules.

Radioactivity risks shall be analyzed in the design in case of depressurization of the cylinder containing gaseous radioactive waste.

8.4 The total volumes and radioactivity levels of aerosols and gases being discharged into the atmosphere shall be continuously and progressively monitored. These parameters shall not exceed the standards as specified in the Sanitary Radiation Safety Rules.

8.5 Gaseous radioactive waste discharge lines shall be fitted with automatic, remote and local shutdown means to prevent uncontrolled discharge.

9 STORAGE FACILITIES FOR CORE FUEL ASSEMBLIES

9.1 Storage facilities for new fuel assemblies and spent core fuel assemblies shall be arranged on board the vessel/floating facility as per Section 6 of the Rules for the Classification and Construction of Nuclear Support Vessels.

PART XIII. PHYSICAL SECURITY

1 SCOPE OF TECHNICAL SUPERVISION

1.1 The system of physical security engineering facilities of nuclear-powered vessels and floating facilities shall be subject to technical supervision by the Register.

1.2 Facilities shall be subject to technical supervision by the Register at the stages of design development, manufacturing, onboard installation, commissioning, operation and alteration (refitting) of physical security systems.

1.3 This Part determines the scope of technical supervision for equipment of physical security systems at the stages of design development, manufacturing, onboard installation, testing and operation.

2 DEDINITIONS AND EXPLANATIONS

2.1 In addition to definitions given in Part I "General', the following definitions have been introduced in this Part:

Physical security personnel is defined as personnel responsible for physical security on board the nuclear-powered vessel as part of their duty regulations.

Physical security control station is defined as a designated space/location equipped with engineering facilities. This space is used for control, in full scope or in part, of physical security engineering facilities in normal and emergency situations by designated physical security personnel.

Physical security facility is defined as a type of equipment to be used by designated personnel for detection of unauthorized actions, receipt of information on attempts and occurrence of such actions, notifications on attempts and occurrence of these actions, detection and suspension of unauthorized actions.

Readout device is a device to be used for reading data from identifier.

Physical barrier is a physical obstacle to prevent intrusion of unauthorized persons to controlled areas, nuclear materials/vulnerable points of nuclear plant.

Protected area comprises open areas of decks of the vessel/floating facility with restricted and controlled access.

Internal area is defined as an area at interior locations of the vessel/floating facility surrounded by physical barriers with restricted and controlled access.

Critical area is defined as an area at interior locations surrounded by physical barriers with continuously restricted and controlled access.

Secured area is defined as a protected, internal or critical area.

Identifier is defined as an assigned or inherent attribute to be used for proving eligibility for access to the secured area.

Identification is a process for identifying the subject/object by its inherent identification attribute.

Unauthorized person is defined as a person who has performed or is attempting to perform the unauthorized action as well as the assisting person.

Unauthorized action is defined as an action or attempt for sabotage/act of terror , theft of nuclear materials, nuclear plants, unauthorized access, carrying prohibited objects, breaking down or causing malfunction of physical security engineering facilities.

3 GENERAL REQUIREMENTS

3.1 Nuclear-powered vessels and floating facilities shall not be operated without ensuring physical security of nuclear materials, nuclear plants, storage facilities for nuclear materials and radioactive waste.

3.2 No measures taken to ensure physical security shall impede immediate and safe entrance/escape of personnel from any space in the event of accidents (fire, flooding, etc).

3.3 The engineering facilities system comprises engineering and technical facilities of physical security.

3.3.1 The engineering facilities include physical barriers and engineering equipment of secured areas. Physical barriers are structural components of hull and superstructures (decks, bulkheads, doors, hatch covers) and purpose-built structures (obstructions, grating, reinforced doors).

See circular 846c **3.3.2** Technical facilities of physical security include components and devices as a part of the following main systems:

.1 Intrusion protection system;

.2 Security alert system;

.3 Access monitoring and control system;

.4 Optoelectronic surveillance and situation assessment system;

.5 Operational communication and address system (including wire and radio communication means);

.6 Data protection system;

.7 Power supply and lighting system.

3.3.3 Engineering and technical facilities of physical security shall be controlled from physical security system control stations. The operator's consoles shall display incoming signals and data in at least two modes of three available (visual, light and audible). The access to control station spaces shall be provided by means of test and access control facilities.

3.3.4 Prior to stating manufacturing process, documents for engineering facilities system of physical security shall be submitted to the Register for examination and approval.

3.4 Electric equipment of engineering facilities system of physical security shall comply with Part XI "Electric Equipment" of the RS Rules.

3.5 The secured and limited access areas shall be enclosed and documented on board nuclear-powered vessels and floating facilities. Spaces shall be subdivided into appropriate categories. The categories of spaces shall be determined at the design development stage.

3.6 When enclosing the secured areas, the critical area shall be located within the interior area and the interior area shall be located within the protected area.

3.7 All entrances/exits to spaces of appropriate categories shall be equipped with detection facilities, access monitoring and control means and surveillance and situation assessment arrangements, if required.

3.8 Failure or breakdown of any component included into physical security technical facilities system shall not result in malfunction of physical security system.

3.9 Single technical facilities of physical security may ensure compliance with requirements imposed to one or several systems (integrated systems and devices).

3.10 Cabling of physical security system shall be properly protected on open decks of the vessel.

3.11 Computers and computer systems included into physical security engineelring facilities system shall comply in full with requirements stipulated for the similar equipment mentioned in Section 7, Part XV "Automation" of the Rs Rules.

3.12 The availability of spare parts and fixtures shall be defined by Manufacturer of technical facilities and shall be agreed upon with the Owner.

4 PHYSICAL BARRIERS AND ENGINEERING EQUIPMENT

4.1 Physical barriers shall comply in full with these requirements and all requirements of Section 7, Part III "Equipment, Arrangements and Outfit" and Section 2, Part VI "Fire Protection" of the RS Rules.

4.2 The following provisions shall be envisaged:

.1 Stalling/slowing down of unauthorized people;

.2 Provision for opening doors from inside the secured space;

.3 Emergency unlocking of doors/locking devices from control station in the event of accidents.

4.3 Engineering equipment of secured areas shall prevent attempts of unauthorized access and carrying of prohibited objects.

4.4 Check points/stations shall be fitted with arrangements for protection of personnel responsible for control and check operations against small-arms weapons.

5 INTRUSION PROTECTION SYSTEM

5.1 Intrusion protection system shall detect attempts of unauthorized actions and unauthorized actions actually occurred, provide personnel with data, transmit appropriate signals to other physical security systems.

5.2 To prevent uncontrolled actions on intrusion protection system, the following shall be ensured:

.1 Remote control of system components state from physical security system control stations;

.2 Backing up all events occurred in the physical security system.

6 SECURITY ALERT SYSTEM

6.1 Security alert system shall notify the physical security personnel on unauthorized actions and indicate the call point.

6.2 Unauthorized shutdown of security alert system devices shall be excluded.

6.3 Data being transmitted to operator from security alert system equipment shall be of higher priority as compared to other signals.

6.4 Security alert system shall transmit alert signals to physical security system control station upon pressing on alert buttons.

See circular 846c

7 ACCESS MONITORING AND CONTROL SYSTEM

7.1 Access monitoring and control system shall provide automatic and remote control for lock/locking devices actuators as per established algorithm and monitoring their state.

7.2 Lock/locking devices actuators shall actuate only upon reading of identification attribute which permits access to the secured area at a given time. In case of power loss in actuators, locks/locking devices shall be secured in the open position.

7.3 The following provisions shall be made: protection of signals being generated within access monitoring and control system, protection of facilities against unauthorized access which entails attempts to change system operation mode or steal/erase data, monitoring of integrity of facilities.

7.4 The alert signal shall be generated in case of breaking/attempting to break components which when impacted may result in unauthorized passage/ malfunction of system operation.

7.5 Facilities and devices of access monitoring and control system central control station shall ensure the following:

.1 Locking/unlocking of doors and automatically recording of events to the event log;

.2 Monitoring authorized access of crew members/other people to secured areas and preventing attempts for unauthorized access within specified time;

.3 Submitting data on attempts for unauthorized access as well as forced actions on gate structural components to the operator of physical security system;

.4 Automatic saving of data (with recording of data and time) on current events, emergency situations, attempts for unauthorized access, states of access monitoring and control devices and elements.

7.6 The following provisions shall be made for people attending lobbies of critical areas:

.1 Possibility for quick escape in case of accident;

.2 Monitoring and surveillance over people within the lobby;

.3 Maintaining internal microclimatic conditions at a required level rated for possible long-term staying of people.

See circular 846c

8 OPTOELECTRONIC SURVEILLANCE SYSTEM

8.1 Optoelectronic surveillance and situation assessment system shall ensure surveillance in secured areas and transmission of visual data to physical security system control point/points and recording of received data.

8.2 Technical facilities shall be protected against unauthorized access.

8.3 Technical facilities shall be tested for faults and control station operator shall be properly notified on such a matter.

8.4 In addition to above mentioned requirements, surveillance and situation assessment facilities shall comply with Para 7.14, Part XI "Electric Equipment" of the RS Rules.

9 SECURITY LIGHTING SYSTEM

9.1 Security lighting system facilities shall comply with requirements of this Section and Section 6, Part XI "Electric Equipment" of the RS Rules.

9.2 Security lighting shall be capable of automatic switching on upon actuation of intrusion protection system.

9.3 All switchgears of security lighting system shall be protected against unauthorized actions.

9.4 Security lighting system shall be switched over to stand-by power supply without decrease in lighting intensity of supervised area.

10 OPERATIONAL COMMUNICATION SYSTEM

10.1 Operational communication system shall be used for voice data exchange between physical security system personnel by means of wire and radio communication.

See circular 846c **10.2** Operational communication system shall comply with Para 7.2, Part XI "Electric Equipment" of the RS Rules.

10.3 Operational communication shall be provided by the system operating independently on other vessel's communication systems and designed only for physical security purposes.

10.4 The operating communication system shall be capable of recording voice conversations both manually and automatically and indicating their time and duration.

10.5 The operating communication system equipment shall be capable of isolating the unauthorized connection.

11 POWER SUPPLY SYSTEM FOR PHYSICAL SECURITY FACILITIES

11.1 Power supply system for physical security engineering facilities shall comply with requirements of this Section and Section 3, Part XV "Automation" of the RS Rules.

11.2 The space where physical security system switchboard is located shall be equipped with access monitoring and control means and intrusion protection system arrangements.

11.3 Physical security facilities shall be switched over to stand-by/ emergency power supply and vice versa without generating alert signals.

11.4 Power supply units and cable networks shall be protected against unauthorized actions regarding their breakdown.

INFORMATION ON SAFETY OF NUCLEAR-POWERED VESSEL (FLOATING FACILITY) (Recommended Content)

1 GENERAL PRINCIPLES

1.1 The Information on Safety is based on initially submitted documents followed by recommendations, supplements and revisions.

1.2 The Information on Safety shall contain systematic analysis of technical issues related to safely of the nuclear-powered vessel (floating facility) with respect to design, construction, operation and decommissioning proving that the vessel (floating facility) does not pose unacceptable risk for the crew, people and environment. The Information shall include sufficient data to allow the **RS** and authorized bodies of a host country to evaluate safety of the vessel/floating facility.

1.3 The Information shall be submitted in a brief form and issues shall be considered based on their importance for safety of the nuclear-powered vessel / floating facility.

1.4 If the provisions of Rule 5, Chapter I, International Convention for the Safely of Life at Sea devoted to appropriate alterations are adopted, the Information on Safely shall include description of appropriate alterations with calculations proving their reliability.

2 PRACTICAL INSTRUCTIONS

2.1 It is required that a provision be made for Information on Safely document to add additional data or include revised sections. All pages of the document shall have clear numbers in sequence and respective dates. Revised pages and supplements shall distinctly differ from the initially submitted materials (revision number and revision date shall be specified).

2.2 It is required that drawings, graphs, diagrams, tables and charts be used whenever they are needed for better understanding of the subject.

2.3 All information to be forwarded shall be understandable. To keep the drawings clear and legible the scale shall not be reduced. The units of

International System and units actually applicable to instruments shall be used.

2.4 The Information on Safely may contain references to other documents, provided that they can be easily obtained by appropriate authorities.

3 GENERAL INFORMATION

3.1 The introduction shall contain the general overview of project including design development, construction and operation of the vessel (floating facility) and its nuclear power unit, as well as conclusions on the vessel (floating facility) safety.

Brief description is required for the following:

.1 Design of vessel (floating facility) and its characteristics;
.2 Steam generating plant and design parameters;
.3 Containment and shielding barrier;
.4 Nuclear power unit;
.5 Auxiliary machinery and systems;
.6 Electric energy systems;
.7 stand-by propulsion plant (if provided);
.8 Collision protection.

3.2 Evaluation is required for nuclear and radiation safely with specification of measures for preventing and restricting consequences of accidents and conclusions on safely for crew, people and environment.

4 DESIGN ENVIRONMENTAL FACTORS

4.1 The section shall contain information on environmental factors adopted as the basis for design development highlighting those points which are important for nuclear safety as well as for general safety of the vessel/floating facility. The section shall justify the choice of design environmental factors, including sea state, basic design storm, fatigue service life, and environmental risk factors in areas of operation.

5 STANDARDS AND RULES

5.1 This section shall provide technical, radiation and administrative safety rules used as the basis for design development, construction and operation the vessel (floating facility) and nuclear power unit.

.1 Design rules:

Standards;

RS rules;

Design standards;

State requirements and rules.

.2 Practical experience in construction and operation;

.3 Operating instructions for operational period of the vessel/floating facility and for decommissioning period for the vessel/floating facility;

.4 Rules for operating the vessel/floating facility in emergency situations:

Anticipated operating faults, emergency conditions;

States permitting to operate the vessel /floating facility beyond specified conditions;

.5 Quality assurance programs applicable for the following: at design development stage of the vessel and main equipment; construction and trials; operation and maintenance.

6 TECHNICAL SPECIFICATION OF DESIGN SOLUTIONS

6.1 This section shall contain technical specification of design solutions related to different systems, structures and components in view of their importance for safely of the vessel (floating facility) and nuclear power unit.

6.1.1 Initial design data included in the section shall define the required characteristics and parameters of the systems, as well as the external conditions required to achieve these specified characteristics.

6.1.2 The specification shall contain information on systems to be examined and structures as follows:

.1 Functions;

.2 Normal and extreme operation parameters;

.3 Choice and characteristics of materials;

.4 Structural layout;

.5 In-service inspections and tests;

.6 Maintenance;

.7 Results of strength analysis;

.8 Results of thermal and hydraulic calculations.

6.2 The specifications and data required in Para **6.1** shall be applied to the following systems:

6.2.1 Vessel/floating facility and vessel's systems:

.1 Arrangement;

.2 Characteristics;

.3 Stability and division into compartments;

.4 Damage control;

.5 Hull structure and strength;

.6 Collision protection;

.7 Navigation;

.8 Communication;

.9 Life saving appliances;

.10 Vessel's machinery: Electric energy.

Main propulsion plant (for instance, main condenser, turbine, steam pipeline, feed water system);

Steering gear;

Fire detection and prevention;

HVAC systems;

Bilge and ballast systems, cargo lifting gears, anchor-and-mooring gear.

.11 Other systems.

6.2.2 Steam Generating Plant:

.1 Primary circuit:

Reactor;

Primary circuit pumps;

Safety valves;

Primary circuit pipelines;

Steam generators;

Pressure compensating system;

Fittings.

.2 Auxiliary systems:

Radioactive wastes, make-up system, third circuit system, sampling system; containment air ventilation and filtration, primary circuit degassing and draining, and others;

.3 Reactor core;

.4 Instruments and controls;

.5 Safety systems:

Reactor control and protection system;

Core emergency cooling system;

Residual heat removal system;

Soluble poison injection system;

Containment cut-off system;
Leak detection system.;
6.2.3 Central control station and emergency cooling control station:
.1 Inspection scope;
.2 Instruments;
.3 Location and description;
.4 Fire protection;
.5 Habitability and access.

7 NORMAL MODES OF NUCLEAR POWER UNIT

7.1 This section shall contain information on functional behavior of the unit in normal operation modes.
7.2 The information on normal operation shall include description of as follows:
.1 Initial state of the nuclear power unit prior to starting up;
.2 Starting up procedures;
.3 Operating at permanent power level;
.4 Changing power in the course of operation;
.5 Shifting to hot stand-by mode and further to cold state;
.6 Quick return to operation at power after unexpected quick shutdown.

8 RADIATION SAFETY

8.1 This section shall contain main data on radiation safety as follows:
.1 Main criteria of radiation protection;
.2 Radiation exposure limits;
.3 Radioactive waste discharge;
.4 Radiation levels for every zone on vessel (floating facility) and procedures of access to zones at different state classes;
.5 Handling radioactive substances.
8.2 Biological shielding description:
.1 Specification of a source to be protected;
.2 Arrangement and application;
.3 Sizes and materials.
8.3 Data on radiation monitoring shall contain the following:
.1 Arrangement;
.2 Type, sensitivity and measurement range of sensors to be used;

.3 Methods of information display and signaling;

.4 Procedures of radiation and chemical monitoring of coolant, feed and cooling water;

.5 Instructions on reliability and durability of radiation monitoring system;

.6 Type and quantity of individual dosimeters.

8.4 The information on radioactive materials discharge into environment shall contain data on instruments and methods for measuring leakages from the unit and data on automatic or manual actuation of discharge-restricting systems.

8.5 The following shall be described: spaces and appliances to be used for treatment of contaminated objects and people as well as decontamination stations and laboratories (indicating their layout).

9 ACCIDENT AND FAILURE ANALYSIS

9.1 This section shall contain detailed information on possible consequences of events affecting the unit or vessel /floating facility as a result of:

.1 Failure or malfunction of systems, components or structures;

.2 Incorrect actions of personnel while operating the unit;

.3 Accidents on board the vessel /floating facility (i.e. fire, collision, grounding and stranding, flooding etc.).

9.2 The anticipated development of events after failures or accidents shall be described:

.1 Root cause of event;

.2 Sequence of events after the prime event .3 Final consequences.

9.3 The analysis shall include the following information:

.1 Initial state;

.2 Assumptions used as a basis for calculations;

.3 Coolant radioactivity values;

.4 Accepted defects of fuel claddings;

.5 Value of leakage from the containment and efficiency of adsorption and filtration;

.6 Accepted automatic actions or necessary operator's actions;

.7 Time period after the event for measures to be taken.

9.4 Accident analysis shall be conducted based on a single failure criterion.

9.5 Faults of the nuclear power unit are as follows:

.1 Unintended radiation variation, including, for instance:

Unintended displacement of the control rod or group of rods with the highest physical weight;

Cold water injection;

Failure of the feed valve, i.e. supply of feed water with maximum flow rate in operation at low power;

.2 Malfunctions of the primary circuit system:

Failure of the make-up system;

Partial or complete breakdown of forced circulation;

Coolant pressure drop (drop of level in volume compensator);

Rupture of the primary circuit, i.e. accident with coolant loss;

Overheating of coolant, rupture of steam generator pipe;

.3 Malfunctions of the secondary coolant circuit system:

Rupture of the main steam generator or feed water main pipeline;

Pressure rise;

Closure of the main shut-off steam valve before the turbine;

Termination of steam withdrawal from the steam generating plant;

Termination of cooling water supply to the main condenser;

Termination of feed water supply;

.4 Other accidents:

Malfunctions of the electric energy system;

Failure of the central control station;

Unintended starting up the emergency cooling system;

Faults of the radioactive waste treatment and storage systems and degassing systems.

9.6 Accidents on board the vessel/floating facility.

The following states shall be considered for the conditions of vessel/floating facility being at sea and in harbor:

.1 Collision;

.2 Grounding and stranding;

.3 Capsizing;

.4 Flooding in shallow waters;

.5 Flooding in deep waters;

.6 Fire within shielding barrier;

.7 Fire in any other location on board the vessel/floating facility;

.8 External hazards in the vicinity of the vessel/floating facility, i.e. fire, explosion, toxic gases, etc.;

.9 Loss of maneuverability;

.10 Crash of helicopter and etc.

10 ACCEPTABLE CONDITIONS FOR VESSEL/FLOATING FACILITY OPERATION

10.1 The following details shall be elaborated in this section: details of operating conditions and requirements of technical, administrative and systematic nature. This shall be applicable to at least the following issues:

.1 Limit conditions for operating the vessel/floating facility (see Para 5.1.4 herein);

.2 Surveys and inspections of technical condition (intervals and scope of records and tests);

.3 Control (references to the Operating Manual and organizational guidelines may be given):

Organization and lines of subordination and responsibility

Procedures for making amendments and obtaining and approvals for operating instructions and directives

Manning (qualification and number of people)

Procedures and instructions for control under normal operating conditions, at anticipated operating faults, emergencies and accidents

.4 Maintenance.

11 DECOMISSIONING

11.1 Procedure for decommissioning of the vessel/floating facility without unacceptable radiation effects on population.

12 CONTENTS

It is recommended that the following typical Contents regarding Information on Safety be provided:

12.1 General

12.1.1 Purpose and Type of Vessel/Floating Facility. Expected Pattern of Application

12.1.2 Chronology of Vessel/Floating Facility Production: Shipyard, Steam Generating Plant Manufacturer

12.1.3 Design, Construction and Operation Supervising Authorities

12.1.4 Design Criteria and Design Standards for Vessel/Floating Facility and Nuclear Power Unit

12.1.5 Quality Assurance Programs

12.2 Vessel/Floating Facility and General Safety
12.2.1 Vessel/Floating Facility General Characteristics and Description
12.2.1.1 General Characteristics
12.2.1.2 General Description
12.2.1.3 Hull Structure and Strength
12.2.1.4 Arrangement of Nuclear Power Unit, Equipment and Control Stations
12.2.1.5 Maneuvering Capabilities
12.2.2 Collision Protection of Reactor Compartment
12.2.3 Stability and Buoyancy under Normal and Emergency Conditions
12.2.4 Navigation and Communication Equipment
12.2.5 Life Saving Appliances
12.2.6 Fire Protection
12.2.7 Vessel's Arrangements:
12.2.8 vessel Systems
12.3 Steam Generating Plant
12.3.1 General Description and Characteristics
12.3.2 Primary Circuit
12.3.2.1 General Characteristics
12.3.2.2 Equipment Redundancy
12.3.2.3 Equipment Arrangement
12.3.2.4 Equipment
12.3.2.4.1 Reactor (Design, Materials, Strength, Core)
12.3.2.4.2 Steam generators
12.3.2.4.3 Circulating Pumps
12.3.2.4.4 Actuating Mechanisms of Safety Control Systems
12.3.2.4.5 Auxiliary Equipment
12.3.2.4.6 Pressure Compensator
12.3.2.4.7 Safety, Pressure-Relief and Shut-off Valves
12.3.3 Auxiliary Systems and Equipment
12.3.3.1 Primary Circuit Coolant Purification System
12.3.3.2 Reactor Make-up and Emergency Cooling Systems
12.3.3.3 High-Pressure Gas System
12.3.3.4 Steam Generator Piping Leak Detection System
12.3.3.5 Intermediate Cooling System
12.3.3.6 Sampling System
12.3.3.7 Degassing and Draining System
12.3.4 Emergency Systems
12.3.4.1 Emergency Cooling System
12.3.4.2 Core emergency cooling system

12.3.4.3 Soluble Poison Injection
12.3.4.4 Steam Generator Overpressure Prevention System
12.4 Safety Control Systems
12.4.1 Construction Principles
12.4.2 Description
12.4.3 Parameters, Instruments, Equipment
12.4.4 Interconnection with Steam Turbine and Electric Energy Units
12.4.5 Control Stations
12.5 Containment
12.5.1 Structure
12.5.2 Strength
12.5.3 Tightness
12.5.4 Pressure Reducing System
12.5.5 Emergency Flooding System
12.6 Shielding barrier
12.6.1 Structure
12.6.2 Strength
12.6.3 Tightness
12.7 Radiation safety
12.7.1 Biological Shielding Design and Materials
12.7.2 Radioactivity in Cooling Systems
12.7.3 Scheme of Vessel /Floating Facility Division into Radiation Zones
12.7.4 Ionization Radiation Levels
12.7.5 Special Measures for Health Safety and Protection Means
12.7.6 Radiation Monitoring
12.7.7 Radioactive Wastes
12.7.7.1 Gaseous Wastes
12.7.7.2 Liquid Wastes
12.7.7.3 Solid Wastes
12.7.8 Ventilation and Conditioning Systems
12.8 Steam-Turbine Unit
12.8.1 Secondary Coolant Circuit Description and General Characteristics
12.8.2 Main Steam System
12.8.3 Main Condenser Cooling System
12.8.4 Feed Water and Condensate Make-up System
12.8.5 Auxiliary Steam Systems
12.8.6 Emergency Propulsion Energy Sources
12.9 Electric System
12.9.1 Electric Energy Sources
12.9.2 Electric Power Plant Load Analysis

12.9.3 Electric Energy Distribution

12.9.4 Steam Generating Plant Emergency Electric Supply Diagram

12.10 Nuclear Power Unit Operation Modes

12.10.1 Initial State. Start-up Preparation

12.10.2 Start-up

12.10.3 Power Operation

12.10.4 Shutdown

12.10.5 Operation from Emergency Power Source

12.11 Operation of Vessel or Floating Facility /reference to the Vessel Operating Manual may be included

12.11.1 Organization of Operation

12.11.2 Crew Number and Qualification

12.11.3 Watch Organization

12.11.4 Personnel Training and Practice Alerts

12.11.5 Operating Documentation

12.11.6 Surveys

12.11.7 Harbor Entering and Berthing

12.11.7.1 Description of Local Conditions

12.11.7.2 Measures to be Taken on Board Vessel /Floating Facility Prior to Entering Harbor

12.11.7.3 Berthing Conditions

12.11.7.4 Organization of Emergency Alert Actions

12.11.7.5 Vessel/Floating Facility Security Measures

12.11.8 Vessel/Floating Facility Rescue operations

12.12 Analysis of Accidents

12.12.1 Accidents Related to Steam Generating Plant Malfunctions

12.12.1.1 Emergency Shutdown of Circulating Pump or Primary Circuit Pumps

12.12.1.2 Rupture of Steam Generating Plant Pipes

12.12.1.3 Termination of Feed Water Supply

12.12.1.4 Termination of Electric Energy Supply

12.12.1.5 Termination of Steam Withdrawal from Steam Generating Plant

12.12.1.6 Rupture of Main Steam Piping

12.12.1.7 Unintended Extraction of Most Effective Control from Reactor Core

12.12.1.8 Reactor Cold Water Injection

12.12.1.9 Primary Circuit Leakage (Accident with Coolant Loss)

12.12.2 Accidents on Board Vessel/Floating Facility

12.12.2.1 Collision (Hit at Reactor Compartment)

12.12.2.2 Grounding and Stranding

OPERATING MANUAL OF NUCLEAR-POWERED VESSEL (FLOATING FACILITY) NUCLEAR POWER UNIT (guidelines for composing the document)

The Operating Manual shall contain all information required for qualified personnel for safe operation of the vessel/floating facility and its nuclear power unit under the normal operating conditions, as well as the instructions regarding measures to be undertaken in case of certain emergencies.

The following data shall be specified in the Operating Manual:

1. Characteristics of the nuclear power unit with diagrams of the systems and other data related to radiation monitoring, biological shielding, fire protection and fire extinguishing means, spare parts.

2. Parameters for normal operation of the steam generating plant and associated systems, including rated and limiting values, as well as permissible deviations.

Among critical parameters the following shall be specified:

2.1 Duration of personnel stay in radiation zones

2.2 Radiation levels in certain zones

2.3 Activity levels for coolant in the primary and secondary circuits, as well as activity levels for liquid, solid and gaseous wastes.

3. Instructions for normal operating modes of the steam generating plant, such as startup, normal operation, power change, and disable, including the following data:

3.1 Functional tests of safety control systems and steam generating plant protection system prior to starting up and in the course of normal operation

3.2 Determination of critical position of the control cascades and reactivity values, as well as reactivity margin of the reactor core and its variation within the core service life;

3.3 Minimum admissible redundancy of the steam generating plant and energy supply equipment for reactor safe startup and operation. The equipment to be tested or repaired shall not be considered operational in evaluation of requirements for redundancy, except for those cases when equipment is made operational by certain test (for instance, generator set startup).

4. The operating instructions for certain emergency conditions with description of typical development of initial events, recommended trouble-shooting procedures, and further operation, if necessary.

5. The instructions for service organization on board the vessel/floating facility, including the following:

5.1 Manning and responsibility of people in charge of nuclear and radiation safety

5.2 Watches at sea and in harbor

5.3 Access to controlled area and containment

5.4 Training of personnel involved in operating the steam generating plant and practice alerts for crews

5.5 Requirements for vessel's documentation related to operating the steam generating plant and radiation situation on board the vessel /floating facility, as well as forwarding reports on equipment failures and emergencies.

6. Instructions for surveying the steam generating plant, containment and hull structures, including data on test intervals, scopes and methods of tests.

7. In addition to any other instructions to be used to ensure safety on board the vessel/floating facility and environmental control the Operating Manual shall include the following instructions:

7.1 Docking and underwater surveys related to radiation safety of people

7.2 Radiation safety

7.3 Handling solid, liquid and gaseous radioactive wastes in storage and handing over (discharge)

7.4 Fire safety

7.5 Personnel actions in emergency situations that can affect safety of the steam generating plant, vessel/floating facility and environment

7.6 Loading, carrying and unloading hazardous cargoes

7.7 Administrative measures to be undertaken to prevent possible intervention during inspection of reactor protection system components.

NUCLEAR-POWERED STEAM GENERATING PLANT. CONTAINMENT LEAK TIGHT CIRCUIT COMPONENTS. PROCEDURE FOR CALCULATING LEAK TIGHTNESS STANDARD VALUES

1 SCOPE

1.1 This Procedure covers containment leak tight circuit components for nuclear-powered steam generating plants of vessels and floating facilities. This Document is to be used for design development and establishes procedure for calculating leak tightness standard values.

2 DESIGNATIONS

L_{PERM}: permissible relative leakage rate (%/day)

P_A: atmospheric pressure, Pa

$P_{\text{MAX. DESIGN-BASIS}}$: absolute air pressure equal to emergency fluid pressure in case of maximum design-basis accident (Pa)

P_1: absolute air pressure within containment in 24 test hours (Pa)

ΔP_P: permissible pressure variation for given L_{permiss}.(Pa)

P_{TEST}:absolute test pressure (Pa)

τ: time within which pressure changes by value of $\Delta P_{\text{permiss}}$ (s)

Q_{PERM}: permissible total air flow through miniature defects of containment leak tight circuit at pressure PMAX. DESIGN-BASIS (W (m^3Pa/s))

Q_i: air flow through one component of leak tight circuit PTEST (W)

V: containment volume, m^3

$в_\Sigma$: total leakage value (leak tightness standard) for the whole containment leak tight circuit (W)

$в_i$: leak tightness standard value for the component of leak tight circuit (W)

$в_{wj}$: leak tightness standard value for welded joint of leak tight circuit (W)

$в_{dji}$: leak tightness standard value for components of detachable joint tight circuit (W)

f_{fji}: leak tightness standard value for components of flange joint tight circuit (W)

$в_i^{sf}$ leak tightness standard value for the component of leak tight circuit with regard to safety factor k (W)

l_i: joint length on the leak tight circuit component (m)

n_i: number of leak tight circuit components

l_{wj}: length of welded joints (m)

l_{dj}: length of detachable joints (m)

l_{fj}: length of flanged joints (m)

n_{sf}: number of stop piping fittings communicating with air pressure under test (stop valves are included into leak tight circuit)

n_{ns}: number of sealings for cable/conductor penetrations

n_{cs}: number of cable sections.

3 TERMS AND DEFINITIONS

3.1 **R**elative leakage rate is a ratio of leakage rate (by weight/volume) to air mass/volume in the controlled volume at given initial parameters (pressure, temperature) expressed as a percentage per unit time (%/day).

3.2 Leakage rate means an air mass/volume escaped from the controlled volume per unit time in kg/h (m3/h) or kg/day (m3/day) at given initial parameters (pressure, temperature).

3.3 Air flow is an air consumption where air quantity is expressed as a product of volume and initial pressure drop, m3Pa/s (W).

Leak tightness standard value is an atmospheric air flow discharged into vacuum under normal conditions: t = 20°C, P_a = 101333 Pa (760 mm Hg), m^3Pa/s (W).

3.4 Leak tightness standard value is related to air flow according to the following formulae:

$$в_i = Q_i \frac{P_a^2}{P_{TEST}^2 - P_a^2} \tag{3.4-1}$$

or

$$Q_i = в_i \frac{P_{TEST}^2 - P_a^2}{P_a^2} \tag{3.4-2}$$

4 CALCULATION PROCEDURE

4.1 The permissible air pressure variation $\Delta P_{permiss.}$ at design values of $L_{permiss.}$ and $P_{max.\ design\text{-}basis}$ and assumption that $T_0 = T_1$ and P_a = const ($P_a = 1.0 \times 10^5$ **Pa**) is estimated by the following formula:

$$\Delta P_{permiss.} = L_{permiss.} P_{Mpa}/100. \tag{4.1}$$

4.2 The total permissible air flow $Q_{permiss.}$ through miniature defects of containment leak tight circuit will be calculated as follows:

$$Q_{permiss.} = \Delta P_{permiss.} V/\tau. \tag{4.2}$$

4.3 Leak tightness standard value $в_{\Sigma}$ of the containment leak tight circuit will be calculated as follows:

$$в_{\Sigma} = Q_{permiss.} P_a^2/(P_{MPa}^2 - P_a^2). \tag{4.3}$$

4.4 For normalized ratio in leak tightness standard value ($в_{\Sigma}$), see Table 4.4.

Table 4.4

Leak tightness standard value

Type of joint for the leak tight circuit component					
Welded joints	Detachable joints	Flanged joints	Stop valves	Sealings for cable/conductor penetrations	Cable section
$0{,}05в_{\Sigma}$	$0{,}32в_{\Sigma}$	$0{,}18в_{\Sigma}$	$0{,}19в_{\Sigma}$	$0{,}21в_{\Sigma}$	$0{,}05в_{\Sigma}$

4.5 Based on data in Table 4.4 leak tightness standard values are estimated as follows:

.1 For welded joints of leak tight circuit:

$$в_{wj} = 0{,}05в_{\Sigma}/l_{wj}; \tag{4.5.1}$$

.2 For components of leak tight circuit with detachable joint:

$$в_{dji} = \frac{0{,}32в_{\Sigma}}{l_{dj}} l_{dji}; \tag{4.5.2}$$

.3 For components of leak tight circuit with flanged joint:

$$в_{dji} = \frac{0,19в_{\Sigma}}{l_{dj}} l_{dji} ; \tag{4.5.3}$$

.4 For stop valves of the leak tight circuit component:

$$в_{sfi} = 0,18в_{\Sigma}/n_{sf} ; \tag{4.5.4}$$

.5 For sealings of cable/conductor penetrations of the component of the leak tight circuit

$$в_{sfi} = 0,21в_{\Sigma}/n_{sf} ; \tag{4.5.5}$$

.6 For cable sections per cable:

$$в_{csi} = 0,05в_{\Sigma}/n_{sf} ; \tag{4.5.6}$$

4.6 Safety factor of 0.1 (κ = 0.1) shall be taken into account to test calculated values more exactly. Then leak tightness standard value with regard to safety factor will be as follows:

$$в_i^{sf} = \kappa в_i. \tag{4.6}$$

5 EXAMPLE

5.1 Supposing that design parameters have the following values:

$L_{permiss.}$ = 1%/day; $P_{max.\ design\text{-}basis} = 5.0 \times 10^5$ Pa; V_f= 680 m^3; l_{wj} = 600 m; l_{dj} = 34.5 m; l_{fj} = 6 m; n_{sf} = 32 items; n_s = 6 items; n_{cs} = 800 items

5.2 Permissible air pressure variation ΔP_{PERM} is calculated by formula (4.1):

$$\Delta P = L_p P_{MPa} /100 = 1 \times 5 \times 10^5/100 = 5000 \text{ Pa}. \tag{5.2}$$

5.3 Total permissible air flow Q_{PERM} is calculated by formula (4.2):

$$Q_{permiss} = \Delta P_{permiss} V_f/f = 5000 \times 680/24 \times 3600 = 40 \text{ W (m}^3\text{Pa/s)}. \tag{5.3}$$

5.4 Leak tightness standard value of the whole leak tight circuit is calculated by formula (4.3):

$$в_{\Sigma} = Q_{\text{permiss}} P_{\text{a}}^2 / (P_{\text{MPa}}^2 - P_{\text{a}}^2) = 40 \frac{(1{,}0 \times 10^5)^2}{(5{,}0 \times 10^5)^2 (1{,}0 \times 10^5)^2} = 1{,}7 \text{ W}. \qquad (5.4)$$

5.5 With regard to Table 4.4. and formulas (4.3), (4.5.1) to (4.5.6), leak tightness standard value for welded joints of containment leak tight circuit is calculated as follows:

$$в_{wj} = 0{,}05 \times 1{,}7 / l_{wj} = 1{,}4 \times 10^{-4}; \qquad (5.5\text{-}1)$$

$$в_{wj}^{sf} = 0{,}1 \times 1{,}4 \times 10^{-4} = 1{,}4 \times 10^{-5} \text{ W}; \qquad (5.5\text{-}2)$$

.1 For the component of leak tight circuit with detachable joint (for example, main cover, $l_{mci} = 20$ m):

$$в_{mci} = \frac{0{,}32 \times 1{,}7}{34{,}5} 20 = 0{,}3 \text{ W}; \qquad (5.5.1\text{-}1)$$

$$в_{mci}^{sf} = 0{,}1 \times 0{,}3 = 3{,}0 \times 10^{-2} \text{ W}; \qquad (5.5.1\text{-}2)$$

.2 For the component of leak tight circuit with flanged joint (for example, fan-penetration joints $l_{fp} = 0.6$ m):

$$в_{tpi} = \frac{0{,}18 \times 1{,}7}{6} 0{,}6 = 0{,}03 \text{ W}; \qquad (5.5.2\text{-}1)$$

$$в_{tpi}^{sf} = 0{,}1 \times 0{,}3 = 3{,}0 \times 10^{-3} \text{ W}; \qquad (5.5.2\text{-}2)$$

.3 For stop valves of the leak tight circuit component:

$$в_{svi} = \frac{0{,}18 \times 1{,}7}{32} = 9{,}6 \times 10^{-3} = 0{,}03 \text{ W}; \qquad (5.5.3\text{-}1)$$

$$в_{svii}^{sf} = 0{,}1 \times 9{,}6 \times 10^{-3} = 9{,}6 \times 10^{-4} \text{ W}; \qquad (5.5.3\text{-}2)$$

.4 For sealings of cable penetrations

$$в_{sci} = \frac{0{,}21 \times 1{,}7}{6} = 0{,}06 \text{ W}; \qquad (5.5.4\text{-}1)$$

$$в_{sci}^{sf} = 0{,}1 \times 0{,}6 \times 10^{-3} = 0{,}6 \times 10^{-3} \text{ Вт}; \qquad (5.5.4\text{-}2)$$

.5 For cable sections per cable:

$$в_{csi} = \frac{0{,}18 \times 1{,}7}{6} = 1{,}4 \times 10^{-4} \text{ Вт}; \qquad (5.5.5\text{-}1)$$

$$в_{csi}^{sf} = 0{,}1 \times 1{,}4 \times 10^{-4} = 1{,}4 \times 10^{-5} \text{ W}; \qquad (5.5.5\text{-}2)$$

5.6 Determination of Standard Values for Bench Tests of Leak Tight Circuit Components

Example: The bench of internal void volume equal to $V_{void} = 2$ m^3 is made for main cover testing. Absolute air pressure at the beginning of the tests shall be taken to be $P_{test} = 2.0 \times 10^5$ Pa. Standard values for tests are calculated by formulas (3.4-1) and (4.2) as follows:

.1 Permissible air flow:

$$Q_{permiss.} = в_{mci}^{sf} \frac{P_{test}^2 - P_{\text{a}}^2}{P_{\text{a}}^2} = 3{,}0 \times 10^{-2} \times 3 = 0{,}9 \times 10^{-3} \text{ W}; \qquad (5.6.1)$$

.2 Permissible pressure drop for 1 test hour:

$$\Delta P_{permiss.} = Q_{permiss}\tau / V_{\text{c}} = 0{,}9 \times 10^{-3} \times 3600/2 = 1620 \text{ Pa}; \qquad (5.6.2)$$

.3 Test standard values:

Initial absolute pressure on the bench:

$$P_{test} = 2{,}0 \times 10^5 \text{ Pa}; \qquad (5.6.3\text{-}1)$$

test duration:

$$\tau = 3600 \text{ s (1 hour)}; \qquad (5.6.3\text{-}2)$$

Permissible pressure drop:

$$\Delta P_{permiss} = 1620 \text{ Pa.} \tag{5.6.3-3}$$

6 APPLICATION NOTE

6.1 Leak tightness standard values for welded joints shall be given in requirements of design documents on containment hull structures.

6.2 Leak tightness standard values shall be given in requirements of design documents on components of containment leak tight circuit with detachable and flanged joints.

LIST OF CIRCULAR LETTERS AMENDING/SUPPLEMENTING NORMATIVE DOCUMENT

Rules for Classification and Construction of Nuclear-Powered Vessels and Floating Facilities, ND No.2-020101-069-E

(Normative document No. and title)

Item No.	Circular letter No., date of approval	List of amended and supplemented paras
1	312-14-846c dated 09.10.2015	Part II: table 2.2; Part VIII: item 17.1.5; Part XII: item 4.3; Part XIII: items 3.3.2, 6.5, 7.7, 10.2

RUSSIAN MARITIME REGISTER OF SHIPPING

HEAD OFFICE

CIRCULAR LETTER	№312-14-846c	dated	09.10.2015

Re:

Rules for the Classification and Construction of Nuclear-Powered Vessels and Floating Facilities, 2012, ND No. 2-020101-069-E

Item of technical supervision:

Product, system

Implementation	from the date of publication
Valid: till	-
Validity period extended till	-
Cancels / Amends/ Supplements Circular Letter No. -	of-
Number of pages:	1+2
Appendices:	Amendments to the Rules for the Classification and Construction of Nuclear-Powered Vessels and Floating Facilities, 2012, ND No. 2-020101-069-E

Technical Director –
Director of Classification Directorate Vladimir I. Evenko

Amends — Rules for the Classification and Construction of Nuclear-Powered Vessels and Floating Facilities, 2012, ND No. 2-020101-069-E

It is necessary to do the following:

1. Bring the content of the Circular Letter to the notice of the RS surveyors and the interested organizations in the area of the RS Branch Offices' activity.
2. Be guided by the provisions of the Circular Letter in reviewing and approval of technical documentation.

Person in charge: V.V. Rozvorsky Department 110 +7 (812) 309-01-04

DMS "THESIS" No. 15-243826

Appendix to Circular Letter

No. 312-14-846c dated 09.10.2015

PART II. CLASSIFICATION

2 CLASSIFICATION SURVEYS OF NUCLEAR-POWERED VESSEL AND FLOATING FACILITIES IN SERVICE

Table 2.2 Symbol E shall be supplemented with the following text: ", when subject to thereof".

PART VIII. NUCLEAR-POWERED STEAM GENERATING PLANTS

17 SYSTEMS AND PIPING

Para. 17.1.5 shall be amended to read: All pipelines shall be connected by welding. Flanged and union joints shall be specially agreed upon with the Register provided that welding is not possible.

PART XII. RADIATION SAFETY

4 RADIOLOGICAL PROTECTION

Para. 4.3. "All sensors of radiation monitoring system devices shall be at least IP68 protected, the other equipment shall be IP23 protected.

PART XIII. PHYSICAL SECURITY

3 GENERAL REQUIREMENTS

Para. 3.3.2 shall be replaced by the following:
"3.3.2 Technical facilities of physical security include components and devices as a part of the following main systems:
.1 Intrusion protection system;
.2 Security alert system;
.3 Access monitoring and control system;
.4 Optoelectronic surveillance and situation assessment system;
.5 Operational communication and address system (including wire and radio communication means);
.6 Data protection system;
.7 Power supply and lighting system."

6 SECURITY ALERT SYSTEM

Section 6 shall be amended to read:
"6.5 Except for the above stated requirements, the technical means of security alert system shall meet the requirements of 7.3, Part XI "Electrical Equipment" of the RS Rules.".

7 ACCESS MONITORING AND CONTROL SYSTEM

Section 7 shall be supplemented with para. 7.7 reading as follows:
"7.7 Except for the above stated requirements, the technical means of access monitoring and control system shall meet the requirements of 5.10, Part XI "Electrical Equipment" and Section 7, Part XV "Automation" of the RS Rules.".

10 OPERATIONAL COMMUNICATION SYSTEM

Para. 10.2 shall be amended to read:

“10.1 Cable operational communication system shall meet the requirements of 7.2, Part XI “Electrical Equipment” and Section 7, Part XV “Automation” of the RS Rules and the requirements of Part IV “Radio Equipment” of the Rules for the Equipment of Sea-Going Ships”.

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8, Dvortsovaya Naberezhnaya,
191186, St. Petersburg,
Russian Federation
www.rs-class.org/ru/