SAFETY DESIGN FEATURES OF THE KLT-40S

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Russian Federation

I-1. Description of a nuclear installation with the KLT-40S reactor

The KLT-40S is a modular reactor unit developed for a pilot floating nuclear cogeneration plant (PATES, in Russian), currently under construction in Severodvinsk, the Russian Federation. The KLT-40S nuclear installation belongs to a class of pressurized-water reactors. The KLT-40S reactor unit is shown in Fig. I-1. Major specifications of the KLT-40S nuclear installation are given in Table I-1. Detailed design description of a floating NPP with the KLT-40S reactor installations is provided in [I-1].

Main design features of the KLT-40S are the following:

- Modular design of reactor unit: the reactor, the steam generators (SGs) and the main coolant pumps (MCPs) are connected with short nozzles, without using long pipelines;
- Four-loop reactor cooling system with forced and natural convection of the coolant in the primary circuit;
- Leak-tight primary circuit with canned motor pumps and leak-tight bellows-type valves;
- Once-through coil type SGs;
- Gas based pressurizer system in the primary circuit;
- Use of passive safety systems;
- Use of proven techniques for equipment assembly, repair and replacement; incorporation of proven diagnostics equipment and proven monitoring systems.

The KLT-40S core is based on marine reactor technologies and incorporates the materials that are exempted from the IAEA definition of direct use material.

To increase uranium fraction, a closely-packed assembly structure of the core is adopted, which provides maximum possible fuel volume in a given core volume. The core contains fuel rods with cylindrical claddings made of corrosion-resistant zirconium alloy. The fuel rods are similar to those of the ice-breaker reactors but incorporate the fuel with higher uranium fraction; such fuel is based on uranium dioxide granules in the inert matrix.

Each reactor unit of the floating nuclear power plant (NPP) is located in a containment that is a leak-tight physical barrier designed to limit the propagation of radioactivity and to localize the fission products in case of a loss of coolant accident (LOCA), using emergency containment cooling systems.

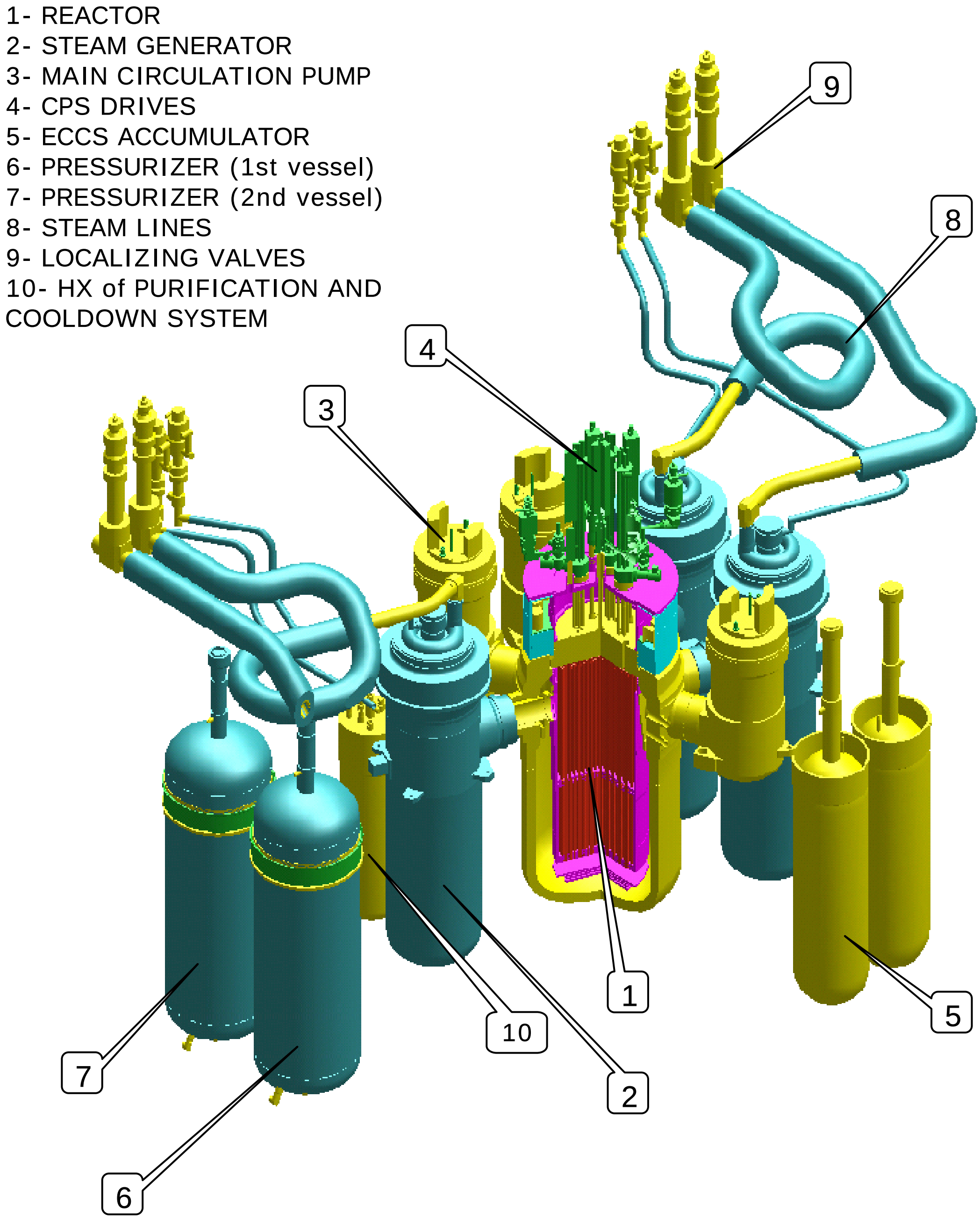
The containment is designed for internal pressure typical of the design basis accidents and the considered beyond design basis accidents, taking into account the emergency temperature conditions. The design value of the containment leakage rate ensures maximum possible limitation of the emergency planning area.

The containment along with the barge structures is designed for the design basis external impacts including a floating NPP sink.

Protection of the systems important for safety from external impacts is provided by a protective enclosure. The protective enclosure is a water- and gas-proof structure included in a ship-hull; it covers the containment and the liquid and solid radioactive waste storages, and provides the additional limitation of a leakage of the radioactive products to other parts of a floating power plant and to the environment, in case of a severe accident.

The containment and the radioactive waste storages are placed in a power compartment located in the middle part of the floating power unit.

A general view of the floating power module is shown in Fig. I-2.



CPS - control and protection system ECCS – emergency core cooling system HX – heat exchanger

FIG. I-1. General view of the KLT-40S nuclear installation.

The floating power unit (FPU) is a flat-deck non-self-propelled ship with a developed multi-level superstructure. An all-welded vessel of the floating power unit has ice reinforcements and special means for hauling and shoring. Nine waterproof bulkheads rising up to the top deck divide the FPU vessel into 10 impermeable compartments.

The floatability of the FPU is provided in case of flooding of any two adjacent compartments for all specification load cases satisfying the requirements of the Russian Marine Register

TABLE I-1. MAJOR SPECIFICATIONS OF THE KLT-40S POWER PLANT

CHARACTERISTIC	VALUE
Thermal power, MW	150
Primary circuit pressure, MPa	12.7
Coolant temperature, °C:	
- at core outlet	317
- at core inlet	279
Parameters of superheated steam downstream of the SG:	
- pressure, MPa	3.73
- temperature, °C.	290
Feedwater temperature, °C	170

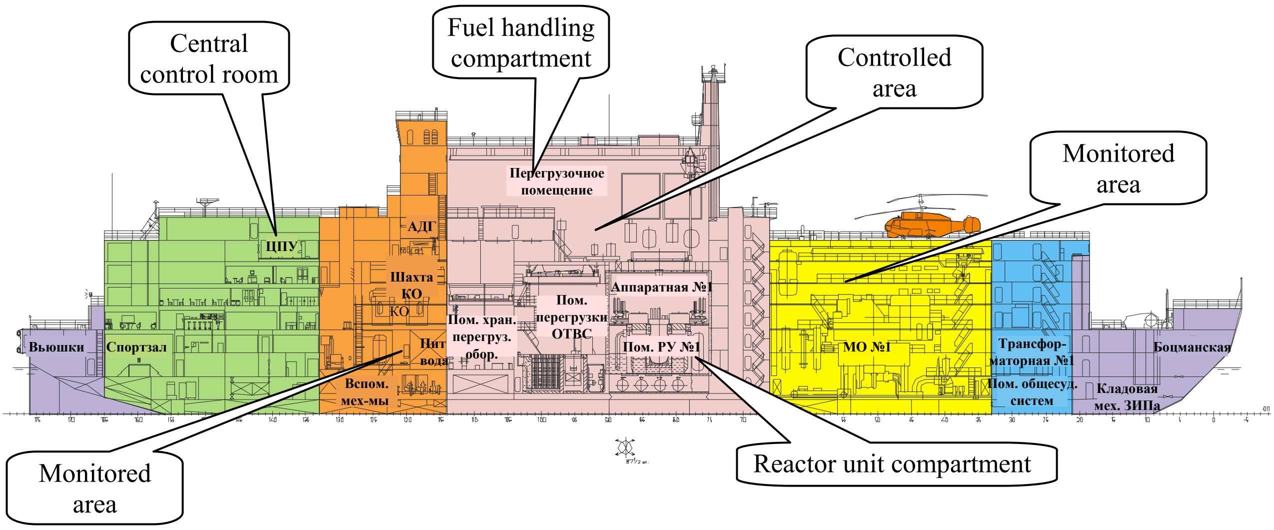


FIG. I-2. Floating power unit with two KLT-40S nuclear installations.

I-2. Passive safety design features of KLT-40S

Passive safety design features of the KTL-40S nuclear installation include both, inherent safety features and dedicated passive (safety) systems.

The so-called self-protection of a nuclear installation is expressed in its capability to prevent the occurrence and to limit the propagation and consequences of the initiating events, which could lead to accidents. The self-protection is, inter alia, achieved by the reliance on natural feedbacks and processes that require no operator intervention, no external power, and no assistance from the emergency teams for a certain period of time which could be used by the personnel to evaluate the situation and to undertake necessary corrective actions.

The self-protection of the KLT-40S is provided by the following features:

(a) Negative reactivity coefficients on fuel and coolant temperature and on specific volume of the coolant; negative reactivity coefficients on steam density and the integral power;

(b) High thermal conductivity of the fuel composition defining its relatively low temperature and, correspondingly, low stored non-nuclear energy;

(c) Adequate level of natural circulation flow in the primary system;

(d) High heat capacity of the nuclear installation as a whole, resulting from high heat capacity of the primary coolant and metal structures, from the use of a "soft" pressurizer system[1], and from a safety margin provided by the design for the depressurization pressure of the primary system under emergency pressure increase;

(e) Compact design of the steam generating unit with short nozzles between the main equipment items and with no large diameter primary pipelines;

(f) The use of restriction devices in the nozzles connecting the primary circuit systems to the reactor, which allows to limit the outflow rate in case of a break; the location of the connection nozzles is selected from the condition that they provide a fast transition to the steam outflow of the primary coolant in case of a break of the corresponding pipeline;

(g) Favourable conditions for the realization of a "leak before break" concept in application to the structures of the primary circuit, provided by design;

(h) The use of once-through steam generators, which limits the rate of heat removal via the secondary circuit in case of a steam line break accident.

The active and passive safety systems (see Fig. I-3) are incorporated in the design of the KLT-40S to carry out the following safety functions:

- Emergency shutdown of the reactor;
- Emergency heat removal from the primary circuit;
- Emergency core cooling;
- Localization of the released radioactive products.

Active safety systems

The KLT-40S nuclear installation incorporates the following active safety systems:

- System of reactor shutdown with shim control rod insertion in the electromotive mode;
- System of emergency reactor cooldown through the steam generator with steam dumping to a process condenser;
- System of emergency reactor cooldown through the heat exchanger of the purification and cooldown system;

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[1] "Soft" pressurizer system is characterized by small changes of the primary pressure under a primary coolant temperature increase. This quality, due to a large volume of gas in the pressurizing system, results in an increased period of pressure increase up to the limit value under the total loss of heat removal from the primary circuit. For KLT-40S, the corresponding time is not less than 1.5 hours since the accident start.

- System of emergency water supply from the emergency core cooling system (ECCS) pumps and the recirculation pumps;
- Filtration system for the releases from the protective enclosure.

Passive systems

The KLT-40S nuclear installation incorporates the following passive safety systems:

- System of reactor shutdown with insertion of the control rods into the core under the force of springs (scram rods) or gravity (shim control rods), when hold-up electromagnets of the control rod drives are de-energized;
- Passive system of emergency reactor cooldown through the steam generator;
- System of emergency water supply from the ECCS hydro-accumulators;
- The containment and the stop valves, normally in a closed position, located at the auxiliary systems of the primary circuit and the adjacent systems;
- Passive system of external cooldown of the reactor vessel;
- Self-actuated devices for start-up of the safety systems;
- Emergency containment cooling system;
- Protective enclosure.

Passive safety systems operate with natural circulation of the coolant or use the energy of a compressed gas.

The emergency heat removal system (EHRS) is intended to remove residual heat from the reactor in beyond design basis accidents involving NPP blackout and failure of active channels. The system includes two channels, consisting of two heat exchange loops each. The capacity of a single EHRS loop (~1% of the nominal reactor power) is sufficient to ensure reliable reactor cooldown and to maintain reactor pressure within the design limits.

Residual heat is removed by natural convection of the coolant in the primary and intermediate circuits and by the evaporation of water from the tank where heat exchangers-condensers (HXC) are located. Water reserve in the EHRS tanks ensures heat removal from the reactor during 24 hours.

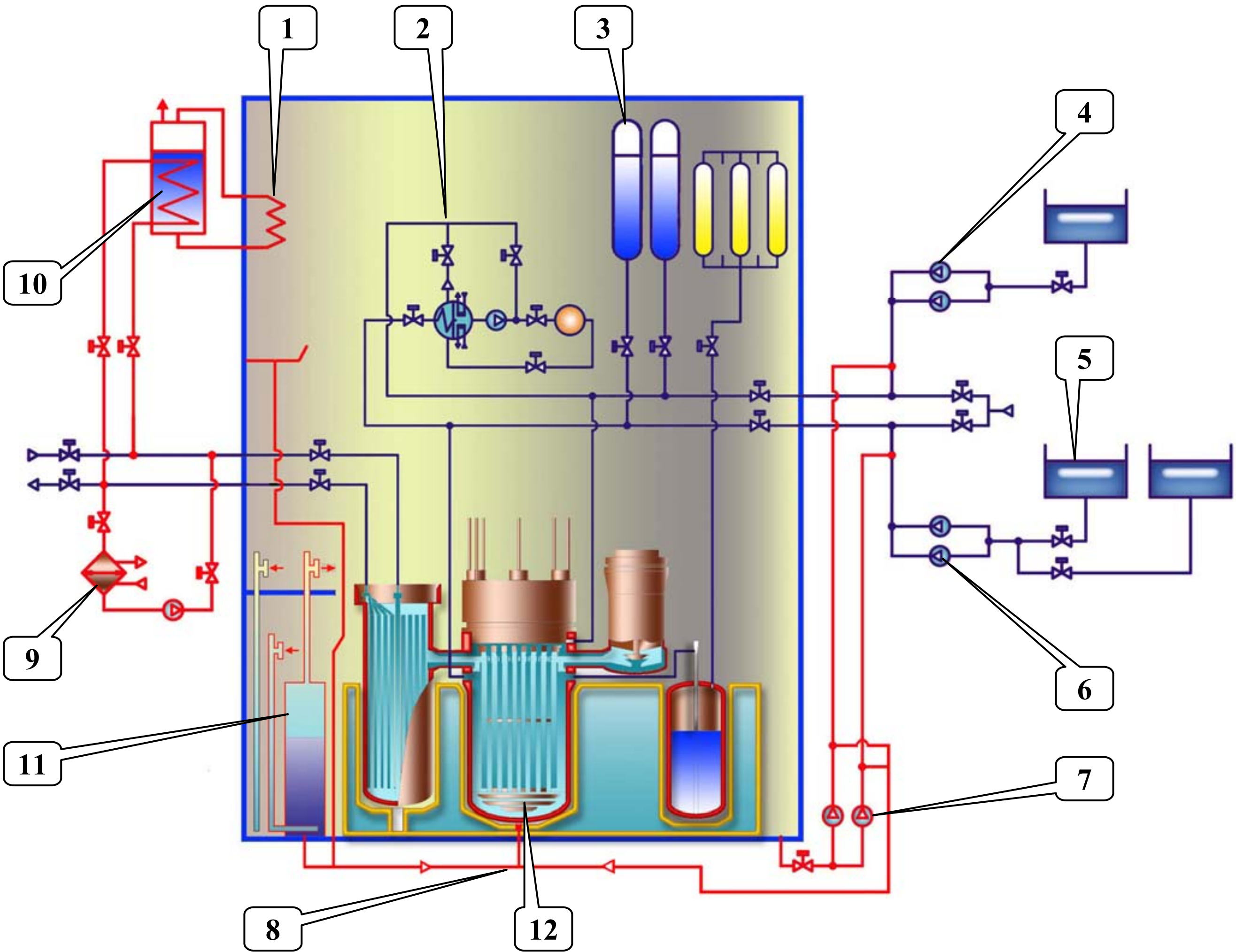
The prototypes of a passive EHRS are cooling systems used in the propulsion reactors. The effectiveness of such systems has been confirmed both by experiments at the test facilities and tests at the operating plants.

The majority of the KLT-40S safety systems employ a two-channel scheme with internal reservation of active elements such as valves and pumps. Using a two-channel scheme of the safety systems in the specific conditions of a floating structure (where it is necessary to save room space and equipment weight as compared to land-based NPPs) allows reducing the amount of bulky equipment such as tanks and heat exchangers.

The elements of both active and passive safety systems belong to the second safety class, according to the top-level Russian regulation OPB-88/97.

The requirements to manufacturing technologies of the devices and equipment for the active and passive safety systems correspond to the regulatory requirements in the nuclear energy area.

For the floating NPPs, specific regulations have been developed and adopted in the Russian Federation, in particular, “The rules of arrangement and safe operation of the equipment and items for light water reactors of the floating nuclear power plants (NP-062-05)”.



1- CONTAINMENT COOLING SYSTEM;
2-PURIFICATION AND COOLDOWN SYSTEM
3-ECCS ACCUMULATORS;
4, 6-ACTIVE ECCS;
5-ACTIVE ECCS TANK;
7-RECIRCULATION SYSTEM PUMPS;
8-RVCS;
9-ACTIVE EHRS;
10-PASSIVE EHRS;
11-CONTAINMENT BUBBLING SYSTEM;
12-REACTOR

ECCS – emergency core cooling system RVCS – reactor vessel cooling system
EHRS – emergency heat removal system

FIG. I-3. Safety systems of KLT-40S.

I-3. Role of passive safety design features in the defence-in-depth

Safety of small sized heat and power plants with the KLT-40S reactors is ensured by the incorporated defence-in-depth strategy. It includes a strategy of accident prevention and mitigation, and envisages the use of a system of physical barriers on the possible pathways of propagation of the ionizing radiation and radioactive materials to the environment. The incorporated defence-in-depth strategy also provides for the use of a system of technical and organizational arrangements to protect the barriers and retain their effectiveness, and includes measures on protection of the personnel, population and environment.

The structure of the defence-in-depth is based on the recommendations of IAEA [I-2, I-3], providing for the following levels:

Level 1 – Prevention of abnormal operation and failure.

Level 2 – Control of abnormal operation and detection of failure.

Level 3 – Control of accidents within the design basis.

Level 4 – Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents.

Level 5 – Mitigation of radiological consequences of significant release of radioactive materials.

The role of inherent and passive safety features and of active and passive safety systems of the KLT-40S nuclear installation at certain levels of the defence-in-depth is highlighted in brief below.

Level 1: Prevention of abnormal operation and failure

Inherent safety features contributing to this level are the following:

- Negative reactivity coefficients on fuel and coolant temperature and on specific volume of the coolant; negative reactivity coefficients on steam density and the integral power, in the whole range of reactor operation parameters;
- High thermal conductivity of the fuel composition defining its relatively low temperature and, correspondingly, low stored non-nuclear energy;
- The use of compact modular design of the steam generating unit with short nozzles between the main equipment, with no long and large diameter primary pipelines;
- The use of flow restriction devices to exclude large and medium break loss of coolant accidents (LOCAs), by design;
- Ultimately leak-tight design of the primary circuit based on welded joints, packless canned pumps, and leak-tight bellows-sealed valves;
- Favourable conditions for the realization of a “leak before break” concept in application to the structures of the primary circuit, provided by design;
- The use of gas pressurizer system that excludes failures of the electric pressurizer heaters;
- The use of a steam generator with lower pressure inside the tubes in the normal operation mode, that reduces the probability of a steam generator tube rupture (SGTR) accident.

Level 2: Control of abnormal operation and detection of failure

For Level 2, the contribution comes from active systems of the control, mitigation, protection and diagnostics that are used in the KLT-40S nuclear installation.

Level 3: Control of accidents within the design basis

For Level 3, the contribution comes from the following inherent and passive safety features, provided by design:

- Limitation of an uncontrolled movement of the control rods (e.g., due to external impact loads or a break of the control and protection system (CPS) drive casing) by an overrunning clutch; or by the movement limiters for an accident with the CPS drive bar break;

- The use of once-through steam generators, which limits the rate of heat removal via the secondary circuit in case of a steam line break accident.
- High heat capacity of the nuclear installation as a whole, resulting from high heat capacity of the primary coolant and metal structures, from the use of a “soft” pressurizer system, and from a safety margin provided by the design for the depressurization pressure of the primary system under emergency pressure increase;
- Installation of the restriction devices in the pipelines of the primary circuit systems and connection of these pipelines to the “hot” part of the reactor.

Also for Level 3, the following passive safety systems of the KLT-40S provide a contribution:

- Insertion of the scram control rods into the core by the force of accelerating springs;
- Insertion of the shim control rods into the core by the force of gravity;
- The use of a passive emergency heat removal system (EHRS), with natural convection of the coolant in all circuits and with the evaporation of water in the storage tanks;
- The level of natural convection flow in the primary circuit adequate for core cooling in the conditions when all MCPs are switched off;
- The use of self-actuating devices in the emergency reactor shutdown system and in the EHRS.

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

For Level 4, the contribution comes from the following inherent and passive safety features, provided by design:

- The protective enclosure provided by design;

Also for Level 4, the following passive safety systems of the KLT-40S provide a contribution:

- The ESSC hydro-accumulators that ensure time margin for accident management in case of a failure of the active ECCS systems;
- Passive system of the reactor vessel bottom cooling that ensures in vessel retention of the core melt;
- Passive containment cooling system, provided to reduce the containment pressure and to limit the radioactivity release.

Level 5: Mitigation of radiological consequences of significant release of radioactive materials

The mitigation of radiological consequences of significant release of radioactive materials is assumed to be provided mainly by administrative measures.

I-4. Acceptance criteria for design basis and beyond design basis accidents

I-4.1. Lists of design basis accidents and beyond design basis accidents

The lists of initiating events, design basis and beyond design basis accidents for a floating NPP with the KLT-40S nuclear installations have been developed on the basis of analysis of possible disturbances of normal operation caused by the equipment failures, personnel errors, internal and external impacts, also taking into account possible additional failures in the safety systems.

The basis for these lists was provided by the corresponding lists of initiating events and accident scenarios for a prototype ice-breaker reactor installation KLT-40; the KLT-40 lists were then modified taking into account changes in structures and systems made on transition to the KLT-40S reactor installation, as well as the experience in design and operation of relevant propulsion and land based NPPs.

The lists of initiating events and accidents adopted for the KLT-40S take into account typical lists given in the safety requirements of the IAEA Safety Standard NS-R-1 [I-2].

Classification of the initiating events is adopted in accordance with the OPB-88/97 terminology, taking into account that initiating events associated with an independent single failure of a safety system element may lead to a pre-accident situation (abnormal plant state with disturbance of the safe operation conditions that does not propagate into an accident) or to a design basis accident (abnormal plant operation with a release of radioactive materials beyond the design barriers).

In safety substantiation of the nuclear installation, all operating conditions of the reactor unit and the floating NPP were taken into account, including the start-up, the heat-up, power operation, refuelling, repair and maintenance, hauling, etc.

The list of initiating events of the pre-accident situations and design basis accidents is given in Table I-2. The list of beyond design basis accidents is presented in Table I-3.

TABLE I-2. CLASSIFICATION LIST OF THE INITIATING EVENTS OF PRE-ACCIDENT SITUATIONS AND DESIGN BASIS ACCIDENTS

CLASS OF INITIATING EVENTS	INITIATING EVENT
1. Faults in operation of reactor unit systems	
1.1. Disruptions of reactivity and core power distribution	1.1.1. Uncontrolled change of shim control rod group position 1.1.2. Main coolant pump (MCP) switching on with deviation from the instruction 1.1.3. Drop of one scram or shim control rod group 1.1.5. Faulty reactor shutdown 1.1.6. Faulty switching on of the standby cooldown pump 1.1.7. Disturbance of the design configuration of control rods of the control and protection system (CPS) at power operation
1.2. Increase of heat removal from the primary circuit	1.2.1. Decrease of feedwater temperature 1.2.2. Increase of feedwater flow 1.2.3. Increase of steam flow (opening of a dump valve and its failure to close, actuation of a safety valve on the steam line and its failure to close) 1.2.4. Guillotine break of the main steam line 1.2.5. Small break of the main steam line 1.2.6. Faulty switching on of the emergency heat removal system (EHRS) channels
1.3. Decrease of heat removal from the primary circuit	1.3.1. Decrease of steam flow (one or two of the SGs switching off; malfunctions in control system; turbo-generator failure; failure of the main condenser) 1.3.3. Decrease of feedwater flow (closure of a feedwater valve; stop of the feedwater pumps) 1.3.4. Termination of a feedwater flow 1.3.5. Guillotine break of the feedwater pipeline 1.3.6. Small break of the feedwater pipeline 1.3.7. Malfunction of equipment cooling by the third circuit

CLASS OF INITIATING EVENTS	INITIATING EVENT
1.3. Decrease of heat removal from the primary circuit (cont’d)	1.3.8. Disruption of heat removal to the outboard water (stop of the fourth circuit pump, break of the fourth circuit pipeline)
	1.3.9. Disconnection of high pressure gas reservoirs (balloons) from the pressurizer in normal operation mode
	1.3.10. Drop of compressed air pressure in the valve-driving system
	1.3.11. Faulty disconnection of the purification and cooldown system
	1.3.12. Faulty disconnection of the cogeneration bleed-off
1.4. Loss of electric power sources	1.4.1. Partial loss of auxiliary power
	1.4.2. Total loss of auxiliary power (blackout of the two switchboards)
1.5. Decrease of the reactor coolant system flow rate	1.5.1. Transition of one or two of the MCPs from high speed to low speed (high speed “blackout”)
	1.5.2. Stop of one or two of the MCPs running at low speed
	1.5.3. Stop of one or two of the MCPs running at high speed
	1.5.4. Transition of four MCPs from high speed to low speed
	1.5.5. Stop of four MCPs
	1.5.6. Seizure of one MCP
1.6. Increase of the reactor primary coolant system inventory	1.6.1. Inadvertent operation of the make-up system
1.7. Loss of coolant accidents (LOCAs)	1.7.1. Guillotine break of the pressurizer surge line
	1.7.2. Guillotine break of the purification and cooldown system pipeline
	1.7.3. Guillotine break of the emergency core cooling system (ECCS) pipeline in a section which cannot be cut off
	1.7.4. Break of the CPS drive support (bar)
	1.7.5. Steam generator tube rupture
	1.7.6. Tube rupture in the heat exchanger of purification and cooldown system
	1.7.7. Tube rupture of the MCP cooler
	1.7.8. Leak of a cooler of the supports of the CPS drives
	1.7.9. Small primary circuit LOCA
	1.7.10. Faults in sampling and draining of the reactor coolant
	1.7.11. Rupture of the sampling pipeline outside the containment
2. Internal impacts	
2.1. Fires	2.1.1. Fires in the floating power unit (FPU) compartments
2.2. Flooding, steaming of the compartments	
2.3. Explosion of the gas balloons	
3. Accidents in a shutdown state	
3.1. Disruptions of reactivity & core power distribution	3.1.1. Drop of a “fresh” fuel assembly to a wrong place during refuelling

CLASS OF INITIATING EVENTS	INITIATING EVENT
3.2. Disruptions in heat removal	3.2.1. Total blackout during long-term cooling of the reactor unit
	3.2.2. Total blackout during the refuelling
	3.2.3. Total blackout during the equipment maintenance
	3.2.4. Termination of heat removal during the refuelling
	3.2.5. Termination of heat removal during the equipment maintenance
3.3. LOCAs	3.3.1. Guillotine break of the pressurizer surge line in reactor hot shutdown state
	3.3.2. Faults in sampling and draining of the reactor coolant
3.4. Disruption of water and gas chemistry in an opened reactor	
3.5. Fire in the reactor equipment compartment during the refuelling or maintenance	
4. Disruptions in nuclear fuel and radioactive waste handling	
4.1. Disruptions at the refuelling	4.1.1. Hang-up of a spent fuel assembly during the refuelling
	4.1.2. Hang-up of a container with spent fuel assemblies
	4.1.3. Drop of a spent fuel assembly
	4.1.4. Drop of a case with a spent fuel assembly
	4.1.5. Blackout of the refuelling equipment
4.2. Disruptions in nuclear fuel storage systems	4.2.1. Depressurization of a cooling circuit and a gas system of the spent fuel and solid waste storage
	4.2.2. Blackout of the cooling system of spent fuel assembly storage tanks or decrease of heat removal from the tanks
	4.2.3. Termination of heat removal from the spent fuel assembly storage tank
	4.2.4. Leak of a case in the spent fuel assembly storage tank
	4.2.5. Flooding or steaming of the storage tank and of the case with spent fuel assemblies
	4.2.6. Disruption of gas content conditions in the spent fuel storage
4.3. Release of radioactive fluids from the equipment and systems	4.3.1. Leaks of the pipelines and the equipment sealing:
	– Leak in the gas removal system;
	– Leak in the drainage and sampling system;
	– Leak in the zero-discharge technology system
	4.3.2. Disruptions during a reloading of the reactor coolant system filter, resulting in a release of the radioactive substances
5. External impacts on the FPU	
5.1. Taking place on the site, as a result of natural events	5.1.1. Break of the rigid mooring bars due to formation of an ice plug with subsequent FPU grounding under the impact of wind and rough water
	5.1.2. Earthquake
5.2. Taking place on the site, as a result of human-induced events	5.2.1. Explosion of an external source on the shore
	5.2.2. Explosion on a moored tanker
	5.2.3. Pressing of the mooring ship
	5.2.4. Break of the shore communication pipelines
	5.2.5. Helicopter crash-landing on the FPU
5.3. Taking place at hauling	5.3.1. Collision of the FPU with another ship
	5.3.2. Grounding

TABLE I-3. LIST OF BEYOND DESIGN-BASIS ACCIDENTS

GROUPS OF BEYOND DESIGN BASIS ACCIDENTS	REPRESENTATIVE SCENARIOS OF BEYOND DESIGN BASIS ACCIDENTS
1. Accidents at leak-tight reactor coolant system	
1.1. Accidents with disruption of reactivity	1.1.1. Inadvertent withdrawal of shim control rod groups driven simultaneously with normal or emergency speed 1.1.2. Inadvertent withdrawal of any of the two shim control rod groups accompanied by a failure of the system of detection and termination of control rod inadvertent movement and a failure of the control system of reactor shutdown on power and/or doubling period signal 1.1.3. Drop of one control rod group with failures in the CPS: failures of interlocks, failures of control rods movement algorithms, failure of the emergency reactor shutdown 1.1.4. Erroneous loading and operation of a fuel assembly in a wrong position 1.1.5. Break of a steam-line inside the containment
1.2. Anticipated Transients Without Scram (ATWS)	1.2.1. “Hang up” of all shim or scram control rod groups or failures of the control system of emergency reactor shutdown on all protection signals, at the following initiating events: (1) Termination of steam flow to the turbine (closure of valves on the main steam-lines); (2) Maximum increase of the steam flow in the secondary system (full opening of the safety valve and its seizure in this position); (3) Termination of the feedwater flow (full closure of the feedwater valve); (4) Switch off of all MCPs; (5) Total blackout of the two auxiliary power switchboards; (6) Inadvertent withdrawal of simultaneously driven control rod groups (at the reactor start-up or at power operation).
1.3. Disruption of heat removal with failures in the emergency heat removal system (EHRS)	1.3.1. Break of the feedwater line with a failure of the fourth circuit and a failure of the system of outboard water supply to process condenser 1.3.2. Break of the feedwater line with EHRS failure to start on automatic signals 1.3.3. Total blackout with failure of all emergency and backup alternate current (AC) sources 1.3.4. Termination of heat removal by the secondary circuit with inadvertent cut off of the high pressure gas balloons 1.3.5. Break of the feedwater line with complete failure of the reactor shutdown system 1.3.6. Partial blockage of the reactor coolant circuit or of the fuel assembly inlet
2. Loss of coolant accidents	
2.1. LOCAs inside the containment	2.1.1. Guillotine break of the reactor coolant system pipeline with a failure of the active ECCS subsystem 2.1.2. Guillotine break of the reactor coolant system pipeline with a failure of the passive ECCS subsystem (hydro-accumulators) 2.1.3. Guillotine break of a ECCS pipeline of one of the channels with a failure of the pumps of the second channel

GROUPS OF BEYOND DESIGN BASIS ACCIDENTS	REPRESENTATIVE SCENARIOS OF BEYOND DESIGN BASIS ACCIDENTS
2.1. LOCAs inside the containment (cont'd)	2.1.4. Guillotine break of a reactor coolant system pipeline with a double-end leak (failure of the cut-off valves of the purification system) and a failure of the active ECCS subsystem 2.1.5. Guillotine break of a reactor coolant system pipeline with a failure to cut off the high pressure gas balloons 2.1.6. Small LOCA with the total blackout, due to the loss of all AC sources 2.1.7. Guillotine break of a reactor coolant system pipeline with the total blackout, due to the loss of all AC sources 2.1.8. Guillotine break of a reactor coolant system pipeline with a failure to close the cut-off valves in the containment ventilation system on automatic signals 2.1.9. Rupture of a CPS drive support
2.2. Accidents with bypassing of the containment	2.2.1. SG tube rupture with a failure of the cut-off valves to close 2.2.2. Break of a steam-line – SG collector with a failure of the cut-off valves to close 2.2.3. Leak of a cooler of the support of CPS drives with a failure of the cut-off valves to close 2.2.4. Rupture of a tube of the MCP cooler with a failure of the cut-off valves to close 2.2.5. Rupture of a tube of the MCP cooler with a failure to cut off the high pressure gas balloons 2.2.6. Rupture of a tube in the heat exchanger of the purification and cooldown system with a failure to close the cut-off valves 2.2.7. Break of a cooling water outlet pipeline in the heat exchanger of the purification and cooldown system with a failure to close the cut-off valves 2.2.8. Rupture of a pipeline of the sampling system with a failure to close cut-off valves located on the lines of the sampling systems and the purification and cooldown system
2.3. Accumulation of a potentially explosive gas mixture in the reactor in an accident with diluent gas release outside the reactor primary coolant system	
3. Accidents in a shut down reactor; accidents during fuel handling	
3.1. Insertion of a positive reactivity	3.1.1. Inadvertent withdrawal of one shim control rod group during dismantling operations in the reactor
3.2. Disruption in heat removal from the reactor	3.2.1. Total blackout with a failure of all AC sources during the refuelling 3.2.2. Total blackout with a failure of all AC sources during the equipment maintenance (maintenance of the SG, MCP, cooldown system pumps, valves)
3.3. Depressurization of the primary circuit	3.3.1. Guillotine break of the pressurizer surge line in a hot shutdown state of the reactor with a failure of the ECCS active subsystem
3.4. Accidents during refuelling	3.4.1. Drop of a spent fuel assembly container: (1) Onto the reactor (2) Onto the spent fuel storage 3.4.2. Destruction of spent fuel assemblies as a result of an inadvertent closure of the container gate or an inadvertent turn of the aiming mechanism 3.4.3. Drop of a container with the case loaded by spent fuel assemblies 3.4.4. Drop of a container with the reactor coolant system filter

GROUPS OF BEYOND DESIGN BASIS ACCIDENTS	REPRESENTATIVE SCENARIOS OF BEYOND DESIGN BASIS ACCIDENTS
3.5. Accidents in spent fuel storage	3.5.1. Failure of a cooling system of the spent fuel storage tanks (all channels)
3.6. Release of radiolysis products from the opened reactor in an accident with loss of heat removal from the reactor (during the refuelling, during the equipment maintenance).	
4. External impacts on the FPU	
4.1. Collisions of the FPU with other ships having the speed above the critical value	
4.2. Fall of an aircraft onto the FPU from high altitude	
4.3. Sink of the FPU	
4.4. Grounding of the FPU, including that on a rocky ground	

I-4.2. Acceptance criteria for design basis accidents and beyond design basis accidents

Substantiation of the KLT-40S NPP safety in design basis and beyond design basis accidents has been performed on the basis of the safety assessment criteria (acceptance criteria) presented in Tables I-4 and I-5.

TABLE I-4. SAFETY ASSESSMENT CRITERIA FOR DESIGN BASIS ACCIDENTS

CRITERION NUMBER	CRITERION FORMULATION
1.	Maximum fuel temperature shall be below the melting point
2.	Specific threshold enthalpy of the fuel rods destruction shall not be exceeded
3.	Minimum value of the departure from nucleate boiling (DNBR) in the core shall be ≥ 1.0, taking into account the most unfavourable deviation of parameters, the maximum non-uniformity of power distribution, and the uncertainties of local power and critical heat flux calculation
4.	The core shall be covered by the coolant
5.	Maximum temperature of the fuel element claddings shall not exceed 500°C
6.	Primary circuit pressure shall not exceed 1.15 of the design pressure value
7.	Containment pressure shall not exceed 1.1 of the design pressure value
8.	Radiation doses for the population (critical group) at the control area* boundary and beyond this area shall not exceed the values that require taking a decision on the measures of population protection in the case of a radiation accident (the values that shall not be exceeded are specified in Tables 6.3 and 6.4 of the NRB-99 [I-4])
9.	Radiation dose to the personnel shall not exceed the dose value planned for liquidation of accident consequences; 100mSv, as established by the NRB-99 [I-4]
10.	Effective neutron multiplication factor (K_{eff}) of the fresh or spent fuel storage shall not exceed 0.95 in normal operation and in design basis accidents
11.	Maximum temperature of the fuel element claddings in spent fuel assemblies during a refuelling process or in the storage shall not exceed 650°C
12.	Pressure in the fuel storage tanks shall not exceed the limiting value of 1.4 MPa

* The control area boundary coincides with the FPU boards, to the bow and stern directions it coincides with the monitored area boundaries, see Fig. I-2.

TABLE I-5. SAFETY ASSESSMENT CRITERIA FOR BEYOND DESIGN BASIS ACCIDENTS

CRITERION NUMBER	CRITERION FORMULATION
1.	Radiation doses for the critical population group at the boundary of the area of emergency action planning and beyond this area shall not exceed the values that require taking a decision on the measures of population protection in the case of a radiation accident (the values that shall not be exceeded are specified in Tables 6.3 and 6.4 of the NRB-99 [I-4];beyond the area of emergency planning, temporary restrictions may be established on consumption of the local agricultural products)
2.	Radiation dose to the personnel shall not exceed the dose value planned for liquidation of accident consequences; 100mSv, as established by the NRB-99 [I-4]
3.*	Pressure in the primary circuit shall not exceed the value that ensures that the elastic deformation of the primary system components is preserved
4.*	Containment pressure shall not exceed the value that ensures that the elastic deformation of the containment system components is preserved
5.*	Time margin to the core uncovery shall be sufficient for the personnel to take actions on accident management (not less than 1 hour)
6.*	Maximum temperature of the fuel element claddings shall not exceed the value corresponding to cladding rupture (taking into account fuel burn-up)

* Additional criteria for the beyond design basis accidents not resulting in core damage

Table I-6 establishes a correspondence between the safety assessment criteria (acceptance criteria) and the design basis accidents.

Table I-7 establishes similar correspondence for the beyond design basis accidents.

TABLE I-6. CORRESPONDENCE BETWEEN SAFETY ASSESSMENT CRITERIA AND DESIGN BASIS ACCIDENTS

DESIGN BASIS ACCIDENT NUMBER ACCORDING TO TABLE I-2	CRITERION NUMBER ACCORDING TO TABLE I-4											
	1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.
1.1.1.	+	+	+		+	+						
1.1.2.	+	+	+									
1.1.3.			+									
1.1.4.			+									
1.1.5.	+		+									
1.1.6.			+									
1.1.7.	+		+		+							
1.2.1.-1.2.4.	+		+		+							
1.2.5.			+									
1.2.6.			+									
1.3.1.-1.3.12.			+			+						
1.4.1.-1.4.2.			+			+						
1.5.1.-1.5.6.			+									

DESIGN BASIS ACCIDENT NUMBER ACCORDING TO TABLE I-2	CRITERION NUMBER ACCORDING TO TABLE I-4											
	1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.
1.6.1.						+						
1.7.1.-1.7.11.			+	+	+		+	+	+			
2.1.1.	+	+	+			+	+					
2.2.	+	+	+			+	+					
2.3.						+	+					
3.1.1.			+									
3.2.1.			+			+						
3.2.2.-3.3.2.				+	+			+	+			
3.4.								+	+			
3.5.				+	+			+	+			
4.1.1.-4.1.5.								+	+		+	
4.2.1.								+	+		+	
4.2.2.											+	+
4.2.3.											+	
4.2.4.											+	
4.2.5.										+		
4.2.6.												+
4.3.1., 4.3.2.								+	+			
5.1.1.-5.3.2								+	+			

TABLE I-7. CORRESPONDENCE BETWEEN SAFETY ASSESSMENT CRITERIA AND BEYOND DESIGN BASIS ACCIDENTS

BEYOND DESIGN BASIS ACCIDENT NUMBER ACCORDING TO TABLE I-3	CRITERION NUMBER ACCORDING TO TABLE I-5					
	1.	2.	3.	4.	5.	6.
1.1.1.			+			+
1.1.2.			+			+
1.1.3.						+
1.1.4.	+	+				+
1.1.5.						+
1.2.1. 1)-6)			+			+
1.3.1.-1.3.6.			+			+
2.1.1.-2.1.9.	+	+		+	+	+
2.2.1.-2.2.8.	+	+			+	+

BEYOND DESIGN BASIS ACCIDENT NUMBER ACCORDING TO TABLE I-3	CRITERION NUMBER ACCORDING TO TABLE I-5					
	1.	2.	3.	4.	5.	6.
2.3.	+	+		+		
3.1.1.	+	+				+
3.2.1.	+	+			+	+
3.2.2.	+	+			+	+
3.3.1.	+	+			+	+
3.4.1.	+	+				+
3.4.2.	+	+				+
3.4.3.	+	+				+
3.4.4.	+	+				+
3.5.1.	+	+			+	+
3.6.	+	+				
4.1.-4.4.	+	+				

I-5. Provisions for safety under external event impacts

Structures, systems and components of a floating NPP with the KLT-40S nuclear installations are developed taking into account possible impacts of natural and human-induced external events, typical of a floating NPP location site and transportation routes, and meet the currently adopted regulatory requirements. The NPP safety is ensured at the specific values of the parameters of natural impacts on the NPP and reactor unit, determined in the design, that have a frequency of 10^{-2} $year^{-1}$; including the impacts of a design (10^{-2} $year^{-1}$ frequency) and maximum design (10^{-4} $year^{-1}$ frequency) earthquakes.

For the FPU location in Severodvinsk (the Russian Federation), the design earthquake magnitude is taken equal to 7, and the maximum design earthquake magnitude is equal to 8 on the MSK scale.

The equipment, machinery, and systems important for safety and their mounting are designed to withstand shock loads corresponding to peak ground acceleration (PGA) of 3g in all directions. Also, they remain operable under inclination and heaving, typical of the FPU operating conditions.

I-6. Probability of unacceptable radioactivity release beyond the plant boundary

Probabilistic safety parameters determined in the probabilistic risk assessment (PRA) of a floating NPP with the KLT-40S reactors are prescribed by a top-level Russian regulatory document, the OPB-88/97 [I-5]. The parameters include core damage frequency and the probability of a large (limit) radioactivity release in accidents.

According to OPB-88/97, the PRA goal is to demonstrate that the cumulative core damage frequency does not exceed 10^{-5} per reactor-year, and the probability of a large radioactivity release is not higher than 10^{-7} per reactor-year.

Level 1 PRA has been performed for a floating NPP with the KLT-40S nuclear installations. According to its results, point estimate of the resulting core damage frequency of the KLT-40S under internal initiating events is about 10^{-7} per reactor-year for the reactor initial conditions, corresponding to normal power operation. Uncertainty analysis of the probabilistic safety attributes, performed with a method of statistical testing (Monte-Carlo method), has shown an upper confidence boundary (95% quantile) for the core damage frequency to be not higher than 10^{-6} per reactor-year.

Low probability of a severe accident with core damage is conditioned by the inherent safety features (self-protection) and other design features of this modular reactor design, as well as by redundancy and diversity of the safety systems in the NPP. Both active and passive safety systems are incorporated in the KLT-40S; these systems are based on the components with high reliability proven by a multi-year operating experience of the prototype (marine) reactors.

I-7. Measures planned in response to severe accidents

In line with the state-of-the-art trends, an approach to severe accident management is based on a combination of the two categories of accident management measures:

- Those aimed at the prevention of core damage (decrease of core damage probability);
- Those aimed at the limitation of the severe accident consequences (accident mitigation).

Measures on the prevention of core damage

The analyses of more probable scenarios of accidents with a loss of core cooling, potentially resulting in core damage, and the PRA results show that the most critical LOCA scenario is that accompanied by a failure of the “normal” ECCS channels caused by the failure of the active elements (pumps or connecting valves of the same type).

To cope with such situation, the KLT-40S design provides for an option to supply water to the reactor via the pipelines of the purification system, using the turbine plant pumps.

Measures on accident mitigation include measures on limitation of the core damage fraction, measures on in-vessel retention of the corium, and measures on limitation of the radiological consequences.

Measures on limitation of core damage fraction

Core damage process in the KLT-40S nuclear installation is relatively slow due to the injection of water from the hydro-accumulator that cools the overheated and partially degraded core elements. Successful realization of the measures on water supply to the reactor at this stage of an accident will lead to the flooding and cooling of the core materials, and would allow preventing the formation of a molten pool upon the reactor bottom head and excluding an impact of the corium on the reactor vessel.

Measures on in-vessel retention of corium

For retention of the molten core inside the reactor vessel, a special system is provided for in the reactor unit design that secures external cooling of the reactor vessel in accidents with the core damage and core melt relocation to the reactor vessel bottom. In-vessel retention of the corium allows excluding from consideration negative phenomena associated with corium release to the containment.

Measures on limitation of radiological consequences

To exclude irradiation of the personnel and population in case of a severe accident, the following protective measures need to be implemented:

(1) To ensure protection of the personnel, it is necessary to exclude staff presence in the compartments adjacent to the containment and in other compartments with high radiation level;

(2) To limit radiation dose to the population living within a 1 km radius from the floating NPP, it may be required (depending on the actual radiation situation) that some protective measures, such as iodine prophylaxis or sheltering, are implemented. As a protection measure, temporary limitation should be established on the consumption of separate agricultural products grown in the area of up to 5 km radius from the floating NPP and contaminated by the radioactive products.

Evacuation of the population is not required at any distance from the floating NPP.

I-8. Summary of passive safety design features for the KLT-40S

Tables I-8 to I-12 below provide the designer's response to the questionnaires developed at an IAEA technical meeting "Review of passive safety design options for SMRs" held in Vienna on 13 – 17 June 2005. These questionnaires were developed to summarize passive safety design options for different SMRs according to a common format, based on the provisions of the IAEA Safety Standards [I-2] and other IAEA publications [I-3, I-6]. The information presented in Tables I-8 to I-13 provided a basis for the conclusions and recommendations of the main part of this report.

TABLE I-8. QUESTIONNAIRE 1 – LIST OF SAFETY DESIGN FEATURES CONSIDERED FOR/INCORPORATED INTO THE KLT-40S DESIGN

#	SAFETY DESIGN FEATURES	WHAT IS TARGETED?
1.	Negative reactivity coefficients on specific volume of the coolant, on fuel and coolant temperature and on reactor power in the whole range of variation of the reactor parameters	In reactivity initiated accidents: limitation of reactor power increase, ensuring reliable core cooling, prevention of pressure and temperature increase in the primary circuit
2.	Absence of liquid boron reactivity control system	Exclusion of inadvertent reactivity insertion as a result of boron dilution
3.	High thermal conductivity of the fuel composition (uranium dioxide granules in the inert matrix)	Prevention of the fuel element cladding temperature increase in loss of flow accidents; prevention of the primary pressure and temperature increase in accidents with disruption of heat removal
4.	Use of a gas pressurizer system	Exclusion of the electric heaters – a potentially unreliable component
5.	Insertion of scram control rods into the core by force of the accelerating springs	Increased reliability of a reactor shutdown
6.	Insertion of shim control rods into the core by gravity force (under their own weight)	Increased reliability of a reactor shutdown
7.	Use of a passive emergency heat removal system	Increased reliability of emergency heat removal

#	SAFETY DESIGN FEATURES	WHAT IS TARGETED?
8.	Adequate level of natural circulation flow in the primary system	Reliable core cooling
9.	Limitation of uncontrolled movement of the control rods by an overrunning clutch and by movement limiters, for an accident with a break of the CPS drive support bar	Decrease of a positive reactivity inserted under impact loads or under a break of the CPS drive casing, or under a break of the CPS drive support bar
10.	Use of self-actuating devices in the safety systems	Increased reliability of an emergency reactor shutdown; increased reliability of a start-up of the emergency heat removal systems
11.	Use of once-through steam generators	Limited increase of heat power removed by the secondary circuit in case of a steam-line break accident
12.	Use of a 'soft' pressurizer system	Damping of the transients; increased time margins for measures on accident management
13.	Provision of a mechanical strength margin on the primary pressure	Increased time margin for measures on management of accidents with heat removal disruption
14.	High thermal capacity of the primary system components	Increased time margin for measures on management of accidents with heat removal disruption
15.	Modular design of the reactor unit	Elimination of long pipelines in the reactor coolant system
16.	Leak-tight reactor coolant system	Decreased probability of loss of coolant accidents
17.	Favourable conditions for the realization of a "leak before break" concept in application to the structures of the primary circuit, provided by design	Reduced probability of a guillotine break for the primary pipelines
18.	Use of the restriction devices in the pipelines of the primary circuit systems	Limitation of the break flow in case of a pipeline guillotine rupture; less strict requirements to the ECCS
19.	Connection of the primary coolant systems to a 'hot' part of the reactor	Ensuring fast transition to a steam flow through the break in case of a pipeline rupture; limitation of the break flow; less strict requirements to the ECCS
20.	Use of hydro-accumulators in the ECCS	Providing time margin for the personnel to take actions on accident management in case of a failure of the active means of emergency water supply (pump failure)
21.	Use of a steam generator with lower pressure inside the tubes in a normal operation mode	Reduced probability of a steam generator tube rupture

#	SAFETY DESIGN FEATURES	WHAT IS TARGETED?
22.	Use of the secondary system pipelines designed for the primary pressure, up to the cut-off valves	Absence of coolant release in the case of a steam generator leak
23.	Use of a passive reactor vessel cooling system	In-vessel retention of the corium
24.	Use of a passive containment heat removal system	Reliable decrease of the containment pressure and limitation of radioactivity release in accidents
25.	Use of the protective enclosure	Limitation of radioactivity release in accidents; additional protection from the impacts of external events

TABLE I-9. QUESTIONNAIRE 2 – LIST OF INTERNAL HAZARDS

#	HAZARDS (SAFETY FUNCTIONS) THAT ARE OF CONCERN (RELEVANT) FOR A REACTOR LINE	HOW THESE HAZARDS (SAFETY FUNCTIONS) ARE ADDRESSED (PERFORMED) IN THE KLT-40S
1.	Prevent unacceptable reactivity transients	- Negative values of reactivity coefficients; Absence of liquid boron system; - Low velocity of control rod movement; minimized number of simultaneously driven control rod groups; - Limitation of uncontrolled movement of the control rods by an overrunning clutch or by movement limiters, for an accident with a break of the CPS drive support bar.
2.	Avoid loss-of-coolant	- Modular design of the reactor unit; elimination of long pipelines in the reactor coolant system; - Installation of the restriction devices in the pipelines of the primary circuit systems; - Connection of the primary coolant systems to a ‘hot’ part of the reactor; - Use of hydro-accumulators within the ECCS; - Use of coolant recirculation system.
3.	Avoid loss of heat removal	- Use of passive emergency heat removal system; - Redundancy of the active systems.
4.	Avoid loss-of-flow	- Adequate natural circulation flow in the primary system; - Redundancy of the circulation pumps; - Use of two coils in the MCP electric motor.
5.	Avoid exothermic chemical reactions	- It is ensured that thermal state of the fuel rods in emergency conditions excludes the exothermic reaction of zirconium oxidation by steam.

TABLE I-10. QUESTIONNAIRE 3 – LIST OF INITIATING EVENTS FOR ABNORMAL OPERATION OCCURRENCES (AOO) / DESIGN BASIS ACCIDENTS (DBA) / BEYOND DESIGN BASIS ACCIDENTS (BDBA)

#	LIST OF INITIATING EVENTS FOR AOO / DBA / BDBA TYPICAL FOR A REACTOR LINE (PWRS)	DESIGN FEATURES OF THE KLT-40S USED TO PREVENT PROGRESSION OF THE INITIATING EVENTS TO AOO / DBA / BDBA, TO CONTROL DBA, TO MITIGATE BDBA CONSEQUENCES, ETC.	INITIATING EVENTS SPECIFIC TO THIS PARTICULAR SMR
1.	Disruptions of reactivity due to control rod malfunctioning	- Negative values of reactivity coefficients; - Low velocity of control rods movement; minimized number of simultaneously driven control rod groups; - Two independent systems of reactivity control – shim and scram control rods; - Use of self-actuating devices – drive circuit breakers, self-actuated on primary pressure; - Mechanical strength margin on the primary pressure.	
2.	Reactivity disruption due to boron dilution	- Boric acid is not used for excess reactivity compensation.	
3.	Loss-of-flow due to pump coastdown	- Adequate (sufficient) natural circulation flow in the primary system; - Use of two coils in the MCP electric motor.	

#	LIST OF INITIATING EVENTS FOR AOO / DBA / BDBA TYPICAL FOR A REACTOR LINE (PWRS)	DESIGN FEATURES OF THE KLT-40S USED TO PREVENT PROGRESSION OF THE INITIATING EVENTS TO AOO / DBA / BDBA, TO CONTROL DBA, TO MITIGATE BDBA CONSEQUENCES, ETC.	INITIATING EVENTS SPECIFIC TO THIS PARTICULAR SMR
4.	Loss of primary system integrity (LOCAs)	- Modular design of the reactor unit; elimination of long pipelines in the reactor coolant system; - Connection of the primary coolant systems to a 'hot' part of the reactor; - Installation of the restriction devices in the pipelines of the primary circuit systems. See Table I-11	Specific initiating event for the KLT-40S is a break of the connection pipeline between the pressurizer and the gas balloons; Specific beyond design basis accident for the KLT-40S is a break of the primary circuit pipeline with a failure to cut off the gas balloons.
5.	Interfacing systems LOCA	- Up to the cut-off valves, the interfacing systems are designed for the primary pressure.	
6.	Loss of power supply	- Use of a passive emergency heat removal system providing the removal of heat during 24 hours.	
7.	Accidents due to external events	- Structures, systems and components of the floating NPP are designed taking into account possible impacts of natural and human-induced external events typical of a floating NPP location site and transportation routes, and meet the regulatory requirements.	

#	LIST OF INITIATING EVENTS FOR AOO / DBA / BDBA TYPICAL FOR A REACTOR LINE (PWRS)	DESIGN FEATURES OF THE KLT-40S USED TO PREVENT PROGRESSION OF THE INITIATING EVENTS TO AOO / DBA / BDBA, TO CONTROL DBA, TO MITIGATE BDBA CONSEQUENCES, ETC.	INITIATING EVENTS SPECIFIC TO THIS PARTICULAR SMR
8.		See Table I-11	Disconnection of gas balloons from the pressurizer during power operation
9.		See Table I-11	Explosion of gas balloons
10.		See Table I-11	Accidents connected with reactor placement on a non-self-propelled ship: - For DBA see item 5 in Table I-2 - For BDBA see item 4 in Table I-3

TABLE I-11. QUESTIONNAIRE 3 (PART 2) – DESIGN FEATURES OF THE KLT-40S THAT PREVENT PROGRESSION OF THE SPECIFIC INITIATING EVENTS TO A MORE SEVERE PHASE

SPECIFIC INITIATING EVENT FOR THE KLT-40S (SEE TABLE I-10)	DESIGN FEATURES THAT PREVENT PROGRESSION OF THE INITIATING EVENTS TO A MORE SEVERE PHASE
Disconnection of the gas balloons from the pressurizer during power operation	- Gas already present in the pressurizer ensures the absence of unacceptable pressure increase; - Availability of the warning and protection emergency signals on primary pressure increase (active systems); - Availability of self-actuating devices providing a reactor shutdown and a start-up of the passive EHRS.
Rupture of a pipeline connecting the gas balloons to the pressurizer	- A flow limiter is installed in the pressurizer surge line; - Availability of the cut-off valves ensuring a disconnection of the gas balloons and leak termination in the case of a break after the cut-off valves.
Explosion of the gas balloons	- Fire-extinguishing systems available in the protective enclosure and in the containment - Pressure sources that have pressure head higher than the design pressure of the balloons do not exist .

SPECIFIC INITIATING EVENT FOR THE KLT-40S (SEE TABLE I-10)	DESIGN FEATURES THAT PREVENT PROGRESSION OF THE INITIATING EVENTS TO A MORE SEVERE PHASE
Collision with another ship	- On-board protection structures available, including reinforced sheets of outer clothing and deck planking sheets adjacent to the board, as well as longitudinal stiffening ribs of the board
Sinking of the FPU	- System of containment flooding is available that prevents containment destruction by the external hydrostatic pressure; this system is provided to protect the environment from possible radioactive contamination in the case of a FPU sink.
Grounding of the FPU, including that on a rocky ground	- The bottom ceiling is isolated from the containment structures by horizontal crimps in the bulkheads
Helicopter crash-landing	- Protective structures consisting of steel planking and other structures of appropriate dimensions and strength are provided.

TABLE I-12. QUESTIONNAIRE 4 - SAFETY DESIGN FEATURES ATTRIBUTED TO DEFENCE IN DEPTH LEVELS

#	SAFETY DESIGN FEATURES	CATEGORY: A-D (FOR PASSIVE SYSTEMS ONLY), ACCORDING TO IAEA-TECDOC-626 [I-6]	RELEVANT DID LEVEL, ACCORDING TO NS-R-1 [I-2] AND INSAG-10 [I-3]
1.	Negative reactivity coefficients on specific volume of the coolant, on fuel and coolant temperature and on reactor power in the whole range of variation of the reactor parameters	A	1
2.	Absence of liquid boron reactivity control system (excess reactivity is compensated by a heterogeneous absorber in the burnable poison rods and by the CPS control rods)	A	1
3.	High thermal conductivity of the fuel composition (uranium dioxide granules in the inert matrix)	A	3
4.	Insertion of scram control rods into the core by force of the accelerating springs	D (by automatic system) C (by self-actuating devices)	3
5.	Insertion of shim control rods into the core by gravity force (under their own weight)	D (by automatic system) C (by self-actuating devices)	3
6.	Use of a passive emergency heat removal system	D (by automatic system) C (by self-actuating devices)	3
7.	Adequate level of natural circulation flow in the primary system	B	1
8.	Limitation of uncontrolled movement of the control rods by an overrunning clutch or by movement limiters, for an accident with a break of the CPS drive support bar	C	3
9.	Self-actuating devices in the safety systems	C	3
10.	Steam generators of a once-through design	A	1
11.	‘Soft’ pressurizer system	A	1, 3
12.	Provision of a mechanical strength margin on the primary pressure	A	1, 3
13.	Modular design of the reactor unit, eliminating long pipelines in the reactor coolant system	A	1
14.	Totally leak-tight reactor coolant system	A	1
15.	Installation of the restriction devices in the pipelines of the primary circuit systems	A	3
16.	Connection of the primary coolant systems to a ‘hot’ part of the reactor	B	3

#	SAFETY DESIGN FEATURES	CATEGORY: A-D (FOR PASSIVE SYSTEMS ONLY), ACCORDING TO IAEA-TECDOC-626 [I-6]	RELEVANT DID LEVEL, ACCORDING TO NS-R-1 [I-2] AND INSAG-10 [I-3]
17.	Hydro-accumulators in the ECCS	C	3
18.	Steam generator with lower pressure inside the tubes in a normal operation mode	A	1
19.	Passive reactor vessel cooling system	D	4
20.	Containment	A	3, 4
21.	Passive containment heat removal system	D	4
22.	Availability of the protective enclosure	A	4

TABLE I-13. QUESTIONNAIRE 5 - POSITIVE/ NEGATIVE EFFECTS OF PASSIVE SAFETY DESIGN FEATURES IN AREAS OTHER THAN SAFETY

PASSIVE SAFETY DESIGN FEATURES	POSITIVE EFFECTS ON ECONOMICS, PHYSICAL PROTECTION, ETC.	NEGATIVE EFFECTS ON ECONOMICS, PHYSICAL PROTECTION, ETC.
Absence of liquid boron reactivity control system	Decrease of the plant cost; operation simplification and cost reduction	Certain deterioration of fuel cycle characteristics
Use of passive systems		Increase of plant construction and maintenance costs
Use of self-actuating devices in safety systems		Increase of plant construction and maintenance costs
Modular design of the reactor unit	Compactness of the reactor unit, decrease of the containment dimensions, decrease of the plant cost	Certain deterioration of maintainability as compared to loop type plants
Totally leak-tight reactor coolant system	Decrease of the amount of radioactive waste, reduction of operation costs	

References

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[I-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, Safety Standards Series, No. NS-R-1, IAEA, Vienna (2000).

[I-3] INTERNATIONAL ATOMIC ENERGY AGENCY, Defence in Depth in Nuclear Safety, INSAG-10, Vienna (1996).

[I-4] Radiation safety regulations (NRB-99): Hygienic regulations, Ministry of Health (Minzdrav) of the Russian Federation, Moscow, 1999 (in Russian).

[I-5] General Principles of Safety Provision for NPPs, OPB-88/97. NP-001-97 (PNAE G-01-011-97). Moscow, Gosatomnadzor RF (1997).

[I-6] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Related Terms for Advanced Nuclear Plants, IAEA-TECDOC-626, Vienna (1991).