Annex to the letter
Date 21.11.2013 No. 13-13/2028

Answers for Austria (letters of July 20, 2010 and April 29, 2011)

Question: Melt localization unit (MLU): design, functionality backgrounds, possible design shortcomings.

Answer: The melt localization unit (MLU) has been designed for the Belarusian NPP with VVER-1200 reactors and intended to improve the energy unit safety in case of a heavy accident involving the core destruction and the melt propagating beyond the reactor vessel. The melt localization unit is one of technical components provided for heavy accident management. Safety becomes higher, because liquid and solid radioactive materials are prevented from propagation beyond the MLU and, as a result, the NPP accident localization zone remains undamaged.

In terms of the containment integrity, two specific time moments during the heavy accident process are especially dangerous:

– the reactor vessel destruction time;

– the time when the melt directly impacts the containment.

The reactor vessel destruction moment is specifically dangerous because of heavy thermal and mechanical impacts affecting the equipment and building structures that are located within the containment and that, while their destruction is in progress, dangerously affect the containment itself. The internal pressure of steam and gas in the reactor vessel and the melt temperature are the parameters determining the reactor vessel destruction process. The higher are these parameters, the higher is the impact on the containment from the melt itself and from the equipment and building structures destroyed by it.

The melt directly affects the containment only after the destruction of reinforced-concrete structures in the inner spaces along the melt path to the containment. This process can be rather long but quite dangerous for the containment integrity. If, under these conditions, no provisions are available to cool the melt, the containment will be destroyed in one or several points. These destructions can result from thermal, physical and chemical effects from the melt, such as radiation, chemical or thermal reaction between the melt and the containment materials.

Also, gases are emitted when the melt interacts with the structural materials and structures. These gases affect the containment in several ways:

– pressure grows in the containment;

– thermal and dynamic impacts on the containment grow as a result of diffusion combustion and explosion of gas mixtures;

– radioactive aerosols' carryover becomes more intensive;

– destruction of building structures becomes more intensive.

Thus, provisions must be made at the heavy accident's stage beyond the reactor vessel to localize the melt in order to prevent the melt impacts on the equipment and building structures within the containment and on the containment itself.

Functionality

In case of a heavy accident involving the destruction of the core and the reactor vessel, the melt localization unit shall keep the melt, the solid pieces of the damaged core, the pieces of the reactor vessel and the vessel internals.

The melt is localized and cooled for indefinite time within the under-reactor room of the concrete vault.

The system for passive heat removal from the containment must remove heat generated by the melt in the MLU during at least 24 hours after the accident onset. After this 24-hour period, water must be replenished in the emergency heat removal tanks; for this purpose, the mobile equipment and water reserves available at the NPP site must be used.

The sprinkler system must be appropriate to fill the vessel internals audit vault not later than 24 hours after the accident beyond the design basis and after restoration of electric power supply for water delivery into the MLU to the surface of the melt.

Operation modes

The MLU implements its functionality in case of a heavy accident involving the melt release.

Under other conditions, the MLU is in the standby mode.

The preliminary data describing the melt in terms of time, composition, weight and energy result from the analysis of the scenarios of heavy accidents involving the reactor vessel destruction.

The MLU design is appropriate to withstand the maximum design earthquake during the long-term melt cooling.

Operational requirements

The MLU meets the key requirements as follows:

- receives and keeps, within the MLU space, the melt, the solid core pieces and the reactor's structural materials;
- stably transfers heat from the melt to the cooling water;
- keeps the reactor vessel bottom (with the melt) if it is broken away or plastically deformed, until the time when the melt leaves the bottom;
- prevents the melt propagation beyond the set limits of the localization zone;
- ensures that the melt in the concrete vault remains subcritical;
- provides water supply to the concrete vault and steam removal from the concrete vault;
- minimizes the carryover of radioactive substances into the containment space;
- minimizes the hydrogen release;
- ensures that the stresses in the structures located in the under-reactor room of the concrete vault remain within their maximum permissible limits under various static and dynamic loads;
- minimum operating personnel intervention is necessary for the MLU operation.

Control

During a heavy accident involving the melt propagation beyond the reactor vessel, the parameters as follows remain controllable:

- temperature of the medium in the MLU;
- water level around the MLU casing.

The MLU operates as a passive system, and only minimum intervention of the operating personnel is necessary.

Tests and inspections

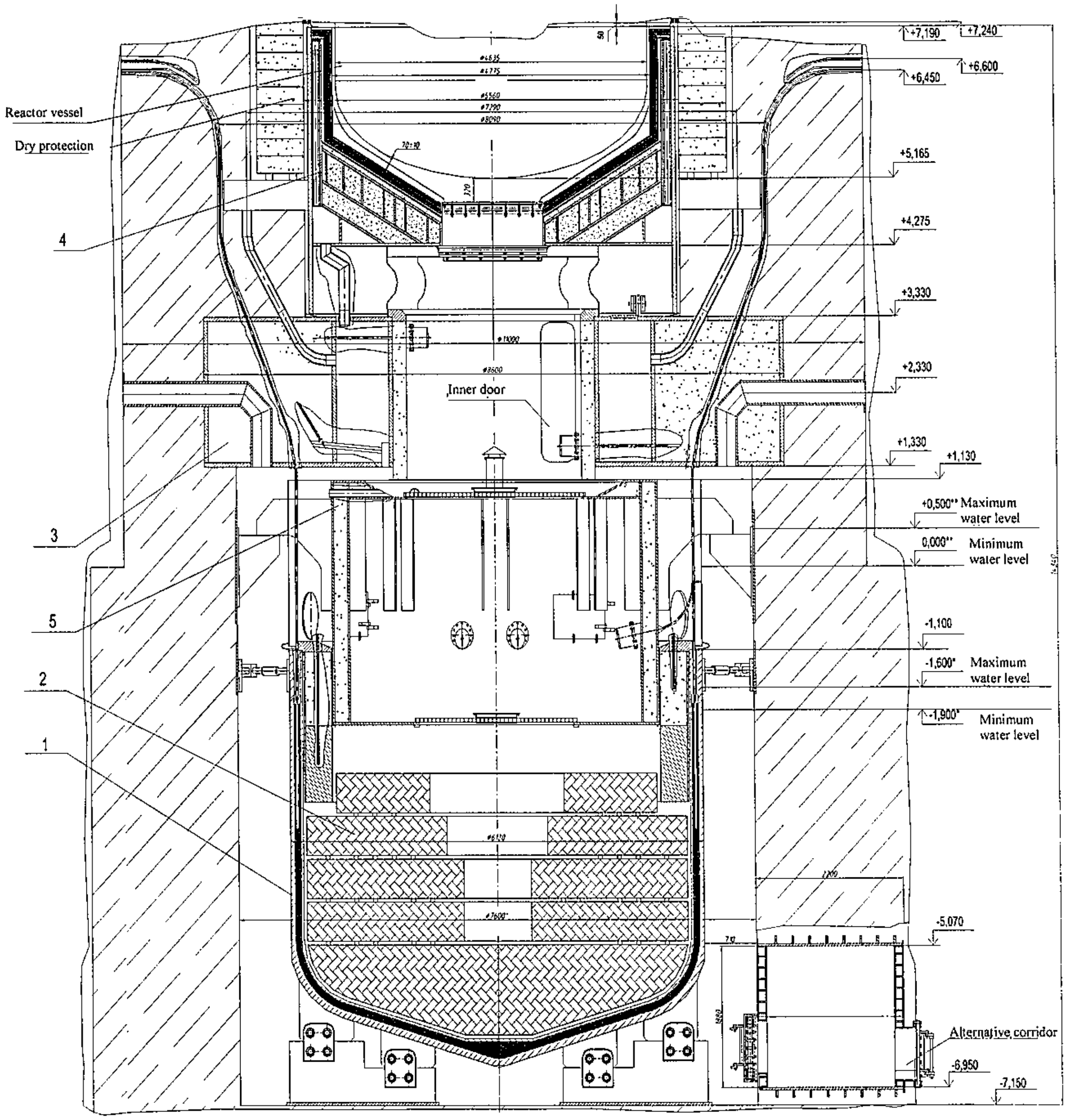
To audit the MLU condition, it remains accessible throughout the NPP normal operation period.

Design

See Figure 1 for the MLU design illustration. The MLU components are as follows (top down, along the corium movement direction from the reactor vessel to the concrete vault base):

- bottom slab;
- cantilever truss;
- service pad;
- filler;
- casing with supports.

Along with these primary components, the MLU includes cover pipes for instrumentation sensors: temperature sensors and water level sensors monitoring the water level around the MLU casing.



1. Casing with supports
2. Filler
3. Cantilever truss
4. Bottom slab
5. Service pad

Figure 1 – Melt localization unit

Casing with supports

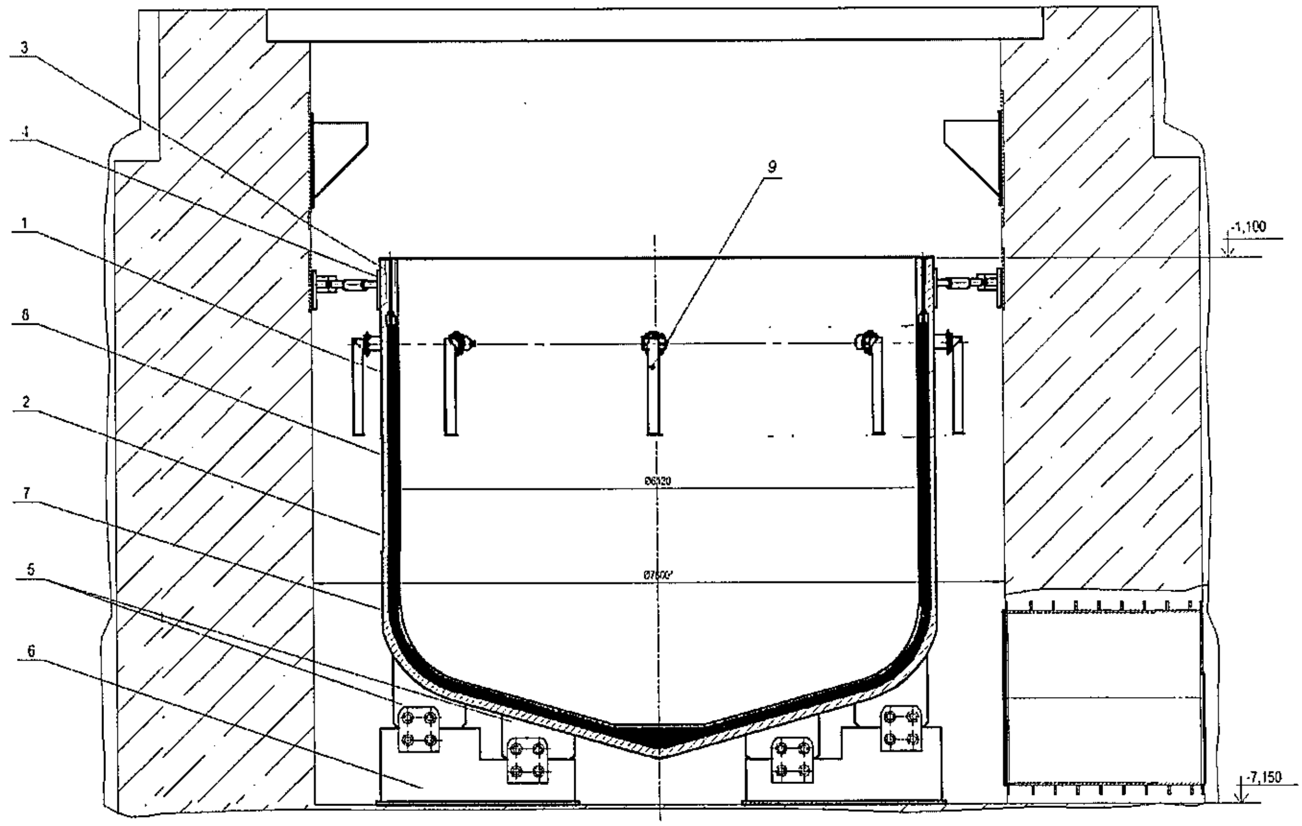
The casing with supports shall be used as a container for the filler to provide stable heat transfer from the melt, when it enters the MLU, to the water circulating between the casing and the concrete vault.

The casing with supports comprises the components as follows:

- outer casing;
- inner casing;
- common flange with holes for temperature sensors;
- ribs providing the gap between the outer and inner casing and fastening the filler;
- 16 lugs used to fasten the twin rods;
- 12 ribs welded to the casing bottom to fasten it on the supports;
- 4 lugs for transportation;

– casing supports;
– 8 passive water supply valves.

The gap between the outer and inner casing is filled with pellets composed of iron oxides and aluminium.



1- outer casing;
2- inner casing;
3- flange;
4- lug;
5- rib;
6- casing support;
7- pellets (iron oxides and aluminium);
8- cover pipe;
9- water supply valves.

Figure 2 – Casing with supports

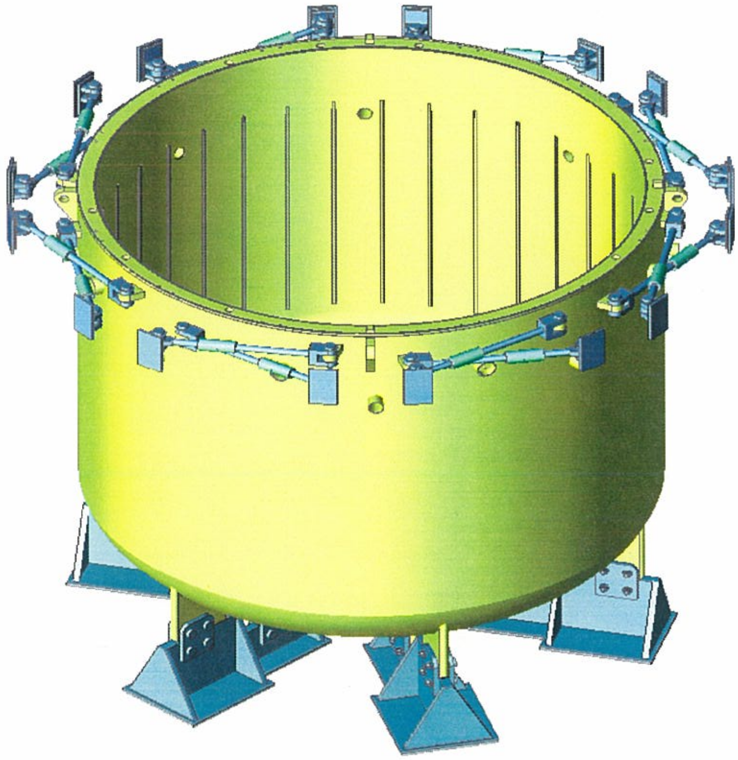


Figure 3 – Casing with supports and twin rods

Filler

The filler's purpose is to receive and distribute the melt.

The filler includes:

– five cartridge units;
– cartridge unit fastening assemblies;
– casing flange's thermal protection.

In terms of design, each cartridge unit is a cylindrical structure with a bottom, an external shell ring and a cover, containing plates made of iron oxides and aluminium, with a small amount of gadolinium oxide added to them.

The iron oxide – aluminium plates are equilateral triangles, with a side length 206 mm and an altitude length 50 mm. The special masonry cement mortar is used as a bed for the plates in the block. The first block's space is partially filled with concrete hematite grout. Each block, as a whole, is an integral load-bearing member.

The cartridge unit, type I, has a bottom matching the casing bottom; other blocks are provided with a through hole in their centers to distribute the melt throughout all cartridge units simultaneously and uniformly. Each cartridge unit is provided with six slots used to fasten the blocks to the MLU casing and to each other; for this purpose, fastening assemblies are used.

With the first block mounted, the gap between the block and the casing as well as six slots are partially filled with the concrete hematite grout.

The concrete thermal protection slab protects the upper part of the casing with the flange.

The results of interaction between the melt and the sacrificial materials are as follows:

– the temperature of heavily overheated metallic component in the melt is effectively reduced;
– the spatial density of energy release in the melt is reduced;
– the release of gases (including hydrogen), aerosols and radioactive nuclides into the containment is reduced;
– the release of thermal energy into the containment during the period immediately after the melt penetration is reduced;
– the melt remains subcritical.

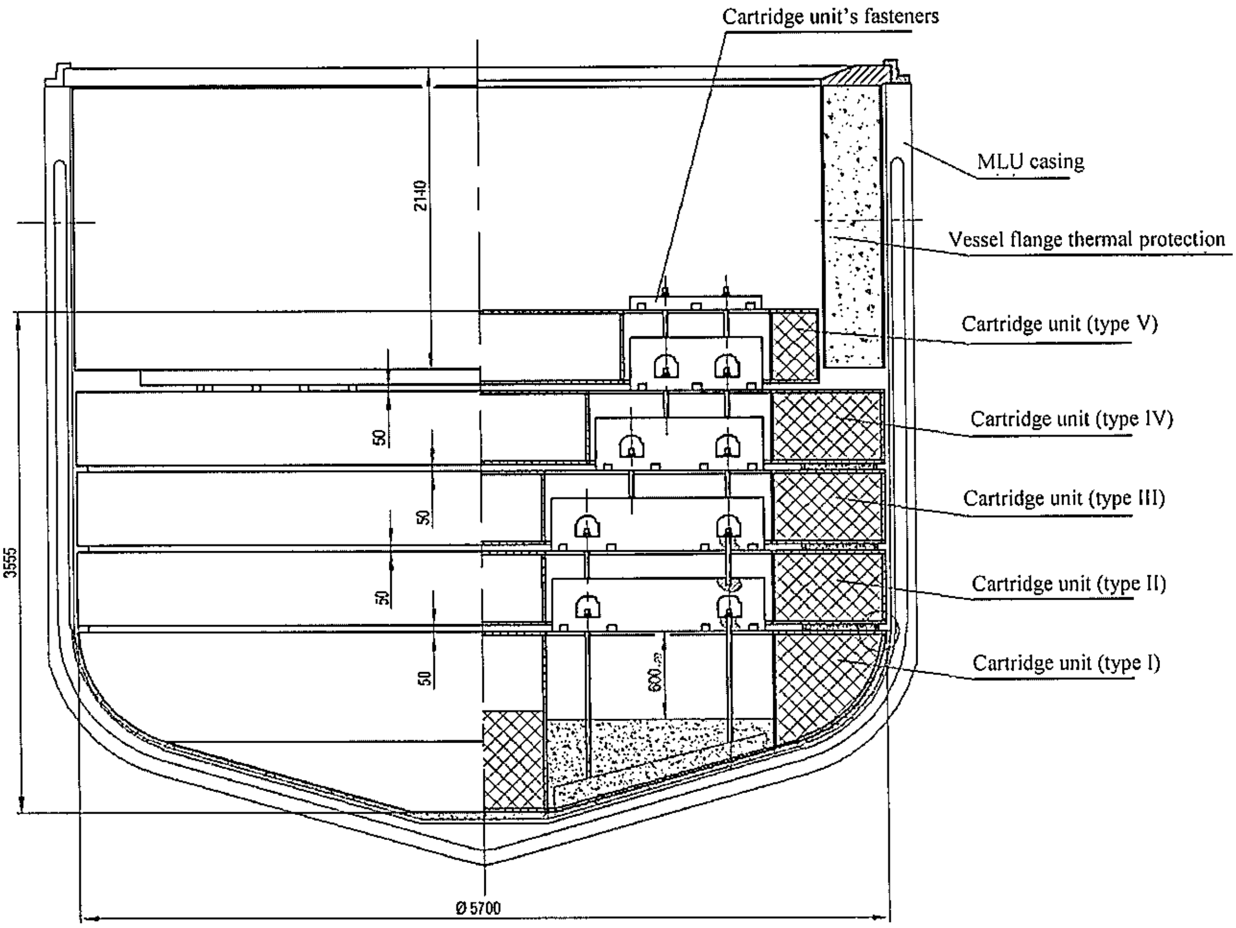


Figure 4 – Filler location in the casing

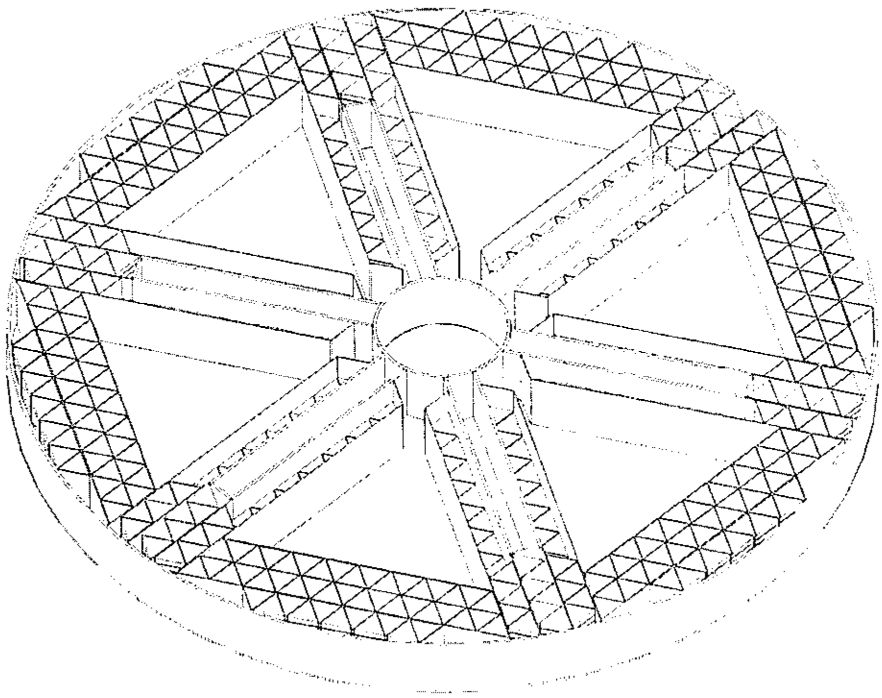


Figure 5 – Location of iron oxide – aluminium plates in the cartridge unit 2 and 3 (the unit cover is not depicted)

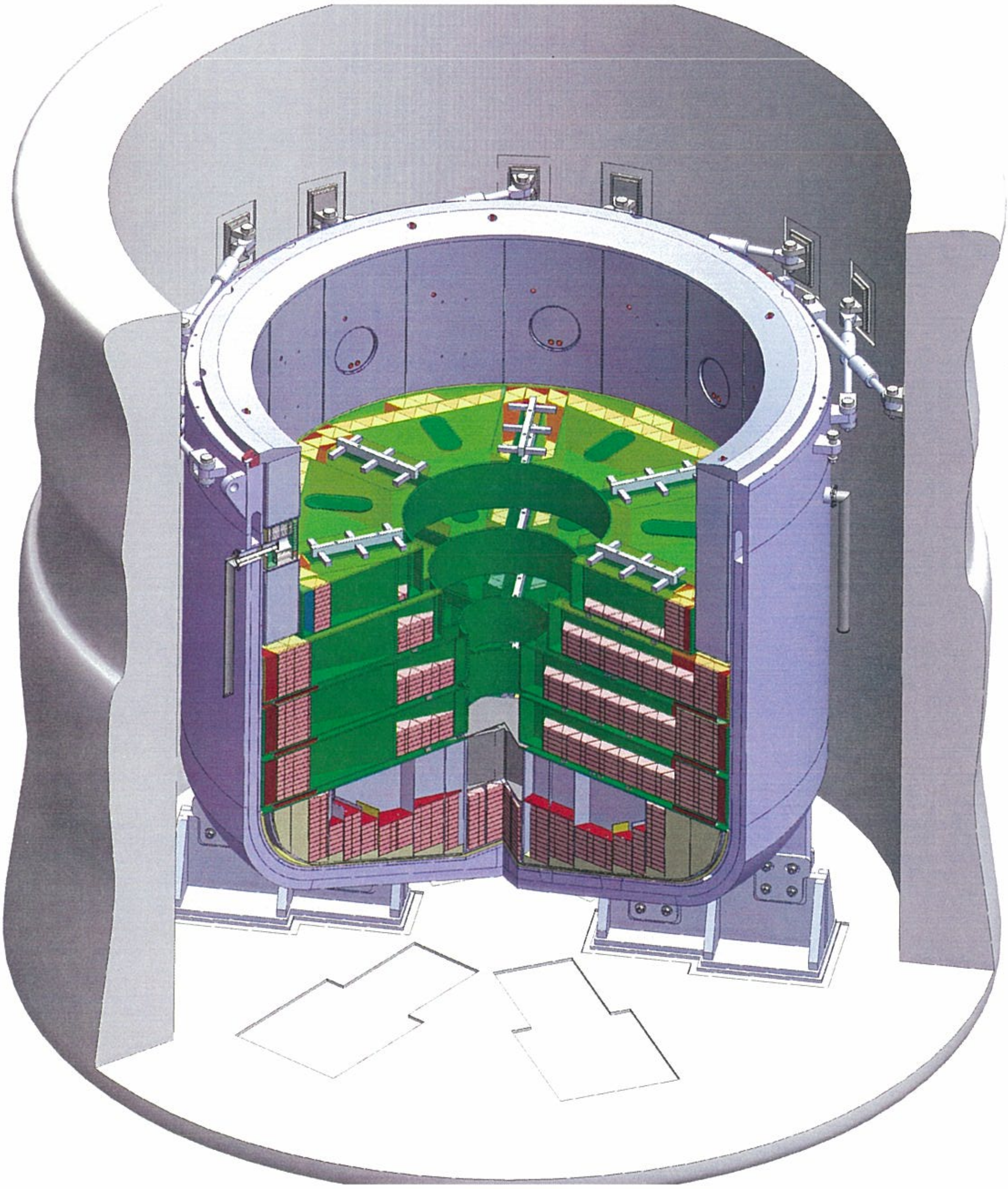


Figure 6 – Filler in the casing with supports

Service pad

The service pad is between the filler and the cantilever truss.

The service pad shall be used to isolate the filler from possible water penetration from the MLU and from the zone under the reactor and to release air streams when pressure becomes high. The service pad components are as follows:

- thermal protection;
- drainage pipelines;
- top and bottom pads.

The cover pipes passing through the service pad are used for temperature sensors mounted in the MLU casing, in the casing flange thermal protection and in the service pad itself.

Until the melt penetration, the filler is tightly covered by the service pad at the reactor vessel side, resulting in benefits as follows:

- no steam explosion at the time moment when the corium penetrates into the filler, due to water drainage from the service pad surface;
- filler remains intact throughout the period of normal operation conditions, in case of devia-

tions from normal operation conditions and in case of design basis accidents involving the reactor unit.

To provide free propagation of the melt into the filler, the service pad is made of thin ribbed steel sheet membranes easily destroyable by the melt.

Cantilever truss

The cantilever truss protects the MLU casing and the MLU communications against destruction by corium; also, the cantilever truss serves as a support for the bottom slab. It includes:

– the casing that includes the upper plate, the base, the middle ring (eight sectors), the inner ring and the outer ring;

– 15 support feet along the outer ring perimeter, rigidly fastening the cantilever truss to the concrete vault;

– the process corridor for the MLU audit and repair, including the airlock to make the bottom slab's air cooling system air-tight;

– the ventilation corridor;

– the outer door providing a pass between the airlock and the reactor compartment rooms;

– the inner door providing a pass from the airlock to the service pad for the equipment audit and repair;

– cover pipes used to connect the instrumentation sensors (the instrumentation sensors are the part of the monitoring and control systems, and they are installed after the complete MLU installation);

– corium spraying pipes used to connect the cooling water supply system from the vessel internals audit vault (the spraying pipes are used to deliver cooling water through the cantilever truss to the corium from above);

– pipes for steam removal, used to remove steam from the gap between the MLU casing and the concrete vault;

– air supply and withdrawal pipes connecting the bottom slab's air cooling system during normal operation.

For the cantilever truss location in the under-reactor room of the concrete vault, see Figure 7.

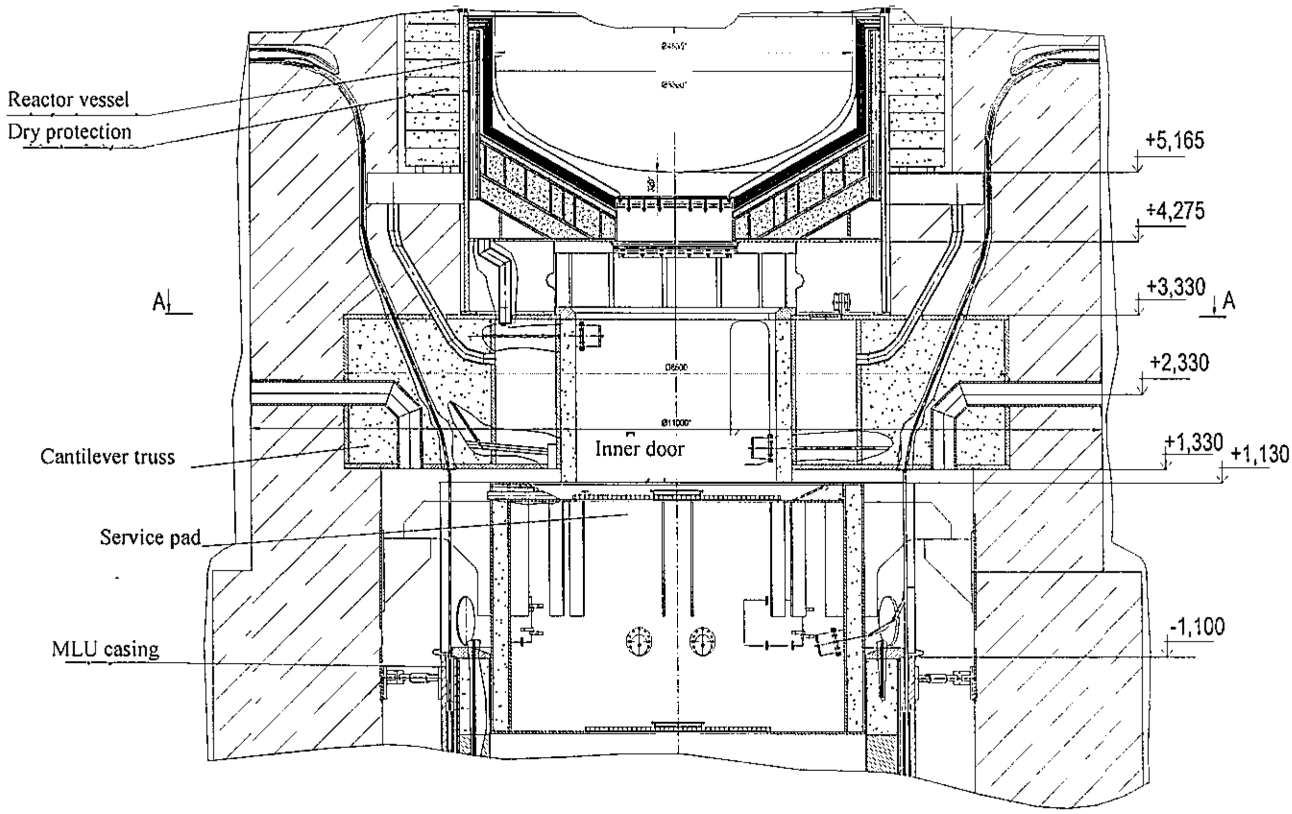


Figure 7 – Cantilever truss location in the concrete vault

Bottom slab

During normal operation, the bottom slab provides thermal insulation for the reactor vessel bottom and reduces neutron radiation and gamma radiation from the reactor vessel bottom.

In case of a heavy accident involving the reactor vessel destruction, the bottom slab receives the melt propagating from the reactor vessel and directs it to the casing with the filler. Also, the bottom slab takes the dynamic loads resulting from the reactor vessel destruction and from the melt discharge, and keeps the reactor vessel bottom if it is broken away or plastically deformed.

The bottom slab includes:

- the metal structure assembled from bearing ribs and shell rings;
- the vertical thermal insulation;
- the conical thermal insulation (concrete);
- the removable thermal insulation (thin stainless steel sheets);
- the biological shield made of lead boards.

The bottom slab is made as a funnel enveloping the reactor vessel lower part over the level of connection between the bottom and the cylindrical part.

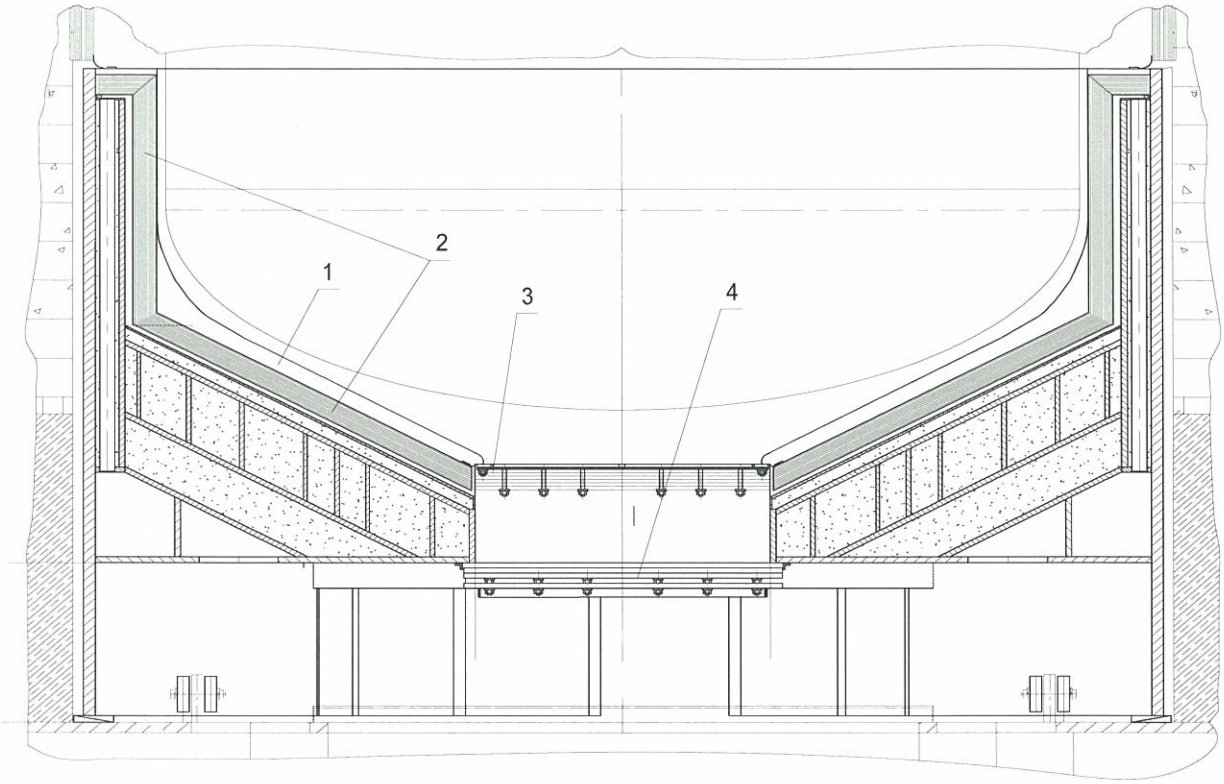
During the melt discharge, the bottom slab may be affected by relatively slow loading resulting from the plastic deformations of the casing and/or impacts arising when the casing cover is broken away by the residual pressure. These loads are taken by vertical bearing ribs protruding from the conical thermal insulation.

The melt flow over the conical part may be blocked by the solidified material. To provide stable melt runoff, the surface in this area is covered with the substrate made of special cement used as an easily fusible lubricant for the melt flow. To provide unrestricted corium runoff, the gap is available between the guiding unit and the lower part of the reactor vessel. The thermal protection made of heat resistant concrete, provided under the substrate, protects the load-bearing structures of the bottom slab against meltdown resulting from the melt runoff.

The guiding shell ring provides a hole in the central part of the bottom slab for the melt runoff. This shell ring restricts the diameter in which the melt drops can splash when the melt runs off.

The thermal protection made of heat resistant concrete, provided in the base of the bottom slab, acts as a thermal shield protecting the structures of the bottom slab against thermal radiation from the melt surface.

During normal operation, the bottom slab provides thermal insulation of the reactor vessel bottom and protects the under-reactor room of the concrete vault against neutron radiation and gamma radiation. For these purposes, the guiding shell ring is covered with the thermal insulation and the biological shield. The thermal insulation and biological shield are easily destroyable to make no barriers for the melt flow.



1 – metal structure; 2 – thermal insulation;
3 – removable thermal insulation; 4 – biological shield

Figure 8 – The bottom slab with thermal insulation and biological shield

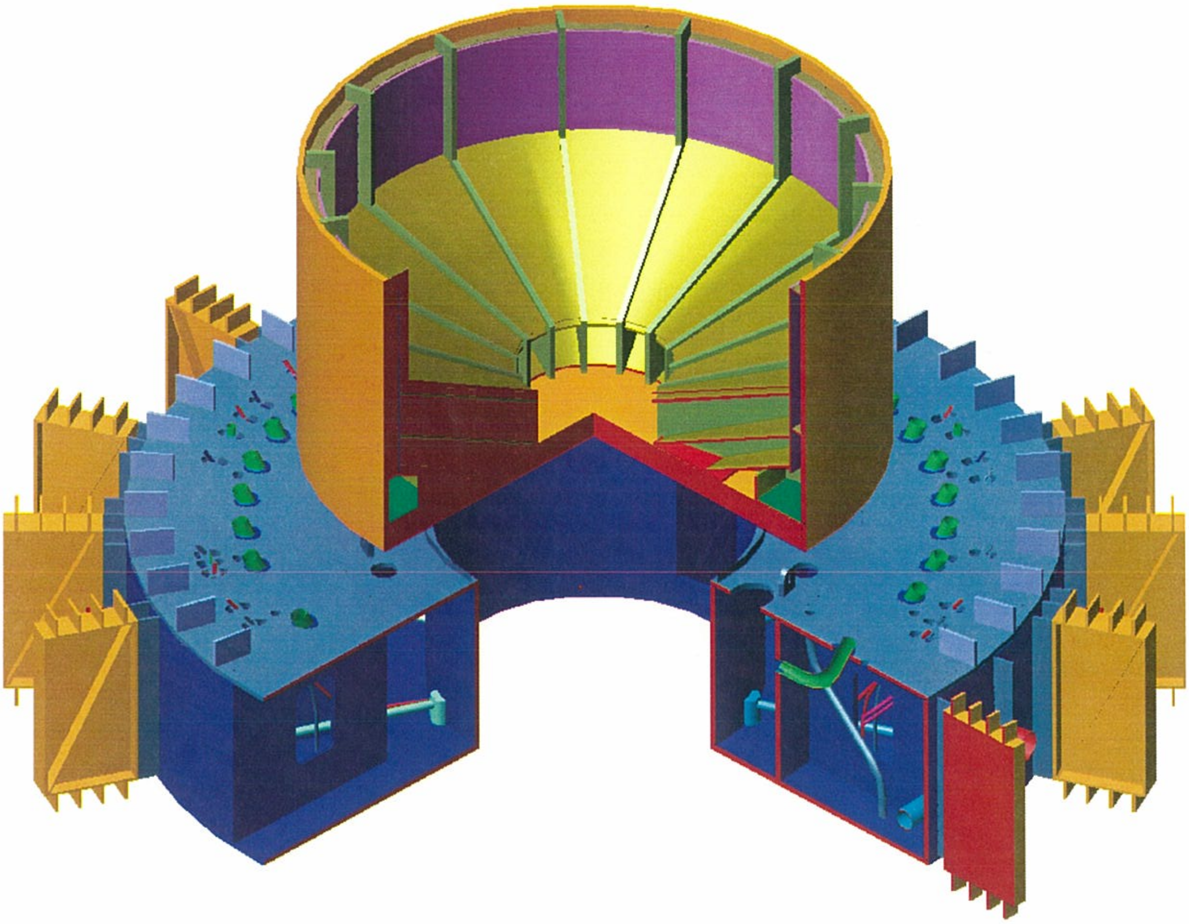


Figure 9 – The bottom slab and the cantilever truss

System control and monitoring

The melt localization unit's operation is linked with the systems as follows:

– the system for emergency water consumption from the vessel internals audit vault delivers borated water from the vessel internals audit vault to the MLU casing in case of heavy accidents involving the core melting;

– the emergency electric power supply system powers the remotely controlled fittings for emergency water consumption from the vessel internals audit vault and the sensors in the MLU parameters' monitoring system. The electric power supply is provided from the separate channel;

– the MLU parameters' monitoring system provides the operator with the information describing the MLU parameters during its intended operation, such as temperature in the concrete vault and in the MLU elements, water level in the MLU heat exchanger;

– the reactor vault cooling system delivers air to cool the reactor's and the bottom slab's dry protection system in the normal operation mode. In case of a heavy accident, the system does not affect any way the MLU operation.

System operation

Normal operation conditions

Under normal conditions for the reactor unit operation, the MLU is in the cold standby mode, and no special measures are necessary to switch it into the hot standby mode and the complete availability mode.

Violation of normal operation conditions

When the normal conditions for the reactor unit operation are violated, the MLU is in the cold standby mode, and no special measures are necessary to switch it into the hot standby mode and the complete availability mode.

Design basis accidents and beyond design basis accidents not resulting in the reactor vessel destruction

During the design basis accidents and the beyond design basis accidents during their in-vessel stages, the MLU is in the cold standby mode, and no special measures are necessary to switch it into the hot standby mode and the complete availability mode. After elimination of the design basis accident, if the absorbed dose rate is 100 mSv/hour or less, the MLU audit must be carried out followed by the relevant service if necessary.

Heavy accidents involving the reactor vessel destruction

The accepted melt localization strategy defines the MLU operation during heavy accidents involving the melt discharge. The important features of this strategy are as follows:

After penetration through the reactor vessel, the molten corium propagates into the space confined, from sides and from below, by the water-cooled steel walls of the MLU casing located in the under-reactor space of the concrete vault.

The MLU's water-cooled space is partially filled with the sacrificial materials consisting of the special composition of steel and relatively light and fusible oxides.

The molten corium propagating from the reactor into the MLU interacts with the sacrificial material; as a result, the heat removal conditions are optimized, the uncertainties resulting from diversity of heavy accident scenarios are smoothed, and the metallic and oxide components in the melt are inverted before water application to the melt surface.

The water supplied by gravity from the floor (0.0 level), from the sump tanks and from the vessel internals audit vault is used to cool the melt. The channels provided in the cantilever truss are used to remove the steam generated in the MLU to the containment space. The cooling water is available to deliver it to the MLU casing passively during 24 hours with the NPP completely de-energized. The water flowing to the concrete vault from the floor (0.0 level) and/or from the sump tanks is used to cool the MLU externally.

The MLU casing removes heat from the melt bath at the bottom and at the side. The special thermal protection shields and subsequent water flow discharged to the melt surface are used to protect the higher building structures against thermal radiation emitted from the melt surface.

Inversion of metallic and oxide components before water application to the melt surface ensures that there would be no steam explosions, because water application on the surface of molten oxides has been proven to be safe, as demonstrated by the results of researches.

The design is appropriate to ensure that there is no water in the MLU casing filler until the melt propagation into it.

Water supply valves are used in the design of the passive channel for water delivery onto the melt surface. When the specified temperature is reached in the installation position, the water supply valve is opened, and water runs from the concrete vault to the melt surface. The valve opening temperature and the valve position are chosen in such a way that the valve is opened after the melt inversion but before the destruction of metal structures in the MLU.

The temperature sensors are used to monitor the temperature in the bottom slab's inner cavity in order to detect the time moment when the melt penetrates into the MLU. If the cavity temperature exceeds 400°C, it means that the melt is in the MLU.

One hour after this time, water must be delivered from above (from the vessel internals audit vault) to the corium.

In case of the beyond design basis accident involving the reactor core melting, electric power supply shall be restored within 24 hours after the accident onset. For such a case, the design includes the provisions to replenish water in the vessel internals audit vault to proceed with water supply into the MLU casing onto the melt surface.

The processes accompanying the melt movement through the MLU, as well as thermal and chemical processes accompanying the melt interaction with the sacrificial material are described in details in the MLU technical design feasibility studies.

Safety estimate

The MLU operation is passive. The MLU components have been designed for operation throughout the full range of parameters and loads for all modes specified in the project. See the MLU technical design for the description of analytical calculations and experiments demonstrating that the MLU is effective.

ABBREVIATIONS

АЭС	NPP	Nuclear power plant
ВВЭР	VVER	Water-Water Power Reactor
ВКУ		Reactor vessel internals
КИП		Instrumentation
МРЗ		Maximum design earthquake
НУЭ		Normal operation conditions
ННУЭ		Deviations from normal operation conditions
ПОЖА		Plates made of iron oxides and aluminium
УЛР	MLU	Melt localization unit
КПВ		Water supply valve
ГОЖА		Granules made of iron oxides and aluminium

Question: Beyond design basis accidents: scenarios, process, background. Range of design basis accidents and beyond design basis accidents considered for the NPP safety analysis. Beyond design basis accident management and measures for minimization of environmental emissions in case of a beyond design basis accident.

Answer: The concept of management with regard to the beyond design basis accidents was finally formulated after Chernobyl disaster as the additional (fourth) level in the NPP defense in-depth strategy (OPB-88/97). For this purpose, the term "beyond design basis accident" was coined: this is an accident resulting from the initiating events not considered for the design basis accidents, or accompanied by additional (in comparison with the design basis accident) failures of safety systems beyond the single failure, or accompanied by the implementation of personnel's wrong decisions.

While the probability of such events is quite low, the up-to-date safety concept requires these accidents to be considered in the projects, limiting their consequences by way of measures for the beyond design basis accident management.

The beyond design basis accident management is a set of activities intended to prevent the design basis accidents from the development resulting in beyond design basis accidents and to mitigate the consequences of beyond design basis accidents. The equipment applicable for these activities includes any available operable equipment intended for normal operation, any safety equipment intended for use in case of design basis accidents, or special equipment intended to mitigate the consequences of the beyond design basis accidents. All these activities and special equipment comprise the before-mentioned fourth level of the in-depth defense.

Level 4 – Beyond design basis accident management:

– prevention of development of beyond design basis accidents and mitigation of their consequences;

– containment protection against destruction in case of a beyond design basis accident and keeping it in operable condition;

– NPP controllability restoration to stop the fission chain reaction, provide stable cooling of the nuclear fuel and keep the radioactive substances within the specified boundaries.

The beyond design basis accident management concept is based on the requirement to limit the radiation exposure resulting from such accidents by way of measures intended to manage the accidents and by way of implementation of the plans of activities, at the NPP site and in the surrounding territory, for the personnel and population protection. These measures are the integral parts of the in-depth defense.

For a part of the beyond design basis accidents considered within the scope of the design, the degree of limitation of the radiation exposure is defined by the radiation safety criterion specified in the document, *Placement of nuclear power plants. The main criteria and requirements for safety* (NP-032-01), stipulating the requirements for NPP location. This criterion sets limits for the so-called limiting accidental emission in case of a beyond design basis accident. The predictable population exposure doses at the boundary of the zone, for which the protective measures are planned, as well as beyond this zone, must not exceed the values, stipulated in the valid radiation safety regulations as the limiting values for making the decision to apply the measures for the population protection in case of a radiation accident involving the territory pollution; and the zone for which the mandatory population evacuation activities are planned must cover the area where, in case of a beyond design basis accident involving the maximum acceptable emission of radioactive substances into the environment, the maximum level of the dose criterion for the mandatory evacuation of the critical population group at the initial stage of the radiation accident, stipulated in the valid radiation safety regulations, can be reached or exceeded.

In accordance with Paragraph 1.2.17, OPB-88/97, the probability of the limiting accidental emission must be below 10^{-7} for a reactor per year. This is necessary to avoid evacuation and other population protection activities beyond the before-mentioned zone for which the accident response measures are planned. If this requirement is not met, the additional technical measures must be taken for the beyond design basis accident management to mitigate its consequences.

This requirement stipulated in Paragraph 1.2.17 is one of the principles specified in OPB-88/97 for exclusion of the beyond design basis accidents from consideration. This is a probabilistic principle applied to the preparation of additional technical measures for the beyond design basis accident management. No additional technical measures must be prepared for the beyond design basis accidents for which the requirement stipulated in Paragraph 1.2.17 is met.

The second exclusion principle is deterministic. It is stipulated in Paragraph 1.2.14, OPB-88/97, and specifies the condition that, if met, means that no measures at all may be provided, both technical and organizational, for the beyond design basis accident management. This condition is the beyond design basis accident impossibility based on the reactor's intrinsic self-protection properties and operation principles.

Thus, if the reactor's intrinsic self-protection properties and design principles do not make the beyond design basis accident impossible, such an accident must be considered in the NPP project for the purpose of preparation of measures for the beyond design basis accident management, irrespective of its probability. However, only organizational measures may be prepared if the first (probabilistic) exclusion principle is met.

Two more probabilistic principles are stipulated in OPB-88/97. One principle (Paragraph 4.2.2) stipulates that the estimated probability of heavy damage or core meltdown must not exceed 10^{-5} for a reactor per year. Another principle (Paragraph 1.2.12, footnote) specifies how the single failure principle shall be applied. This criterion is an addition to the deterministic principle; it means that the ruptures of vessels and equipment casings may be excluded from consideration in the project if they are made and operated in accordance with the highest requirements stipulated in the applicable rules and regulations. For this purpose, the reactor vessel destruction probability must be proven to be not higher than 10^{-7} for a reactor per year.

While the probabilistic safety criteria used in OPB-88/97 are evaluative, the probabilistic safety analysis must be carried out in accordance with these criteria.

The general description of additional technical means for the beyond design basis accident management are given in OPB-88/97. They are described in more details in NP-010-98, *Regulations for design and operation of localizing safety systems*. In this document, Paragraph 2.1.8 stipulates that, for the purpose of localization of the beyond design basis accidents taken into consideration, as a rule, the NPP design shall provide technical means that shall prevent the containment and its reinforced-concrete structures from damage when the pressure or temperature exceed the design values, keep the molten fuel within the accident localization zone (and ensure that the molten fuel remains subcritical), prevent hydrogen explosions and limit the radioactive product emissions into the environment.

In accordance with OPB-88/97, Paragraph 4.2.4, if secondary critical masses can result from core destruction or fuel meltdown in case of a beyond design basis accident, technical measures must be provided to ensure that the radiation exposure not exceeds the maximum predictable doses for population stipulated in Paragraph 1.2.17 for the beyond design basis accidents.

Along with technical requirements described above, Russian rules and regulations stipulate the requirements for organizational measures intended to provide the beyond design basis accident management.

For example, the requirements are stipulated that describe the procedures for preparation and for bringing into force the special instructions prescribing the personnel activities for safety protection in case of a beyond design basis accident. In accordance with OPB-88/97, Paragraph 1.2.16, these instructions shall be based on the analysis of beyond design basis accidents.

To prepare the beyond design basis accident management instructions, the limits, in terms of time and parameters, shall be determined for the considered accidental conditions that can arise while the beyond design basis accident evolution is in progress; the procedures shall be prepared describing how to provide the operators with information necessary for their activities; the priorities shall be specified, in terms of diagnostics and functions, for each level of severity; and the set of necessary general functional instructions shall be prepared, i.e. the instructions describing how to activate the specific safety functions. For these purposes, analytical calculations shall be carried out for several typical scenarios of the beyond design basis accidents resulting in various accidental conditions.

In accordance with this task, the list of considered accidental conditions shall be prepared, and it shall be used as the basis for preparation of the list of accidental scenarios subject to further analytical calculations. As a result of these calculations, the efficiency of implementation of relevant critical safety functions, possible consequences in case of failure to implement these functions and other before-mentioned characteristics in terms of time and parameters shall be assessed.

Technical protection means provided within the scope of the beyond design basis accident management

See Table 1 for the most significant protection functions and relevant supplementary technical means for the BDBA management, and the management procedures.

Table 1 – Technical protection means

Protection function	System implementing the protection function	System management procedure
1 Reactor switching to the subcritical condition	Emergency boration system (ATWS mode)	Automatic
2 Heat removal and reactor unit cooling by steam generators	System for passive heat removal by steam generators (PHR SG system)	Automatic
3 Heat removal from the containment	System for passive heat removal from the containment (PHRC system)	Automatic
4 Pressure reduction in the primary circuit, to prevent the scenario when the melt propagates from the reactor vessel under high pressure	PHR SG system	Automatic
5 Reduction of gas aerosol release through the leakages in the double containment	Emergency ventilation system for vacuum keeping and environment cleaning in the space between the shells	Automatic
6 Hydrogen suppression within the containment	System for hydrogen removal from the containment	Automatic
7 Volatile iodine localization in the containment	Volatile iodine chemical bounding system	By an operator
8 Molten fuel localization in case of an accident involving the reactor vessel destruction	Core melt keeping system (the trap)	Automatic
9 Core melt cooling in the trap	System for emergency water consumption from the vessel internals audit vault	By an operator

For the purpose of the system protection against common cause failures and to improve the NPP safety parameters as a whole, the project provides the functional redundancy for the systems implementing the primary safety functions as described in Table 2.

Table 2 – Functional redundancy modes

Safety function	Primary safety system	Functional redundancy modes
Reactivity control	Emergency reactor protection system	1. Emergency boration system 2. Makeup and boron control system

Safety function	Primary safety system	Functional redundancy modes
Residual heat removal from the reactor unit	Emergency cool-down system (EFP + BRU-A)	1. System for passive heat removal by steam generators (PHR SG) 2. Normal heat removal system (BRU-C + auxiliary feed pump) 3. Cooling-down through the first circuit (residual heat removal system)
Keeping the sufficient coolant amount in the reactor to cool the core	Low-pressure emergency injection system	High-pressure emergency injection system
	High-pressure emergency injection system	Makeup and boron control system
Radioactive product localization (as for the heat removal from the containment)	Sprinkler system	System for passive heat removal from the containment (PHRC system)
Controlling and supporting safety functions	Controlling safety system, subchannel A*	Controlling safety system, subchannel B*
	Emergency diesel generators	Block diesel generator

* In each of four controlling safety system's channels, different hardware platforms are used to implement the subchannels A and B (i.e. different microprocessor equipment and hardwired logic).

To provide the NPP safety level in accordance with the applicable regulations as well as in accordance with the Belarusian NPP Requirements Specification, the project provides for the set of safety systems and supplementary technical means for the BDBA management. For the description of these systems and their structure, see Table 3. For the simplified layout of these safety systems and technical means, see Figure 1.

Table 3 – Set of safety systems and supplementary technical means for the BDBA management

Description	**Number of channels and effectiveness**
Protecting, localizing, supporting and controlling safety systems	
1. High-pressure emergency injection system	4 × 100 %
2. Low-pressure emergency injection system	4 × 100 %
3. Emergency boration system	4 × 50 %
4. System for emergency feed water supply and heat removal through the BRU-A	4 × 100 %
5. Sprinkler system	4 × 50 %
6. Residual heat removal system	4 × 50 %
7. Intermediate cooling circuit system for essential services	4 × 100 %
8. Cooling water supply for essential services	4 × 100 %
9. Ventilation system for the safety system premises	4 × 100 %

Description	Number of channels and effectiveness
10. Containment localizing reinforcement system	2 × 100 %
11. Borated water storage system	2 × 100 %
12. Emergency gas removal system	2 × 100 %
13. First circuit overpressure protection systems	2 × 100 %
14. Second circuit overpressure protection system	2 × 100 %
15. Main steam pipelines cutoff system (quick-response isolation cutoff valve)	2 × 100 %
16. Emergency diesel generator power supply system	4 × 100 %
17. Safety activation system	4 sensors for each parameter, 4 logical channels in each, with 2/4 logic
18. Emergency reactor shutdown system	4 sensors for each parameter, 4 sets of 2/4 logic at the 1st voting level, and 2 sets of 2/4 logic at the 2nd voting level
Passive safety system	
19. ECCS accumulator tank system	4 × 50 %
20. Reactor compartment's sealed enclosure system	1 × 100 %
21. Containment hydrogen removal system (1 subsystem)	1 × 100 %
Supplementary technical means for the BDBA management	
22. System for passive heat removal by steam generators (PHR SG system)	4 × 33 %
23. System for passive heat removal from the containment (PHRC system)	4 × 33 %
24. Melt localization system	1 × 100 %
25. Containment hydrogen removal system	1 × 100 %
26. Volatile iodine chemical bounding system	1 × 100 %
27. Ventilation system for vacuum keeping in the space between the shells	2 × 100 %
28. System for emergency water consumption from the vessel internals audit vault	2 × 100 %

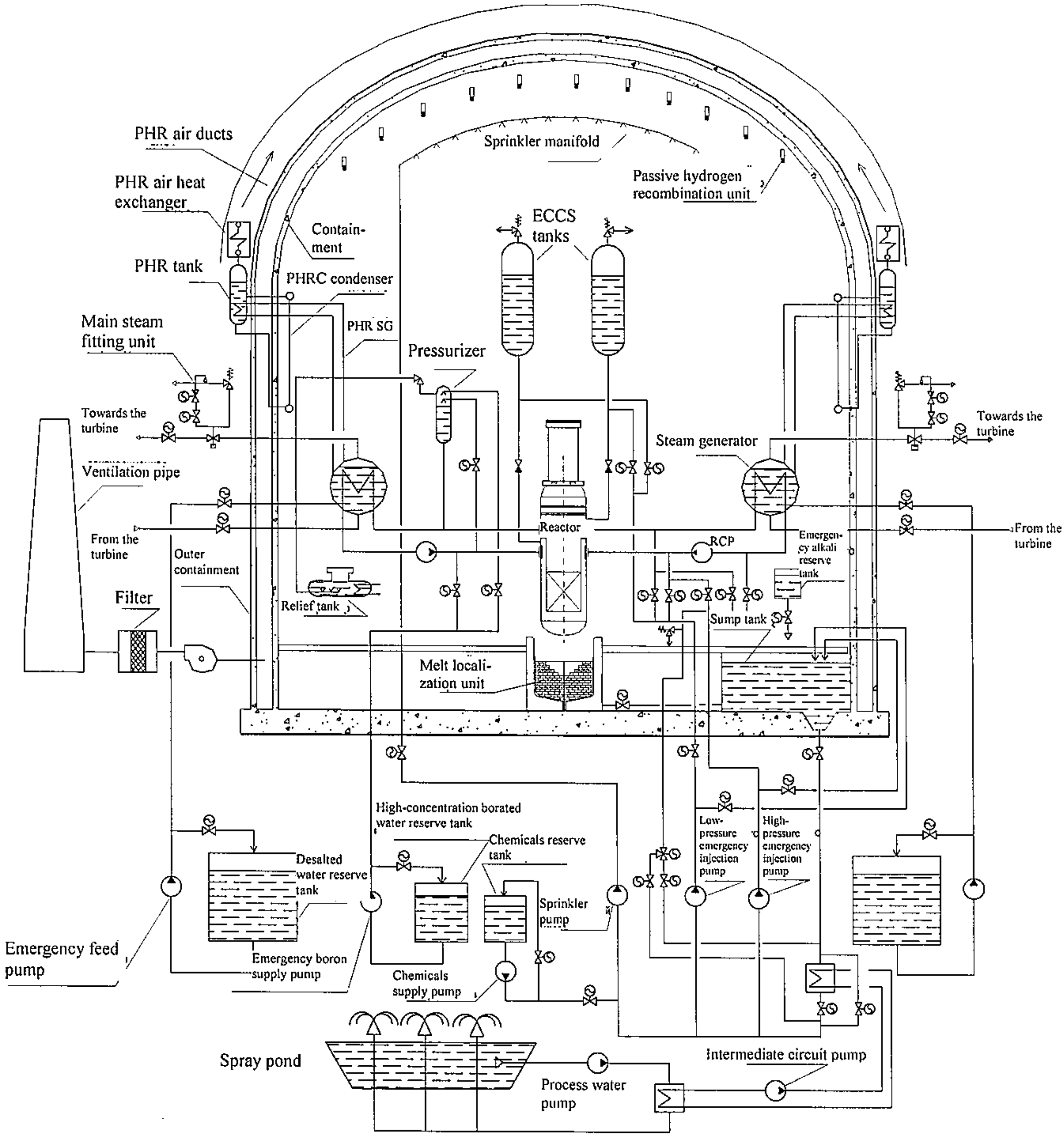


Figure 1 – Safety systems and technical means for the BDBA management

Beyond design basis accidents: preliminary list

The preliminary list of beyond design basis accidents has been prepared for the Belarusian NPP (see Table 4). The recommended list of initiating events (NP-006-98, Annex 15-1), the list of design basis accidents (the Requirements Specification for the technical design of the reactor unit, Annex D) and the preliminary results of the probabilistic safety analysis (PSA) for the NPP with VVER-1200 were reviewed, taking into consideration the safety system failures beyond the single failure (in addition to the design basis accidents) and the initiating events not considered for the design basis accidents. These data were used as a basis for the solutions with regard to the special technical means for the BDBA management.

Table 4 – Preliminary list of beyond design basis accidents

Beyond design basis accident description
1. Failure of all AC electric power sources for 8 hours and for 24 hours
2. Complete interruption of the spent fuel pool cooling for 8 hours and for 24 hours
3. Range of steam pipelines' breaks within and outside the containment, including the maximum steam pipeline diameter, with one pipe broken in the steam generator
4. Complete interruption of feed water supply
5. Coolant loss accidents with large leaks, involving the failure of the active part of the low-pressure ECCS
6. Coolant loss accidents with small leaks, involving the failure of the active part of the high-pressure ECCS
7. Long (up to 24 hours) interruption of heat removal in the normal cool-down and emergency cool-down systems, with the reactor cover removed and/or with the reactor sealed.
8. Coolant leakage from the first circuit to the second one in case of multiple breaks of steam generator pipes, or the leak along the manifold in the first circuit of the steam generator (with the effective nominal diameter 100 mm)
9. ATWS accidents in cases as follows: - interruption of non-emergency AC power supply for the auxiliary fixed equipment (NPP blackout); - faulty closing of the quick-response isolation cutoff valve; - uncontrollable removal of one control or a group of controls during operation at the minimum controlled reactor power or during power operation; - unintentional dilution of boric acid in the coolant for the first circuit; - unintentional opening of the safety valve in the steam generator, the relief valve (BRU-A) or the turbine bypass valve (BRU-C), with the subsequent failure to fit these valves; - loss of normal feed water flow rate (except to the feed water pipeline break).
10. Accidents in the nuclear fuel handling system: - failure of all AC electric power sources for 24 hours, with the emergency reactor protection system failing to operate; - self-sustaining chain reaction arising for the nuclear fuel storage and handling systems; - complete dewatering of the spent fuel pool; - process equipment or building structure fall on the ceiling slab of the nuclear fuel storage compartment or on the nuclear fuel in the storage; - class 1 storage flooding with water.

Beyond design basis accident management concept

The safety concept for the Belarusian NPP reactor provides for measures intended to manage the beyond design basis accidents, to prevent these accidents from the development resulting in heavy accidents and to mitigate the consequences of heavy accidents.

The primary goals of the accident management are as follows:

- prevent the core damage;
- prevent the reactor vessel melt-through;
- prevent the containment failure;
- reduce the radioactive emissions into the environment.

The NPP personnel is capable to provide the accident management even if the safety elements and systems partially fail. The backgrounds for this capability are as follows:

- inertial nature of accident processes and their self-limitation due to the reactor's self-protection properties, characteristics of active and passive safety systems, and design margins;
- overlapping functions provided by the safety systems and the BDBA management technical means;
- making use of capabilities provided by the systems intended for normal operation;
- in-depth protective barriers for the radioactivity release designed with due consideration of conditions common for beyond design basis accidents;
- safety systems designed with due consideration of conditions existing for beyond design basis accidents;
- making use of auxiliary measures, such as cabling and hydraulic jumpers, adapters, portable triggering devices etc.

The first level in the BDBA management strategy implementation consists of safety systems intended, in terms of their functions, for safety purposes (reactor subcritical condition, fuel cooling, radioactive product localization). For the description of these systems, their functions, structure and redundancy, see Tables 1, 2, 3, 5.

The second level in the BDBA management strategy implementation consists of supplementary technical means for the BDBA management (see Tables 1, 3, 5).

The normal operation systems are also applicable for the BDBA management:
- normal reactor shutdown systems;
- systems and equipment used to deliver water into the steam generators and to release steam from these generators;
- systems and equipment used to deliver water into the reactor, including those applicable to deliver liquid absorbers into the reactor;
- systems and equipment used to remove heat from the reactor unit components, such as the intermediate circuit systems and cooling water supply systems.

To reach the goals listed above, the primary tasks as follows must be implemented. Their implementation is possible during various BDBA stages, and various technical means are applicable for these purposes (see Table 5).

Table 5 – BDBA management

BDBA management task	Technical means
1. Core damage prevention	Normal operation and safety systems for heat removal through the second circuit, PHR SG system, ECCS accumulator tank system, high-pressure and low-pressure emergency injection systems, makeup and boron control system, emergency gas removal system.
2. Reactor vessel melt-through prevention	High-pressure and low-pressure emergency injection systems, makeup and boron control system, emergency gas removal system.
3. Containment failure prevention	Emergency gas removal system, sprinkler system, PHR SG system, PHRC system, hydrogen suppression system, melt localization trap.
4. Reduction of radioactive emissions into the environment	Sprinkler system, volatile iodine suppression system, inter-shell space ventilation and cleaning system, melt localization trap.

See below for the brief explanation of Table 5.

The core damage and meltdown can be prevented by restoration of residual heat removal. To remove residual heat, the second circuit can be used, or cooling water can be fed into the first circuit from active systems.

If the second loop can be used, the time available for the operator to prevent the core destruction depends on the time before the circulation failure in the loops of the first circuit. For example, if the reactor block is de-energized, and no feed water is delivered to the steam generators, the time before the complete circulation loss can be one hour or longer. This time is available for the operator to restore the heat removal function through the second circuit. Water supply from one EFP after some time (about two hours) will be the sufficient measure to prevent the core damage. The second system that can provide heat removal from the steam generator is the PHR SG system. If the second circuit cannot be used to remove residual heat release, the operator can use the water makeup and release for the first circuit and release the first circuit coolant to the containment. For this purpose, water supply is available from any operable pump (in the makeup and boron control system, high-pressure or low-pressure emergency injection systems). To reduce pressure in the first circuit, the provisions are made to release the coolant through the emergency gas removal system.

<u>Reactor vessel melt-through prevention</u>

Excessive heating and core position variations in the reactor vessel are the primary causes of the physical phenomena endangering the reactor integrity. Thus, the operating personnel must, in accordance with the emergency instructions, proceed with the activities to restore the operability of systems that can supply water to the first circuit.

Prompt personnel's activities for water delivery from active ECCS subsystems can restore the fuel cooling and prevent the reactor vessel damage. Because ECCS water contains the appropriate concentration of boric acid, the core or corium remain subcritical during various stages of their cooling.

If the taken measures prove to be ineffective, and the reactor vessel destruction by melt becomes inevitable, the operator must make efforts to reduce pressure in the reactor in order to ensure that the reactor vessel melt-through takes place under minimum pressure. For this purpose, the emergency gas removal system is provided in the design.

<u>Early containment damage prevention</u>

Design measures intended to prevent the containment from early destruction are as follows:

- no environments or things affect the containment walls directly;
- the corridors between the containment rooms mitigate possible impact loads on the containment walls;
- the melt localization unit and large amount of water in the containment, provided in accordance with the design, prevent quick heating and accumulate the molten core energy;
- the emergency gas removal system prevents loads resulting from high-pressure melt release from the reactor;
- the hydrogen control and removal system prevents accumulation of dangerous hydrogen concentrations.

<u>Containment damage prevention during late accident stages</u>

The primary technical means intended to prevent the containment from late destruction are as follows:

- the active system for long-term heat removal from the containment (the sprinkler system);
- the passive system for long-term heat removal from the containment (the PHRC system);
- the melt localization system;
- the hydrogen removal system having the capacity sufficient to reduce the hydrogen concentration to the safe level during late stages of an accident.

It should be noted that, in case of a heavy accident involving the NPP blackout (including the failure of emergency diesel generators and reactor block diesel generators) the active sprinkler system becomes inoperable. In such a case, the passive heat removal system removes heat from the containment. The PHRC system design is appropriate to remove heat from the containment to the final absorber, the ambient air, for a long time.

<u>Reduction of radioactive emissions into the environment</u>

The most effective way to keep the fission products in case of a heavy accident is to preserve or restore the tightness of the first circuit or to provide the containment tightness. If the first circuit's tightness cannot be preserved or restored, one more barrier against radioactive emissions from the fuel is water; however, this is the case if the water supply for fuel cooling is sufficient. The water keeps the significant part of fission products. They predominantly remain in the reactor vessel if its tightness is not damaged. However, the fission products are partially released as radioactive gases and aerosols; other part of these products propagates with water into the containment.

The design includes effective systems for localization of radioactive substances within the NPP if the fuel is damaged beyond the operational limit and/or if the coolant circuit tightness is breached:

- double protective shell, with the inner shell's design leakage not exceeding 0.2 % of the volume per 24 hours under the maximum design excessive pressure;
- the sprinkler system for pressure reduction under the shell and for removal of fission products from the atmosphere of the rooms in the tight shell;
- the system for volatile iodine suppression at various accident stages;
- the emergency ventilation system provided with aerosol and iodine filters. The system keeps vacuum in the space between the shells;
- the melt localization unit keeping the core melt if it penetrates beyond the reactor vessel.

Instrumentation for BDBA control and management

The Belarusian NPP is provided with appropriate instrumentation to assess correctly the plant condition and the accident risk level. The measurement channels used to manage heavy accidents are designed for heavy accident conditions and intended to carry out the measurements in the relevant range of parameters. For example, temperature sensors in the reactor's concrete vault provide information about the corium condition beyond the reactor vessel.

The primary instrumentation for heavy accident management includes the detectors, communication lines and equipment used to read and display the parameters and conditions as follows:

- neutron flux;
- temperature (core outlet, first and second circuit, containment);
- water level (first and second circuit, containment);
- pressure (first and second circuit, containment);
- radioactivity (second circuit, containment);
- corium condition within and outside the reactor vessel (temperature, location, criticality);
- reactor vessel temperature;
- temperature in the reactor's concrete vault;
- atmosphere composition in the containment (e.g. hydrogen concentration);
- safety systems condition.

The detectors, communication lines and secondary equipment are made completely independent and can operate during 72 hours in case of the complete power failure. For this purpose, the special power supply channel is provided in the design; it includes the single-channel AC power supply system powered by the mobile diesel generator, 20 kW, 400/230 V, and the single-channel DC power supply system (220 V) powered by the accumulator batteries having the capacity sufficient for continuous operation during 24 hours or longer.

Beyond Design Basis Accident Management Manual

The Belarusian NPP administration prepares and issues the *Beyond Design Basis Accident Management Manual* in accordance with the NPP process description and the NPP Safety Analysis Report and completely in line with the design documents.

The *Beyond Design Basis Accident Management Manual* stipulates the personnel activities for the BDBA management, the purpose of which is to prevent the destruction of physical barriers on the way of propagation of fission products, and to mitigate the BDBA consequences. The *Manual* shall be used at the reactor block control board and at the standby control board.

With the great number of possible scenarios of beyond design basis accidents, the event-driven approach is inapplicable to the BDBA management arrangement. The total range of beyond design basis accidents cannot be covered in principle, taking into consideration possible combinations of several failures and the accident diagnostics difficulties. The event-driven approach is applicable to simple, easily recognizable accidents within the design basis.

In accordance with the recommendations given by the specialists of the Scientific and Engineering Center for Nuclear and Radiation Safety, as well as those listed in IAEA documents, the symptom-oriented approach shall be applied for the *BDBA Management Manual* preparation. The advantage of this approach for such modes of operation is that the personnel's activities are oriented towards protection of physical barriers, irrespective of particular events and failures, but in accordance with the real condition of the reactor unit and equipment in terms of the most significant criteria.

The paramount goal of the beyond design basis accident management is the prevention of uncontrollable radioactive product propagation beyond the boundaries specified in the design.

The Belarusian NPP does not pose any danger for population while the protective barriers as follows remain unbroken:

- fuel matrix;
- fuel elements' shells;
- reactor coolant circuit boundaries;
- reactor unit's containment.

The conditions as follows are necessary to keep the protective barriers' integrity:

- quick reactor shutdown and keeping the core in subcritical condition;
- heat removal from the core during an accident and upon the stabilization of parameters in the post-accident condition;
- heat removal from the reactor unit;
- first circuit protection against pressure rise, hydraulic shocks, thermal loads;
- accident consequences localization by sealing the reactor unit shell for minimization of radiological consequences and for keeping the radioactive products within the specified boundaries and amounts.

Within the scope of the symptom-oriented approach to the beyond design basis accident management, critical functions as follows are controlled and restored to meet the above-listed conditions:

- reactor core subcritical condition;
- reactor core cooling;
- heat removal from the first circuit to the second circuit;
- first circuit integrity;
- containment integrity.

If the situation arose breaching the critical safety function (CSF), activities must be implemented immediately to restore this CSF before breaching the integrity of the relevant barrier.

Equipment failures or errors in personnel activities re initiating event increase the core damage risk. Therefore, operability of the equipment intended to restore the CSF can be considered at the *Equipment Operability* CSF, having the priority status number zero.

The set of activities intended to control and restore the CSF and the personnel's activities intended to restore the failed equipment operability are the additional and last defense line against the potential emission of radioactive materials into the environment.

The critical safety functions as follows are provided:

– *Equipment Operability* CSF (F-0). This function is breached if the auxiliary power supply of the reactor block is completely lost, if the reactor block control from the BCB is lost, or if the equipment is flooded at below-zero levels in the reactor compartment;

– *Reactor Core Subcritical Condition* CSF (F-1). The accidents as follows can involve the *Reactor Core Subcritical Condition* CSF breach: range of steam pipeline breaks in the containment; long (up to 24 hours) interruption of heat removal in the normal cool-down and emergency cool-down systems with the reactor cover removed and/or with the reactor sealed; accidents with the emergency protection system failing to operate;

– *Reactor Core Cooling* CSF (F-2). The accidents as follows can involve the *Reactor Core Cooling* CSF breach: long (up to 24 hours) interruption of heat removal in the normal cool-down and emergency cool-down systems with the reactor cover removed and/or with the reactor sealed; coolant leakage from the first circuit to the second one in case of multiple breaks of steam generator pipes, or the leak along the manifold in the first circuit of the steam generator (with the effective nominal diameter 100 mm); coolant loss accidents with small leaks, involving the failure of the active part of the high-pressure ECCS; coolant loss accidents with small leaks, involving the failure of the active part of the low-pressure ECCS; accidents with the emergency protection system failing to operate;

– *Heat Removal from the First Circuit to the Second Circuit* CSF (F-3). The accidents as follows can involve the *Heat Removal from the First Circuit to the Second Circuit* CSF breach: loss of normal feed water flow rate, with the emergency protection system failing to operate;

– *First Circuit Integrity* CSF (F-4). The accidents as follows can involve the *First Circuit Integrity* CSF breach: complete interruption of feed water supply; long (up to 24 hours) interruption of heat removal in the normal cool-down and emergency cool-down systems with the reactor sealed; unintentional opening of the pilot-operated relief valve in the steam generator, the BRU-A or the BRU-C, with the subsequent failure to fit these valves;

– *Containment Integrity* CSF (F-5). The accidents as follows can involve the *Containment Integrity* CSF breach: range of steam pipeline breaks in the containment; coolant loss accidents with large leaks involving the failure of the active part of the low-pressure ECCS; long (up to 24 hours) interruption of heat removal in the normal cool-down and emergency cool-down systems with the reactor cover removed and/or with the reactor sealed;

– *Fullness – First Circuit Coolant Amount* CSF (F-6). The accidents as follows can involve the *Fullness – First Circuit Coolant Amount* CSF breach: coolant leakage from the first circuit to the second one in case of multiple breaks of steam generator pipes, or the leak along the manifold in the first circuit of the steam generator (with the effective nominal diameter 100 mm); unintentional opening of the pilot-operated relief valve in the steam generator, the BRU-A or the BRU-C, with the subsequent failure to fit these valves; range of steam pipeline breaks outside the containment.

CSF condition monitoring procedure:

The personnel activities that shall be carried out to restore all CSFs (from F-0 to F-6) are described in the *Beyond Design Basis Accident Management Manual.* The CSF condition must be monitored continuously, starting from the time when the accident arose. To assess the current CSF condition, parameters must be monitored and the CSF state trees must be analyzed in accordance with their priorities.

The CSF condition must be monitored by the NPP shift foreman or by the reactor control leading engineer.

The CSF condition information shall be recorded in the *CSF State Tree Control Sheet* (as the word "normal" or the relevant CSF instruction code) and delivered to the person in charge for the accident elimination.

CSF State Tree Control Sheet blanks shall be kept at the reactor block control board.

In accordance with the available CSF condition information, the NPP shift foreman shall make a decision to terminate the operating personnel's controlling activities stipulated by the *Emergency Operating Instructions* and to initiate the activities in accordance with the relevant section of the *Beyond Design Basis Accident Management Manual*, or to terminate the activities in accordance with one section of the *Beyond Design Basis Accident Management Manual* and to initiate the activities in accordance with other section of this *Manual*, or to proceed with the activities in accordance with the *Emergency Operating Instructions* after the CSF restoration.

Design basis accidents:

1. Range of steam pipelines' breaks within and outside the containment.
2. Steam generator feed water pipeline break.
3. Unintentional opening of the pressurizer safety valve, with the subsequent failure to fit this valve.
4. Small coolant leaks resulting from a pipeline break in the first circuit (with the effective diameter less than 100 mm).
5. Steam generator heat exchange pipe break followed by 60°C/hour cooling-down.
6. Compensable leak within the containment.

7. Accidents involving the coolant loss from the reactor during a shutdown at the unsealed reactor or during the refueling procedure.
8. Leak in the spent fuel pool or pipeline break resulting in the water level drop in the pool.
9. Shaft jamming or break in one main circulation unit.
10. Ejection of controls from the control and protection system resulting from the drive housing rupture.
11. Connection of a loop not being in operation, without the previous power reduction.
12. Large coolant leaks resulting from breaks of pipelines in the first circuit (with the effective diameter exceeding 100 mm), including the main circulation pipeline break.
13. Leak from the first circuit to the second one resulting from the steam generator manifold cover breakaway.
14. Package fastening damage during transportation of the nuclear fuel.
15. Fall of a transportation container with spent fuel assemblies.
16. Spent fuel assembly sticking during the refueling procedure.
17. Equipment failures in the nuclear fuel storage and handling systems, including complete power interruption.
18. Reduction of the homogeneous absorbent concentration in the spent fuel pool.

ABBREVIATIONS

ATWS		Anticipated transient without scram
АЗ		Emergency protection
АПЭН	EFP	Emergency feed water pump
АЭС	NPP	Nuclear power plant
БВ		Spent fuel pool
БЗОК	QICV	Quick-response isolation cutoff valve
БПУ	BCB	Reactor block control board
БРУ-А	BRU-A	Quick-response reduction unit for steam release into the atmosphere
БРУ-К	BRU-C	Quick-response reduction unit for steam release into the turbine condenser
ВВЭР	VVER	Water-water power reactor (series of pressurized water reactors)
ВИУР		Reactor control leading engineer
ВКУ		Vessel internals
ВПЭН		Auxiliary feed pump
ГО		Containment
ЗПА	BDBA	Beyond design basis accident
ИЛА		Emergency operating instructions
КИП		Instrumentation
КФБ	CSF	Critical safety functions
ПГ		Steam generator
РПУ	SCB	Standby control board

РУ		Reactor unit
РУЗА		Beyond design basis accident management manual
САОЗ	ECCS	Emergency core cool-down system
СБ		Safety system
СПОТ ЗО	PHRC system	System for passive heat removal from the containment
СПОТ ПГ	PHR SG system	System for passive heat removal by steam generators
УЛР	MLU	Melt localization unit
УСБ		Controlling safety system
ЯТ		Nuclear fuel

Question: Probabilistic safety analysis (PSA). Probabilistic analysis methods: advantages and disadvantages. PSA analysis for the Belarusian NPP.

Answer: The advantage of the probabilistic safety analysis (PSA) is that it is a powerful tool for recognition of potentially vulnerable aspects in the NPP blocks' structure and control. One of the aspects in PSA application is the project implementation process providing the opportunities to improve understanding of the block structure features, complex systems' functions and personnel activities. For example, the results of the system reliability analysis carried out within the scope of this PSA have demonstrated that the no-failure characteristics of the system for passive heat removal through the steam generators can be improved if the bypass lines are mounted for the emergency heat removal tank filling and making-up and if the valve test period is reduced in the line for the emergency heat removal tank filling.

The purpose of the first-level probabilistic safety analysis (PSA-1) is to calculate the estimated frequencies of accident sequences resulting in heavy fuel damage in the core and in the spent fuel pool, and to estimate the total frequency of heavy fuel damage.

As for the disadvantages of the PSA development, it should be noted that the probabilistic estimates before and after the personnel's emergency activities cannot be calculated, because the operational instructions have not been prepared, and the estimates available for the reference NPP (Taiwan NPP) are used for this purpose.

PSA results:

The probabilistic safety characteristics have been calculated for the NPP:

- the maximum total frequency of heavy core damage, calculated for all accident sequences, is 10^{-6} for a reactor per year;
- to prevent the necessity to evacuate population in the areas beyond the protective activity planning zone specified in accordance with the regulatory requirements for the NPP location, efforts must be taken to ensure that the estimated maximum probability of the limiting accidental emission specified in accordance with these requirements is 10^{-7} for a reactor per year (PSA-2 task).

As a result of the final classification, when the reactor block is in the power operation mode, the estimated average total frequency of the core damage is $3.82 \cdot 10^{-7}$ for a reactor per year.

The process equipment systems making the most significant contributions into the core damage frequency are as follows:

- intermediate circuit system for essential services;
- makeup systems for the first circuit;
- emergency heat removal systems;
- systems involved into the Feed&Bleed procedure.

The analysis of the most significant contributors into the core damage frequency for the Belarusian NPP demonstrates that the estimated core damage frequency resulting from the quantification can be reduced by way of the implementation of several technical solutions and operational procedures. See below for the recommendations prepared on the basis of the PSA analysis results.

Technical solutions:

- implement the detailed procedure for calculation of the estimated reactor vessel destruction frequency.

Operational procedure improvement:

- the activities of personnel during the accident onset must be based on the requirements stipulated in the system operation instructions.

More studies:

- analyze the general mode failures in the intermediate circuit system for essential services with the half of system's channels being in the normal operation mode;
- study the failures in the supporting systems with the equipment characteristics and temperature exposure parameters specified more exactly.

The logical model of the reactor unit shall be improved: modeling of the supporting systems shall not be excessively conservative.

For the low power and shutdown modes, the average total frequency of the core damage is $3.7 \cdot 10^{-7}$ per year.

The most significant contributors into the core damage frequency are as follows:

- personnel errors in the initiating event responses;
- the system for residual heat removal from the first circuit;
- low-pressure ECCS.

The analysis of the most significant contributors into the core damage frequency for the Belarusian NPP in the low power and shutdown modes demonstrates that the estimated core damage frequency resulting from the quantification can be reduced by way of the implementation of several technical solutions and operational procedures. See below for the recommendations prepared on the basis of the PSA analysis results.

Technical solutions:

- analyze how the sprinkler system can be used for the first circuit makeup in standby modes.

Operational procedure improvement:

- the activities of personnel during the accident onset must be based on the requirements stipulated in the system operation instructions.

Question: Passive safety systems. Drawings, operation characteristics, operation principles, reference.

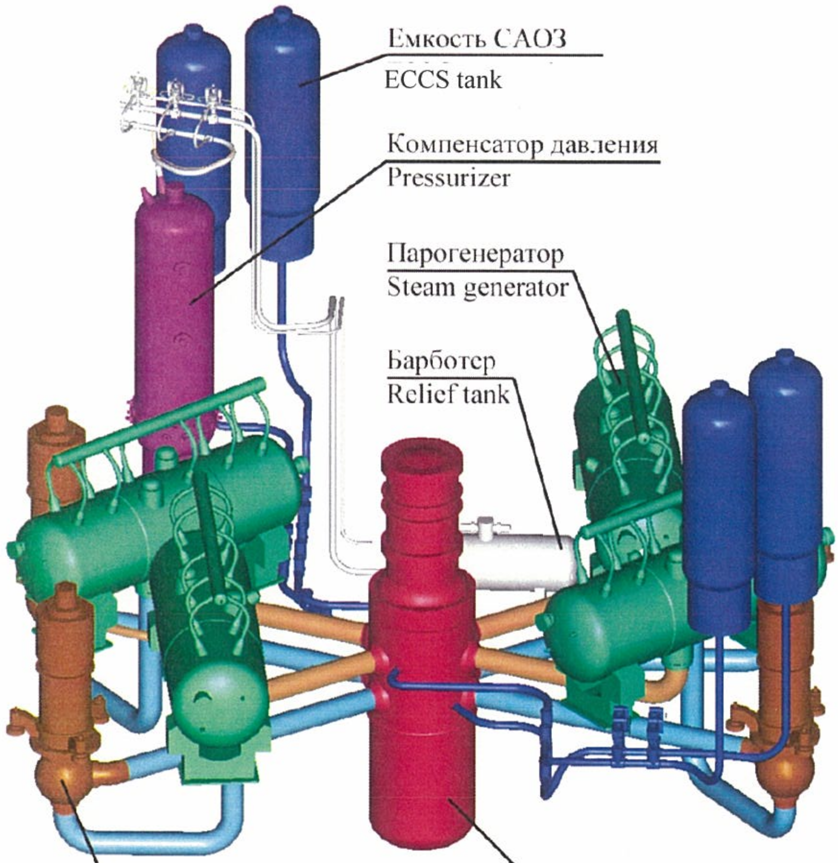
Answer: In the Belarusian NPP design, the accident management measures based on the passive operation principle are as follows:

1. Passive core cooling system;
2. Containment and steam generator passive heat removal system;
3. Hydrogen explosion safety system;
4. Melt localization unit.

<u>Passive core cooling system</u>

The system shall be used to feed the boric acid solution, with the feed rate sufficient to cool the reactor core until the pumps in the low-pressure emergency injection system are activated, in case of a design basis accident involving the coolant loss, with the boric acid concentration not less than 16 g/kg and the temperature not less than 20°C, when the pressure in the first circuit is less than 5.9 MPa.

The ECCS passive part consists of four identical channels completely independent from each other. Each channel in the ECCS passive part includes the ECCS tank (60 m^3), fittings and pipelines. Two channels are connected with the reactor collection chamber; two other channels are connected with the reactor pressure chamber.

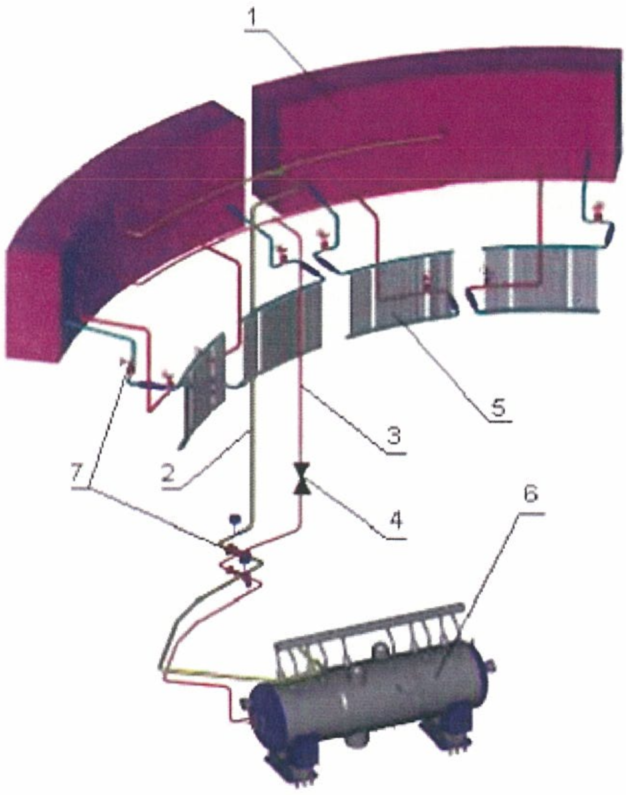


Containment and steam generator passive heat removal system

The system for passive heat removal from the containment is a technical facility intended to control the beyond design basis accidents. It is capable to remove heat from the containment during a long time (the system can operate in a standalone mode for 24 hours or longer) in case of a beyond design basis accident involving the complete electric power loss (the blackout).

The system is capable to reduce and to keep within the design limits the pressure in the containment and to remove the final heat absorber propagating into the space in the containment in case of beyond design basis accidents including the accidents involving heavy core damage.

The system for passive heat removal through the steam generators is capable to remove, during a long time, the residual core heat to the final heat absorber through the second circuit in case of a beyond design basis accident. The system serves as a backup for the active system for heat removal to the final heat absorber if the latter fails to provide its design functions.



1 – Emergency heat removal tanks
2 – Steam pipelines
3 – Condensate pipelines
4 – PHR SG system valves
5 – PHRC system heat exchangers
6 – Steam generator
7 – Cutoff fittings

Hydrogen explosion safety system

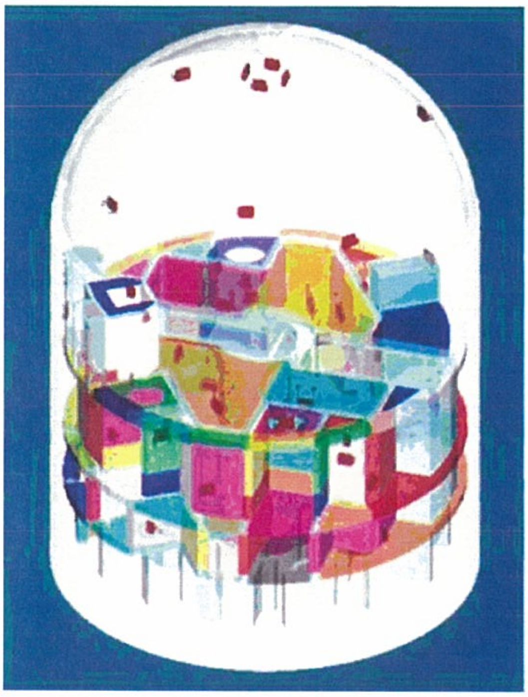
The system provides the functions as follows:

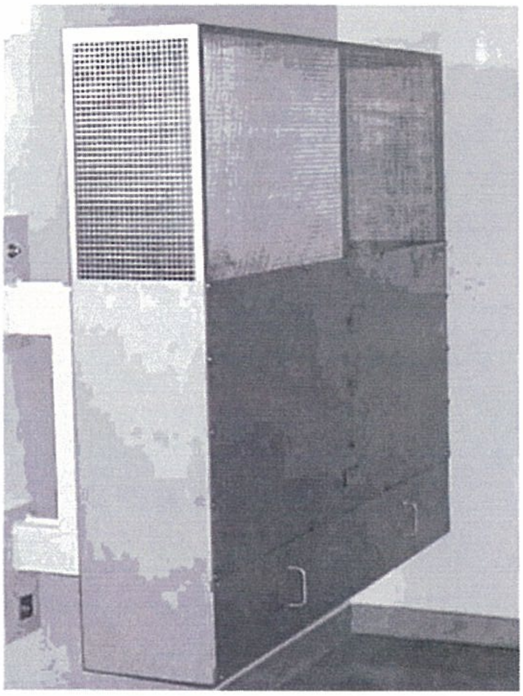
- in case of a design basis accident, it keeps the hydrogen concentration in the water vapor and air mixture below the flame propagation concentration limits throughout the design range of the environment parameters variation in the rooms under the containment;
- in case of a beyond design basis accident, it keeps the hydrogen concentration within the levels appropriate to prevent detonation and rapid burning in large volumes (comparable with the volumes of main compartments in the containment).

The hydrogen removal system equipment includes the set of passive autocatalytic hydrogen recombination units and the test bench for sampling check tests.

The recombination unit is activated when the atmospheric hydrogen contacts the autocatalytic coating on the surface of plates in the cartridge. The chemical combination reaction arises between hydrogen and oxygen, accompanied with heat release. Plates become hot, and heat exchange arises between the plates' surfaces and the environment. As a result of heating, density of gas mixture in the cartridge is reduced. Stable convective flow arises in the recombination unit's casing, resulting in continuous supply of gas mixture to the catalyst and steam removal through the punched holes provided in the casing top.

Passive autocatalytic hydrogen recombination units: positions in the containment





Installed passive autocatalytic hydrogen recombination unit

<u>Melt localization unit</u>

The melt localization system or unit is a unique safety technology designed in Russia. This is a technical facility specially provided to manage heavy beyond design basis accidents at the stages beyond the reactor vessel.

The melt localization unit is mounted in the reactor's concrete vault. It shall:

- receive and keep solid and liquid corium components;
- protect the heat transfer from corium to the cooling water;
- protect the reactor vault against thermal and mechanical corium impacts;
- keep the melt subcritical;
- reduce the propagation of hydrogen and radionuclides into the space in the containment.