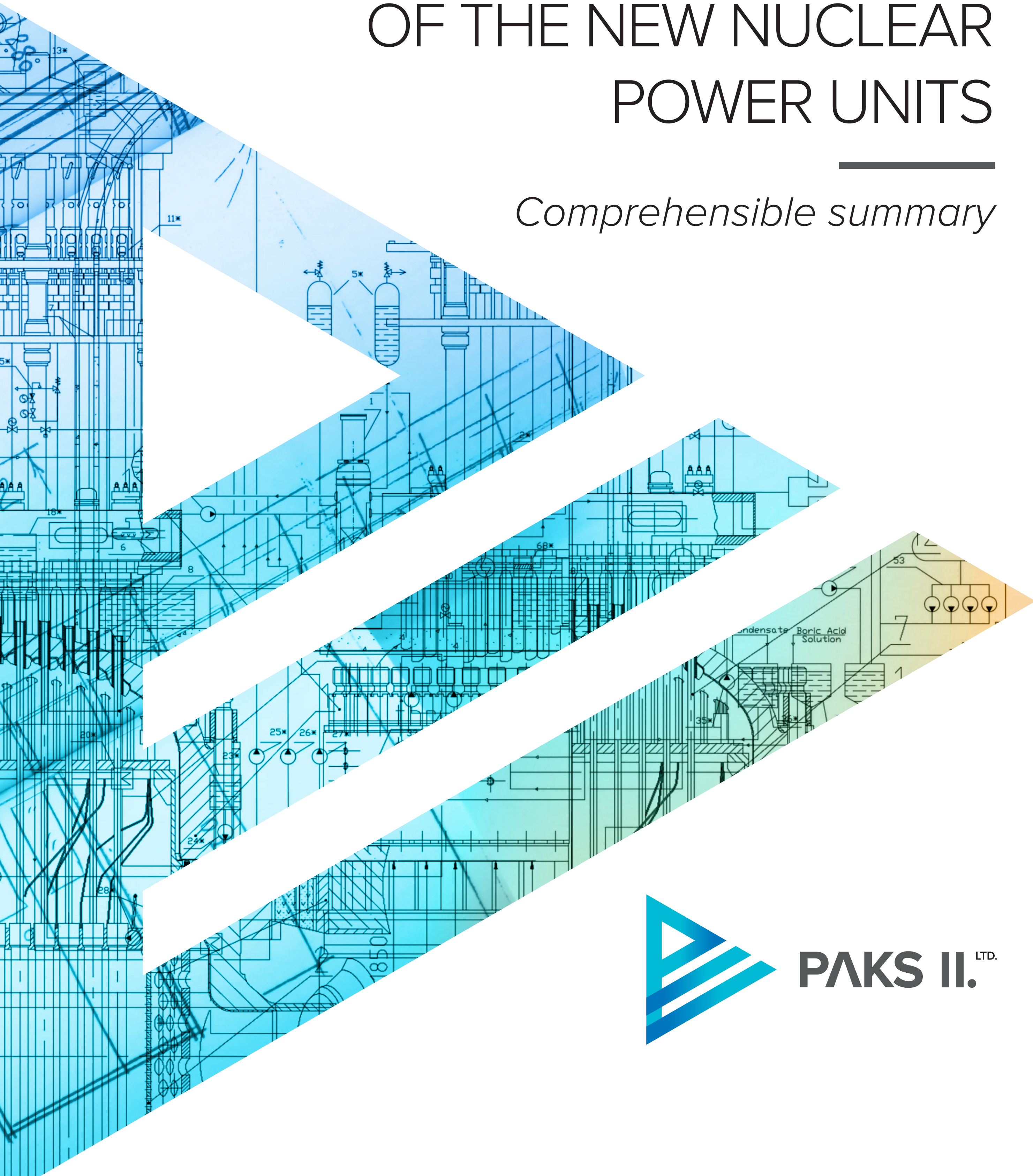
IMPLEMENTATION LICENSING

OF THE NEW NUCLEAR POWER UNITS

Comprehensible summary



PAKS II. LTD.

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PREFACE

The main objective of Paks II. Nuclear Power Plant Private Company Limited by Shares (Paks II. Ltd.) is the preparation for implementation, obtaining necessary licenses and permits, construction and operation of the new nuclear power plant units at Paks site.

The implementation of Paks II. project will make a significant contribution to the domestic economic growth and sustainable supply of safe, affordable and environmentally friendly electricity to Hungarian consumers.

At the present moment, approximately two thirds of the domestic electricity demand is supplied by the nuclear power plant units operating in Paks; however, their lifetime will expire in the 2030-s. In the long run, the two new nuclear power units are intended to replace the phased out old capacities.

Implementation of the new units is the key industrial investment of the century in Hungary. During the peak years of the construction, over 10 000 people will be working at the project and another 10-15 000 workers throughout this country will be involved in the performance of project-related tasks and activities.

The goal of Paks II. Ltd. is to construct a nuclear power plant corresponding to the best technical standards and complying with the most stringent requirements that will be operating safely for decades, securing the supply of environmentally friendly electricity for Hungarian consumers, while boosting the further development of domestic industries, trade and education.

Pursuant to the implementation contracts signed on 9 December 2014, preparation for the construction of two VVER-1200 pressurised water reactor units with the electrical output of 1200 MW each is in progress at the Paks site. Over 6000 licenses and permits are required for the implementation; the most important ones are the so-called facility-level licenses, i.e. the environmental license, site license, implementation license, commissioning license and license for operation.

The environmental license was received by Paks II. project on 29 September 2016, the site license was granted on 30 March 2017. It should be emphasised that the procedure of the Environmental Impact Assessment conducted by Paks II. was qualified by the Secretariat of Espoo Convention as „best practice" to be followed.

The next – most important milestone up to the present moment – will be the issuance of the implementation license by the HAEA. After receiving the above license, Paks II. will be in capacity to obtain further licenses and permits required to commence the activities related to the construction, procurement, equipment manufacturing and installation.

Due to the fact that the project is expected to have an overall impact on the national economy, primarily in the field of long-term electricity generation and supply, Paks II. Ltd. has always put in focus the provision of open and credible information. This summary has taken into account all the above aspects. The purpose of this summary prepared for the general public is to provide a clear and easy understandable overview of VVER-1200 power units to be constructed in Paks and the Preliminary Safety Analysis Report being the fundamental substantiating document of the implementation license application.

VVER-1200 POWER UNITS OF GENERATION 3+

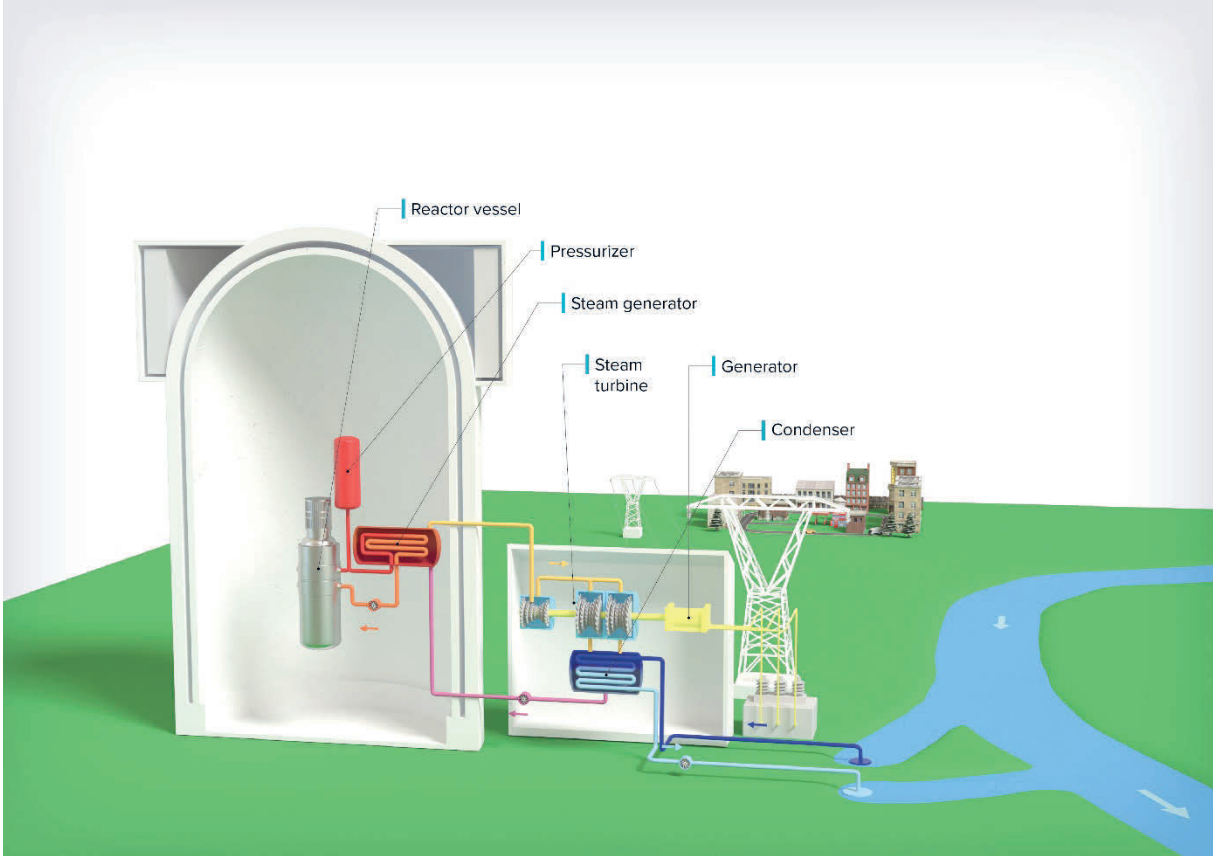
The power unit of VVER-1200 type designed by Rosatom is a modernised power unit of the third generation belonging to the most widely spread reactor type in the world - i.e. to the so-called pressurised water reactors. In case of pressurised water reactors, thermal energy generated by fuel assemblies installed in the reactor pressure vessel is removed by the water coolant under high pressure.

The source of thermal energy in pressurised water reactors is the controlled chain reaction undergoing in fuel assemblies placed in the reactor core, in the process of which fission products and fast fission neutrons are produced from U-235 nuclei under the effect of slow (thermal) neutrons. The greater part of energy released during the nuclear fission is carried over together with fission products in the form of kinetic energy. After the collision with nuclear fuel atoms, these fission products lose their energy appearing in the form of coolant heat. In this case, the water coolant performs two functions: on one side, it functions as moderator facilitating the loss of a great part of energy by fast fission neutrons when colliding with hydrogen atom nucleus and generating slow (thermal) neutrons; on the other side, it assures – by means of removing heat released during nuclear fission – the cooling of fuel assemblies and transfer of thermal energy required for electricity generation, to steam-generators.

Due to safety reasons, the process of electricity generation in pressurised water reactor units is divided into two parts: primary and secondary circuit. The primary coolant comes into contact with the reactor core and reactor structural elements and, correspondingly, contains small amounts of radioactive materials. The secondary circuit is physically separated from the primary, thus, no radioactive materials can enter the secondary circuit. The primary-secondary boundary is the steam-generator and the walls of several thousands of heat-exchanging tubes inside.

Thermal energy released in the reactor and carried over by the primary coolant generates steam in the secondary side of the steam-generator. High-pressure re-heated secondary steam is delivered to the turbine and after expansion rotates the turbine rotor. The turbo-generator located on the same shaft as the steam turbine converts kinetic energy of rotation motions - based on the principle of magnetic conductivity – into electric power.

After passing through the turbine, the steam enters condensers – in case of the new units in Paks, cooled by the Danube water – where this steam is cooled-down and condensed, thus, leading to further improvement of the cycle’s thermal efficiency. The condensate is transferred to steam-generators via feedwater system, closing the secondary cycle.



Schematic overview of the nuclear power plant operation

The commercial operation of power units belonging to the third generation was commenced in the 2000-s; the common feature of these units was the assurance of safety enhancement along with the maintenance of economic competitiveness. The primary objective during the designing process was safety enhancement along with the consideration of nuclear events occurred up to that moment; however, compact arrangement, application of efficient construction methods, better availability factor and longer lifetime were also among important aspects in view of the competitive electricity generation market.

Development of VVER type power units

Rosatom possesses more than 40 years of experience in the design and operation of VVER type power units. 72 power units of VVER type were constructed during the past years and as of June 2020, 62 power units of VVER type are operating worldwide.

The first VVER reactor was commissioned in 1964 in Novovoronezh. The electrical output of the first unit accounted for 210 MW, while the second reactor commissioned in Novovoronezh had an electrical output of 365 MW. The power units built in Novovoronezh served as the basis for further development and the operating experiences gained there played an important role in the designing of VVER-440 type units.

The construction of VVER-440 reactors started in the 1970-s. Robust and safe design of this power unit with 440 MW electrical output belonging to the second generation contributed to the possibility of extending the original 30-year designed lifetime; the majority of operators used this possibility of plant lifetime extension. As a result of the lifetime extension, many VVER-440 power units are still in operation today. Four nuclear power units operating in Paks represent a modernised, further improved version of this reactor type. The units of Paks Nuclear Power Plant were connected to the grid and commenced electricity generation during 1983-1987; due to the safe condition, the lifetime of the units of Paks Nuclear Power Plant was extended and the license for another 20 years of operation was granted.

The next stage of VVER history is presented by VVER-1000. Along with a significant increase of the unit's electrical output, priority attention was paid to safety enhancement implemented via the use of several innovative safety-related solutions. A VVER-1000 power unit commissioned in 2007 in Tianwan (China) was the first one on the world scale equipped with a core catcher. VVER-1000 type with 31 reactors currently being in operation (as of June 2020) is the most frequent VVER unit type in the world. The total VVER-1000 operating experience accounting for 500 reactor-years gained in the reactors operating in Russia, Ukraine, Czechia, Bulgaria, Iran, India and China confirms high nuclear safety level of this reactor type.

The VVER-1200 power unit is the newest and most modern member of the VVER reactor family, the design of which is based on a several-decade experience of Rosatom in the development and operation of nuclear power plants. The first two units of VVER-1200 type were commissioned in Russia: at the moment, two power units are generating electricity in Novovoronezh and another one in Sosnovy Bor; one more power unit of this type is currently under construction at the site in Sosnovy Bor. In addition to the projects being under implementation in Russia, 11 power units of this type are being constructed all around the world from Finland to Belarus and Turkey.

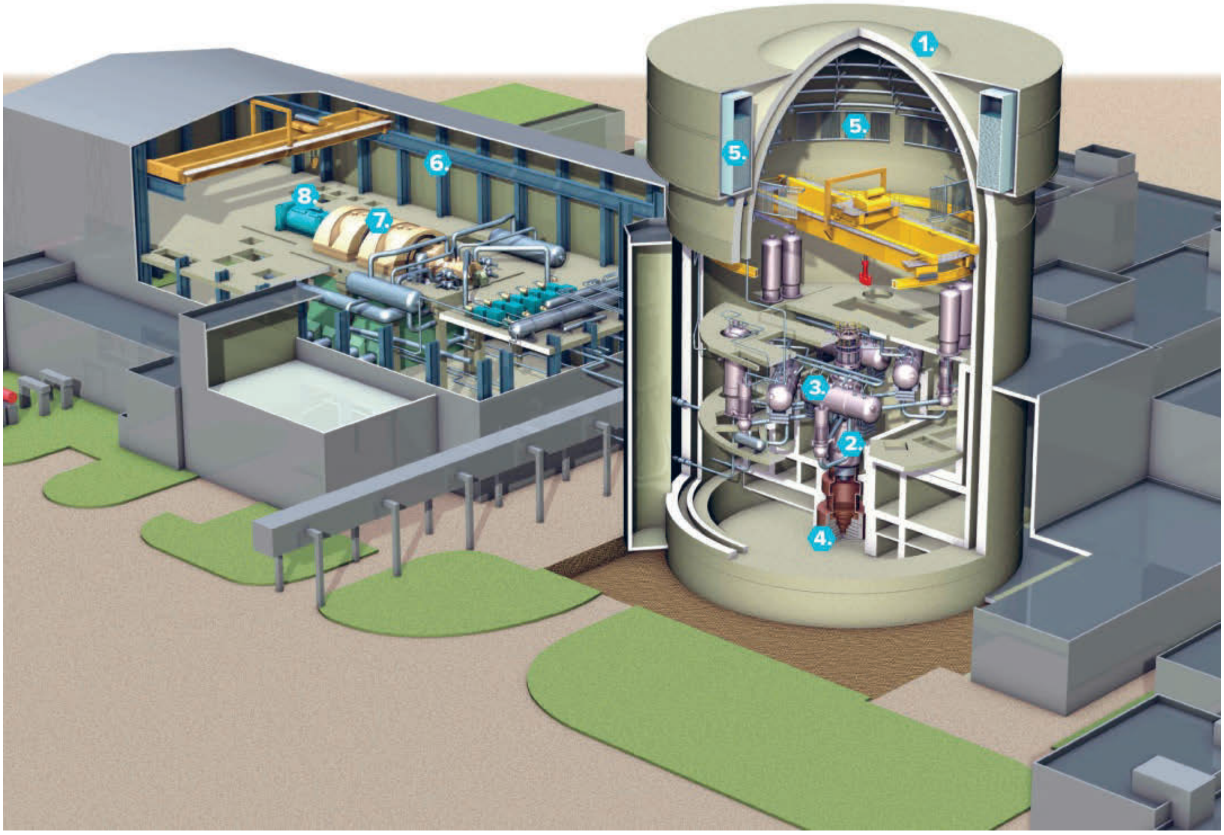
Key characteristics of VVER-1200 3+ power unit

The most important technical and safety characteristics of VVER-1200 units designed by Rosatom are as follows:

Reactor type	pressurised water reactor
Thermal output	ca. 3220 MW
Electrical output	ca. 1200MW*
Fuel	U-235, max. enrichment 4,95%
Number of primary loops	4
Number of steam-generators	4
Coolant temperature at reactor outlet	328,9 °C
Coolant temperature at reactor inlet	298,2 °C
Primary pressure	16,2 MPa
Secondary pressure	7 MPa
Primary flow-rate	86000 m^3/h
Amount of steam produced by steam-generators	6400 t/h
Safety systems	active and passive safety systems
Designed lifetime	min. 60 years
Availability factor	>90%
Plant efficiency	37%
House load consumption	7.1%
Time required for maintenance (for a 7-year cycle)	4 x 16 days 2 x 24 days 1 x 30 days
Demand for operating personnel (in terms of electrical output)	0,42 person/MWe

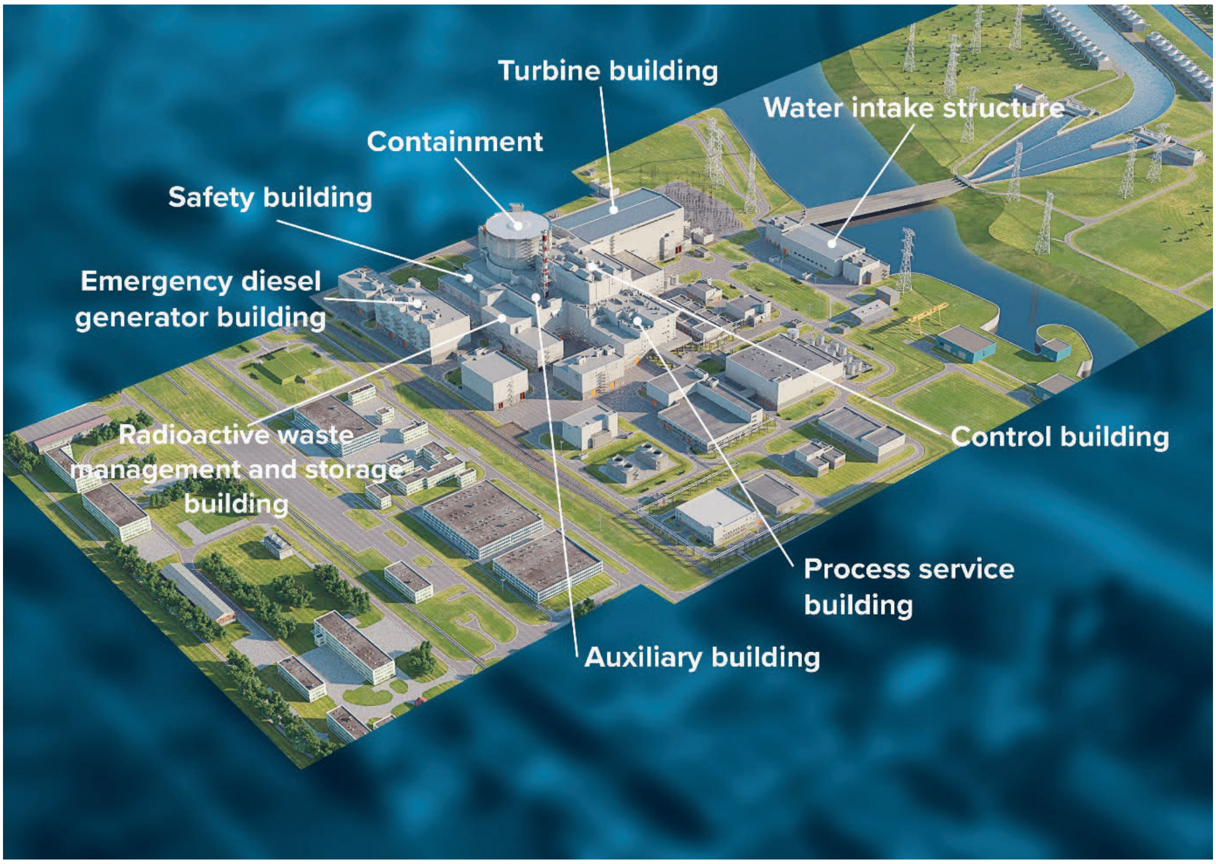
* Based on the turbogenerator system planned for use, the rated electrical output of new Paks units will be 1262 MW.

Schematic design of a VVER-1200 power unit:



1. Containment
2. Reactor pressure vessel
3. Steam-generator
4. Core catcher
5. Passive containment heat removal system
6. Turbine building
7. Turbine
8. Generator

Most important buildings of a VVER-1200 power unit



The optimal arrangement in case of VVER-1200 units envisages a two- or a multi-unit nuclear power plant. In most cases, buildings and civil structures are connected with operation of the given unit; however, the use of a two-unit arrangement provides the possibility for implementing so-called operation support systems not related directly to the unit operation at the common facility level. The majority of plant buildings are aimed at servicing either one or the other plant unit; however, several buildings are intended for common use by the units, which is advantageous in terms of construction- and subsequent operational costs.

Many various buildings and civil engineering structures are required to operate a nuclear power plant; herein below we will describe the most important ones concerning safety and electricity generation.

Containment

The containment is the main building of the power unit – all other buildings and engineering structures of the so-called "nuclear island" are grouped around it. In terms of radiation protection, containment is a part of the limited access area accommodating the reactor pressure vessel, reactor coolant system, steam-generators, spent fuel pool containing spent fuel assemblies, emergency cooling systems, as well as associated auxiliary-, service and control systems.

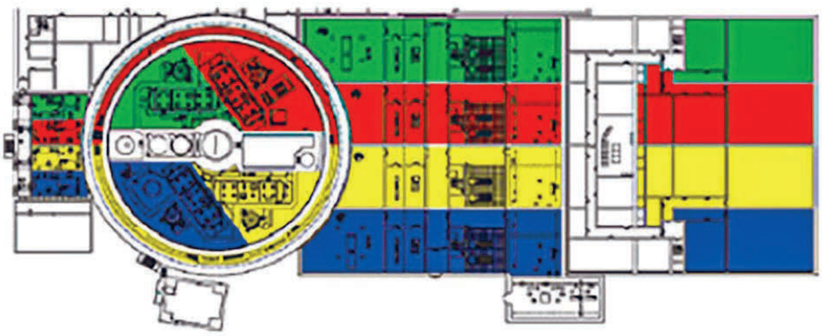
The purpose of the double containment is the isolation of systems containing radioactive materials from the environment, on one side, and the protection of process systems located within the containment against external hazards, on the other side. Secondary containment with a 52,4 m diameter is made of reinforced concrete, its height including the roof – not fulfilling any safety function – is 72 m. The secondary containment wall-thickness varies in different points, sometimes being several-meter thick. The primary containment with a 44 m inner diameter is a cylindrically shaped building sealed by a dome. The wall-thickness in the containment cylindrical area is 1,2 m. Walls of the primary containment made of pre-stressed reinforced concrete are protected by a 6 mm thick carbon steel lining, assuring hermetic isolation of the containment inner space from the environment.

Building of process service systems

The building of process service systems is designed for accommodating over-pressure protection systems of steam-generators, reactor coolant system valves and pipelines of feedwater- and makeup water systems. Each of these systems comprises four safety trains, which are physically separated from each other within the building.

Safety building

The safety building is designed for the installation of equipment and associated pipelines of safety systems. Four independent safety system trains based on the principle of redundancy will be physically separated from each other inside the building (installed in different rooms). The safety building will also accommodate those technology- and component cooling systems, which are intended for servicing cooling- and residual heat removal systems of the spent fuel pool.



Redundant trains of safety systems

Control building

The control building accommodates the unit main control room used for the performance of unit control and supervision by the operating personnel. The control building includes also the emergency control room used for the unit control and supervision in case of unavailability of the main control room due to some reasons. The building contains instrumentation and control systems required for the operation of main- and emergency control rooms, communication systems and related auxiliary equipment.

Auxiliary building

In terms of radiation protection, auxiliary building is a part of the limited access area designed for the installation of water- and gas treatment systems, radioactive waste processing system and certain pieces of equipment belonging to ventilation systems of the limited access area.

Emergency diesel generator building

The building is designed for the installation of emergency diesel generator and protection against external hazards. Diesel generators assure emergency power supply required for safe shutdown, cool-down and maintenance of safe shutdown condition of the power unit in case of any loss of normal operational power supply. In order to guarantee independent arrangement of four diesel generators and auxiliary systems, the building is divided into four sections, physically separated from each other. The system of diesel generator fuel supply and storage is able to assure a 72-hour long fuel supply to diesel generators without re-filling.

Fresh fuel storage building

Fresh fuel storage is used for both power units, thus, it is arranged in a common building. The building guarantees protected and safe storage of fresh fuel assemblies and absorber rods necessary for the operation of the units. The building contains equipment and devices required for movement and transportation of fresh fuel assemblies and absorber rods within the plant site.

Radioactive waste management *and storage building*

Radioactive waste management and storage building is used by both power units. The building is designed for processing solid radioactive wastes generated during the operation of power units, and interim storage of solid and solidified radioactive wastes. The building is divided into two areas: the area for solid radioactive waste management and the area for solid radioactive waste storage. The building design assures sufficient space for the interim storage of solid and solidified radioactive wastes with low- and medium activity level until their transportation to the National Radioactive Waste Repository in Bátaapáti. Solid radioactive wastes with high activity level – generated in small amount during the operation of power units – will be also stored in the building until decommissioning.

Turbine building

Turbine building is designed for the installation of a steam turbine, turbogenerator, as well as service- and auxiliary systems required for their operation, e.g. condensate purification system, oil systems of the turbine and turbogenerator, certain equipment of feedwater systems.

Cooling water pump station for turbine hall consumers (water intake structure)

Water intake structure accommodates equipment and systems assuring water intake control and supervision for the fresh water supply to provide cooling for turbine condensers and other equipment located in the turbine building, and to replenish water losses. Fresh water is supplied from the Danube via cold-water channel.

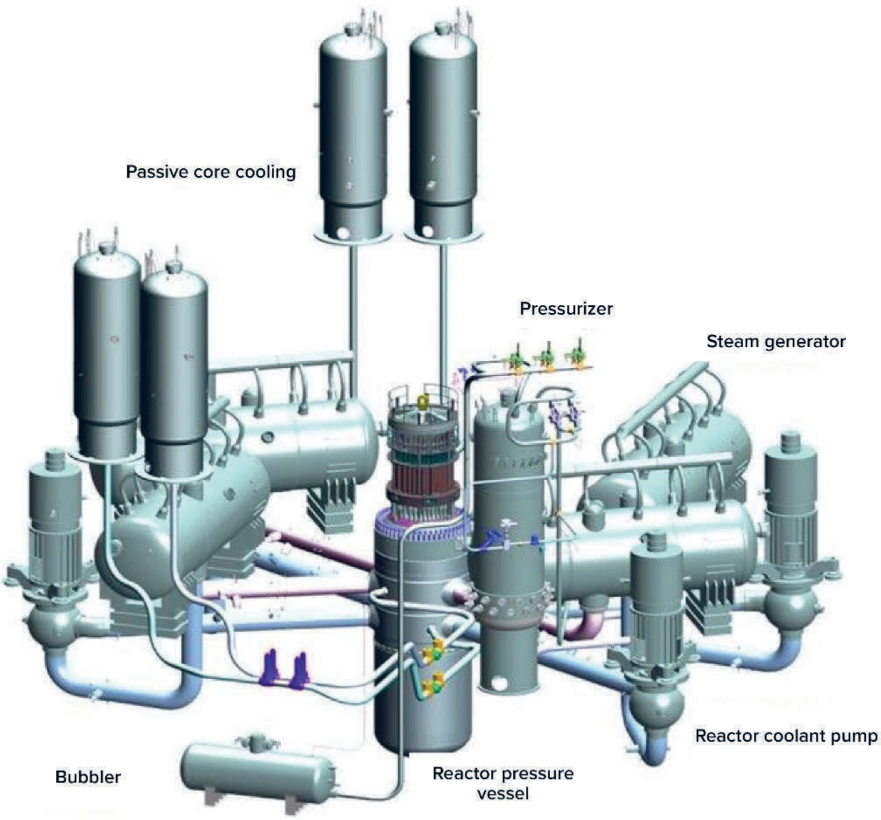
Main systems of VVER-1200 power units

Many systems and pieces of equipment are required for unit operation, control, supervision and performance of protection-focused interventions. Some of those are special systems and equipment applied only in the nuclear industry, while some of them are found in standard power plant technologies; however, the latter are designed in accordance with more stringent nuclear safety and environmental qualification requirements, due to their use at the nuclear power plant. The third group includes equipment and systems known from other industrial or power plant applications (for instance, steam turbine and turbogenerator) but in their case as well one should take into consideration differences originating from sizes and dimensions.

Herein below, we will describe the most important systems and equipment of VVER-1200 power units.

Reactor coolant system

Reactor coolant system is placed in the containment. Main equipment of the reactor coolant system covers the reactor, steam-generators, reactor coolant pumps, reactor coolant loops and pressurizer system. The reactor coolant system of a VVER-1200 power unit consists of four reactor coolant loops, symmetrically arranged around the reactor. The reactor coolant system supplies thermal energy for the steam turbine and turbogenerator located in the secondary circuit. The boundary between the reactor coolant system and secondary circuit is the surface of steam-generator heat-exchanging tubes. The reactor coolant system performs fundamental safety functions, as well. The reactor coolant system assures control over chain reaction undergoing in the reactor core (safety function related to reactivity management), maintains proper and continuous cooling of the reactor core (safety function related to core cooling) and forms a physical barrier preventing the escape of radioactive materials available in the reactor coolant system (safety function related to the retention of radioactive materials). Pressure of the reactor coolant system at VVER-1200 power units is 16,2 MPa, while coolant flow-rate is 86000 m^3/h at reactor thermal power of 3200 MW.



Reactor coolant system

Reactor

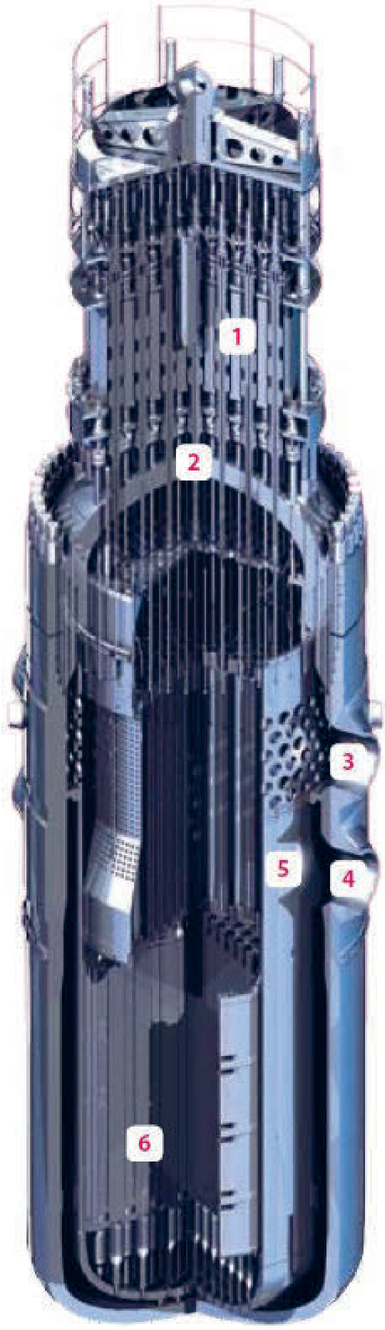
The reactor is the most important system of the reactor coolant system and – in terms of nuclear safety – of the entire unit. Accordingly, designing, manufacturing and subsequent operation of the nuclear reactor shall be in full compliance with extremely stringent requirements. Main reactor parts are the upper unit, reactor pressure vessel, reactor internals (e.g. reactor cavity) and active core. The reactor is 14100 mm high; its weight is 876 tons without the coolant.

The core of a VVER-1200 reactor is made up of 163 hexagonally shaped fuel assemblies. One fuel assembly consists of 312 fuel rods. Hermetically sealed fuel rod claddings manufactured from zirconium alloy are loaded with uranium dioxide fuel pellets having the diameter of 7,6 mm and maximum enrichment of 4,95% (by U-235). Ceramic matrix of uranium dioxide and hermetic sealing of fuel rods play a decisive role in the retention of radioactive materials originating in the process of unit operation.

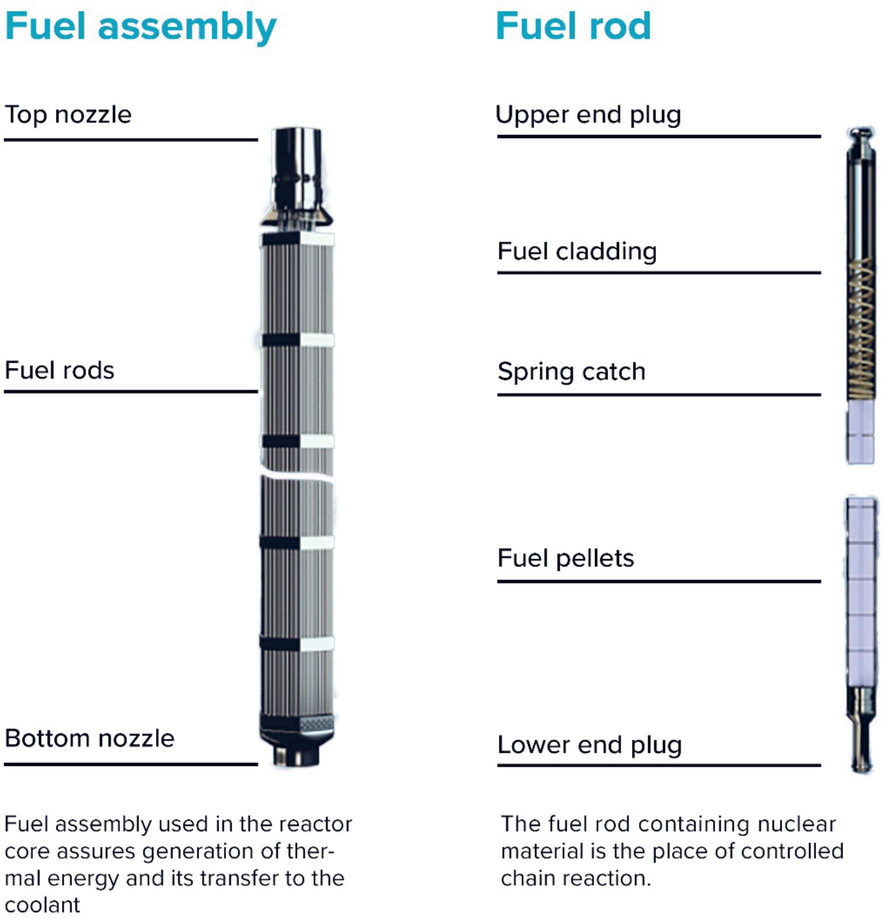
Fine regulation of the reactor power is carried out by changing the concentration of boric acid added to the water coolant, using neutron-absorbing properties of boron. For the purpose of fast power regulation or prompt reactor shutdown, there are 121 control rods (absorbers) in the reactor containing boron- and dysprosium alloys (also characterised by good neutron-absorbing properties).

Due to its extremely important role in nuclear safety, the reactor should be regularly examined during the unit operation (e.g. by non-destructive methods of material testing).

Reactor



1. Upper unit
2. Dome of reactor pressure vessel
3. Hot leg flange
4. Cold leg flange
5. Reactor cavity
6. Fuel assemblies



Nuclear fuel

Reactor coolant loops

Four reactor coolant loops are used to link the reactor, steam-generators and reactor coolant pumps, assuring coolant flow. The nominal outer diameter of a reactor coolant loop is 990 mm; wall-thickness is 70 mm and total length is 146 m.

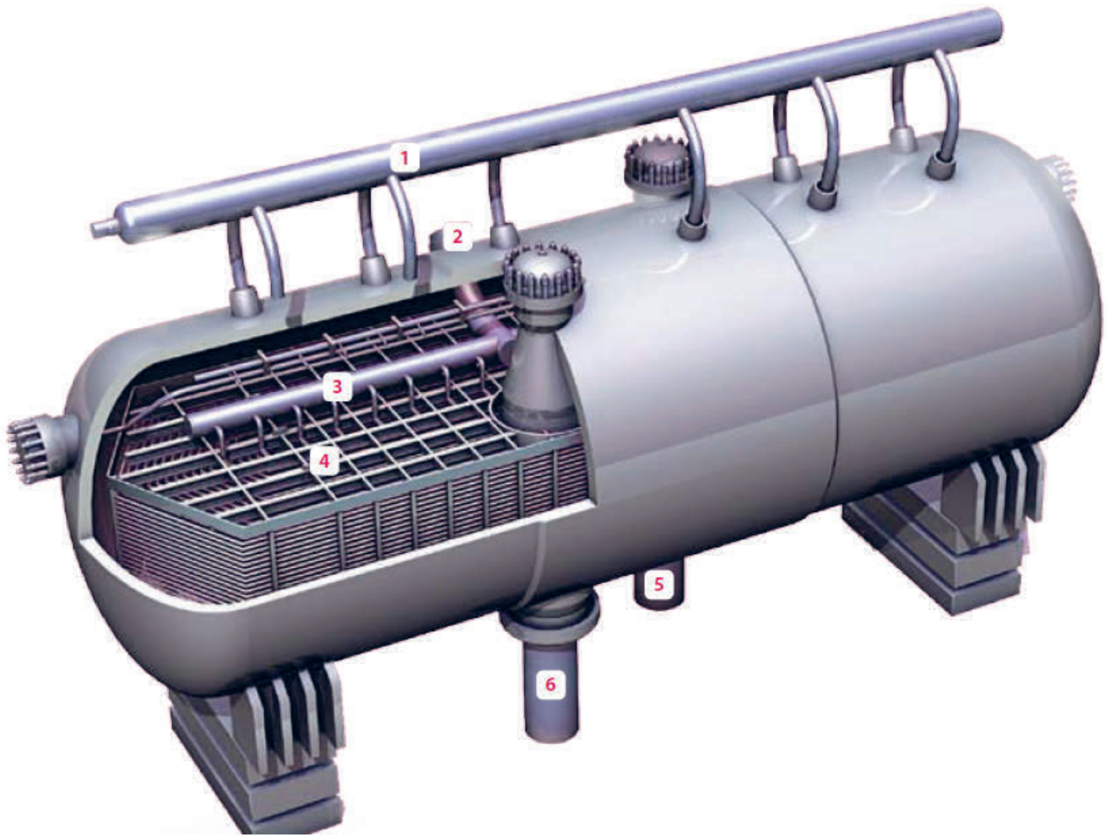
Steam-generator

Traditionally, VVER power units are designed with horizontally arranged steam-generators, compared to the vertical arrangement of steam generators in case of other pressurised water reactor units. A huge advantage of horizontal arrangement confirmed by several decades of VVER operating experience is the significant decrease of corrosion processes.

Steam-generators produce reheated dry steam using thermal power generated in the reactor core. The heat transfer process takes place via the surface of 10858 steel heat-exchanging tubes with the diameter of 16 mm and wall thickness of 1,5 mm located horizontally in the U-shape of steam-generators. Tube ends are fastened by welding to the internal surface of steam-generator inlet and outlet collectors. The reactor coolant circulates inside these heat-exchanging tubes, assuring that radioactive materials cannot escape into the steam-generator secondary side, and, correspondingly, to the secondary circuit.

Steam-generators play a certain role in the core cooling, that is why feedwater supply to steam-generators – in addition to normal feedwater supply lines – is assured also with the help of emergency feedwater system. For the case of station blackout or loss of steam-generator feedwater supply, steam-generators of VVER-1200 power units are equipped with the passive heat removal system.

Steam-generator

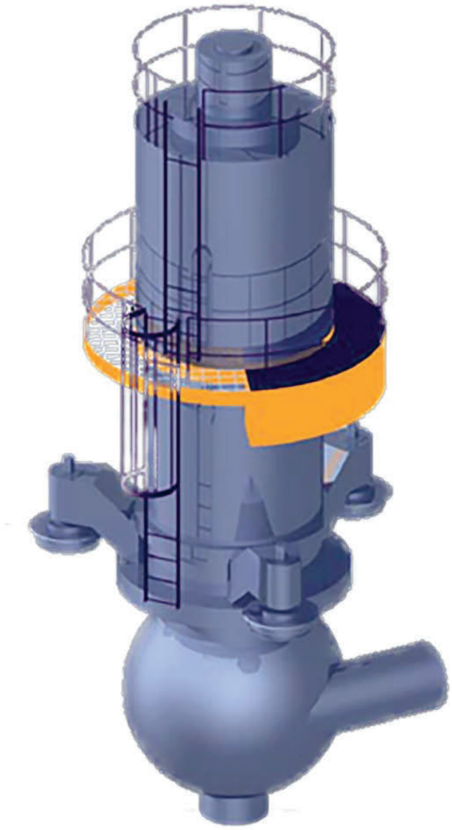


1. Steam-generator main steam header
2. Feedwater supply
3. Feedwater distribution collector
4. Heat-exchanging tubes
5. Hot leg collector
6. Cold leg collector

Reactor coolant pump

Coolant circulation between the reactor and steam-generators is assured by vertically arranged centrifugal pumps - one for each of the reactor coolant loops. Coolant flow-rate assured by the pump is 21500 m^3/ h at the rated reactor power.

Cooling of the pump motor and lubrication of all pump bearings are done by water, which allows to significantly reduce the risk of fire.

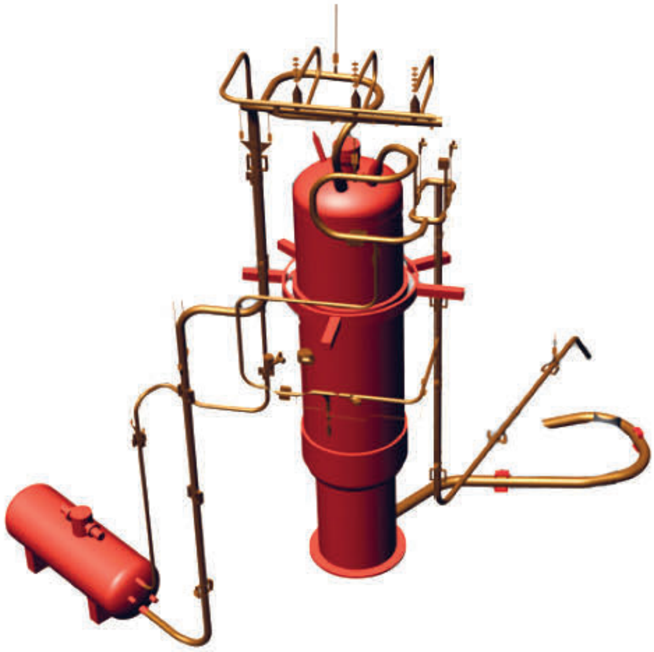


Reactor coolant pump

Reactor pressure control system

Another element of the reactor coolant system is the pressurizer connected to the reactor coolant system in a non-isolated manner. The pressurizer system is intended for the regulation of reactor coolant system pressure, is responsible for the overpressure protection of the reactor coolant system and compensates the volume changes in the coolant. During the unit start-up, pressurizer system creates pressure in the reactor coolant system and assures pressure reduction in case of reactor coolant system cool-down.

The main equipment of pressurizer is a 55 m^3 pressure control vessel, which is divided into water - and steam space during the unit operation. The pressure of steam space above the water space is identical to that of the reactor coolant system. There is an injection pipeline connected to the pressurizer steam space, which can be used in case of necessity for water injection to protect the reactor coolant system equipment against over-pressure and to reduce the pressure of steam space and reactor coolant system.



Reactor Pressure Control System

Passive emergency core cooling system (hydroaccumulators)

In the event of loss-of-coolant accidents accompanied by pressure reduction in the reactor coolant system, temporary cooling of the reactor core - until the actuation of active emergency core cooling systems - is assured by hydroaccumulators. Four hydroaccumulators are connected to the reactor coolant system of VVER-1200 power units. The internal pressure in hydroaccumulators is maintained by the N_2 blanket located above the water level. Should the primary pressure drop below 5,9 MPa, borated water stored in the hydroaccumulators will passively flow to the reactor pressure vessel under the effect of N_2 blanket pressure being higher than that of the primary circuit, assuring cooling of the reactor core.

Hydroaccumulator

Main steam supply system

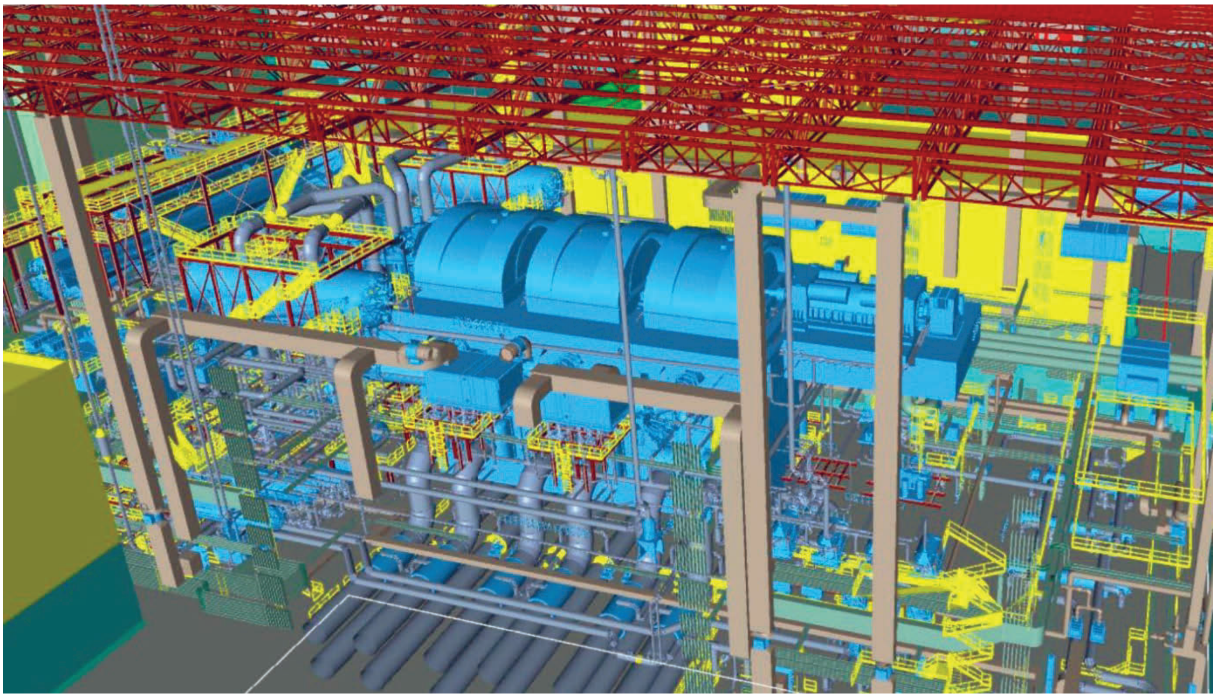
The task of the main steam supply system is to transfer the reheated steam with the pressure of 6,9 MPa and temperature of 286 °C generated by steam-generators at the rate of 6400 t/h, to the turbine.

The system is equipped with main steam isolation valves designed for the prompt isolation of steam-generators from the main steam supply system in case of a steam line break, feedwater line break or damage of a steam-generator heat-exchanging tube. Main steam supply system is equipped also with the atmospheric discharge valve to discharge steam into the atmosphere to support the unit cool-down during normal operation or in case of emergency.

Turbine

The steam turbine is designed for the transformation of high-pressure re-heated steam energy arriving from steam generators, into mechanical energy. The steam flows across the turbine and in the process of expansion transforms into mechanical energy. The mechanical energy appears on the turbine rotor and drives the turbogenerator.

In case of VVER-1200 type reactors, Rosatom recommends the use of MLZ type turbine manufactured by Power Mechanics or ARABELLE type steam turbine manufactured by General Electric. The latter type will be implemented in Paks. The half-speed (1500 1/min) ARABELLE steam turbine consists of one low-pressure-intermediate pressure cylinder and three low-pressure cylinders.



Turbogenerator

Turbogenerator

Turbogenerator – the rotor of which is placed on the same shaft as that of the steam turbine – transforms mechanical energy into electric power.

The turbogenerator is a four-pole half-speed (1500 1/min), three-phase synchronous machine (with rated active power of 1270 MW and rated terminal voltage of 24 kV) with hydrogen – cooled rotor and stator housing and water-cooled stator winding.

A rotating diode exciter is connected to the turbogenerator, and a 24 kV bus bar ensures the electrical connection between the turbogenerator and the generator transformer.

Nuclear safety characteristics of VVER-1200 power units

During the development of VVER-1200 power units, specific attention was paid to safety enhancement. Rosatom took into consideration all nuclear safety requirements of the International Atomic Energy Agency and made the following conclusions based on the lessons learnt from nuclear events and accidents:

- it is necessary that long-term cooling of the reactor core should be assured in case of station blackout;
- in addition to the primary ultimate sink, an alternative ultimate sink should be assured for long-term residual heat removal;
- technical and technology-related solutions should be prepared for managing very low probability core meltdown accidents.

Safety concept applied for this type of power units is based – in the full compliance with international recommendations in the nuclear safety field – on the defence-in-depth principle meaning that various independent defence levels ensure that even extremely low probability failures and deviations from normal operating conditions will be detected, compensated and managed, as appropriate. Another feature of the applied safety concept is inherent safety. Inherent safety means that in case of an unauthorised reactor power increase, the chain reaction – after achieving some certain level – will terminate on its own, without any human intervention, according to the law of physics, and the reactor will enter the safe, so-called sub-critical state.

In sense of stringent safety-related design requirements, it should be confirmed for the Paks site that in case of hazards resulting from natural disasters (e.g. great earthquakes) with the frequency of occurrence exceeding 10^{-5}/year (return frequency of 100,000 years) and external events induced by man-made activities (e.g. heavy aircraft crash) with the frequency of occurrence over 10^{-7}/year (return frequency of 10,000,000 years), safe state of the power unit can be maintained. In terms of technology- or operator-related failures the confirmation of compliance should be conducted for initial events with the frequency of occurrence exceeding 10^{-6}/year (return frequency of 1,000,000 years).

Results of safety analysis

The fulfilment of nuclear safety requirements should be confirmed by means of safety analyses conducted in accordance with international recommendations and national legal regulations, based on site-specific hazards and their frequencies for the Paks site, as well as with respect to the design and operating characteristics of nuclear power units to be constructed in Paks.

The results of completed safety analyses demonstrate that all the requirements specified in the Nuclear Safety Code in respect of both environmental releases and connected additional radiation exposure of the population and probability of accident occurrence are fulfilled in full.

Radiation exposure, release of radioactive materials

Along with the analysis of low probability accidents and very low probability severe accidents, the analysis and evaluation of the personnel exposure anticipated during the normal operation of the nuclear power unit and public radiation exposure originating from environmental releases was carried out as well. The radiation protection design of the power units was based on the ALARA principle (As Low As Reasonably Achievable).

Due to small time demand for the unit maintenance, estimated doses arising from external radiation exposure are low. The efficient retention of radioactive materials generated in the reactor coolant system is assured by physical barriers. The spread of radioactive contamination is limited by water- and gas treatment systems and aerosol- and iodine filters of ventilation equipment – as a result, doses arising from internal radiation exposure will be low too. Taken as a whole, radiation exposure of the personnel – even in the case of the most exposed workers – will remain far below the annual dose limit (20 mSv).

The volume of radioactive environmental discharge during normal operation – as a result of technical and technology-related solutions described herein above in terms of the operational exposure, assuring the retention of radioactive materials and limiting the spread of contamination – will remain far below the discharge limits established in the environmental license. Thus, the annual value of the additional radiation exposure affecting the most exposed public group and arising from radioactive discharges will be below 100 nSv. In terms of radiation exposure, this dose is practically negligible; this value is identical to the radiation exposure resulting from an hour-long natural background exposure.

Safety systems of VVER-1200 power units

In accordance with defense-in-depth principle, nuclear safety of VVER-1200 power units is assigned to certain protection levels and is based on safety systems timely and efficiently performing all necessary interventions. Safety systems of VVER-1200 power units have been designed with respect to the following principles:

Passive safety systems (systems able to perform their functions without power supply or human interventions) are designed for managing very low probability events and severe accidents covered by design extension conditions.

The design of active safety systems (power supply dependent systems) is based on the **principle of redundancy.** The active safety systems are usually composed of four parallel trains; the operation of one of them is already sufficient for the full-scope fulfilment of the expected safety intervention.

Spatial **physical separation** of safety system redundant trains guarantees the protection of these systems against common-cause failures (e.g. fire).

Along with redundancy, there is another important principle – **principle of diversity** – i.e. performance of safety actions and interventions is carried out with the use of different equipment and systems preferably based on various operating principles (e.g. passive and active systems).

Most important systems of VVER-1200 power units are as follows:

Most important active safety systems	
High pressure safety injection system	4 x100%
Low pressure safety injection system	4 x 100%
Emergency boron injection system	4 x 50%
Emergency feedwater system	4 x 100%
Containment sprinkler system	4 x 50%
Reactor emergency cool-down system	4 x 100%
Intermediate cooling circuit for essential consumers	4 x 100%
Essential service water system	4 x 100%
Heating-, ventilation- and air-conditioning systems important to safety	4 x 100%
Containment separation system	4 x 100%
Borated water storage system	2 x 100%
Emergency gas removal system	2 x 100%
Primary overpressure protection system	3 x 50%
Secondary overpressure protection system	2 x 100%

Emergency diesel generators	4 x 100%
Reactor trip system and engineered safety feature actuation system	4 independent trains

Passive safety systems – design basis accidents	
Passive emergency core cooling system (hydroaccumulators)	4 x 33%
Containment hydrogen control system (emergency)	1 x 100%
Containment	✓

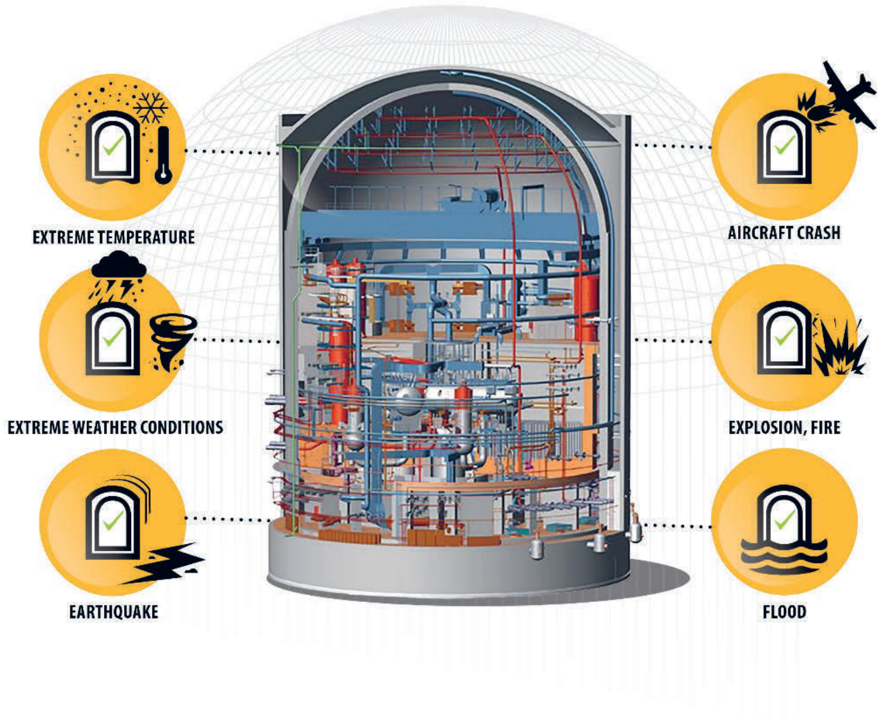
Most important systems – accident situations	
Steam-generator passive heat removal system	4 x 33%
Containment passive heat removal system	4 x 33%
Core catcher	✓
Containment hydrogen control system (accident)	✓

VVER-1200 power units are designed with several safety systems that should be described in particular:

Primary and secondary containment

The reactor, its process and auxiliary systems, as well as the spent fuel pool designed for the intermediate storage of spent fuel assemblies are protected by the primary containment. The containment inner wall is a pre-stressed reinforced concrete structure with steel lining. In addition to active and passive heat removal systems, the primary containment ensures the retention of radioactive materials and prevents their escape into the environment in case of operational occurrences and severe accidents.

The secondary containment made of reinforced concrete is able to protect the primary containment even in case of a heavy aircraft crash. The containment is designed to withstand a 0,34 g horizontal acceleration earthquake, extreme wind loads (e.g. tornado), heat and ice loads, loss of air pressure due to external explosion.

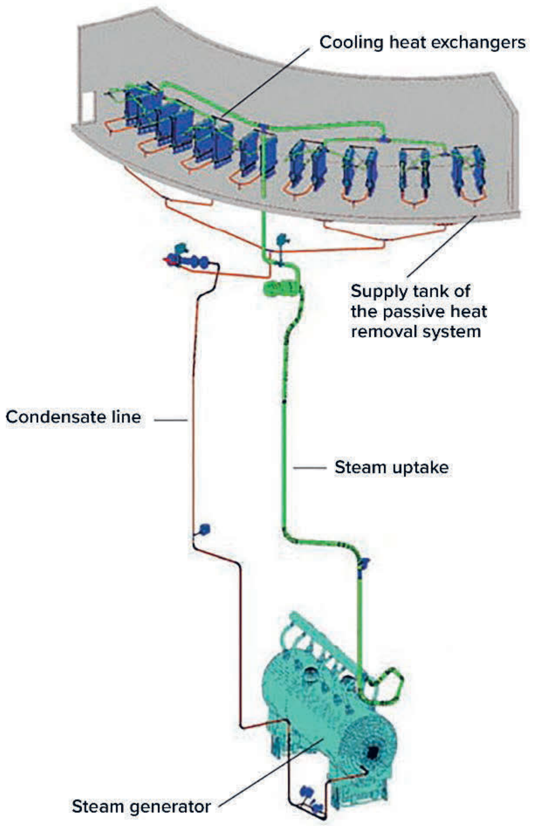


Primary and secondary containment

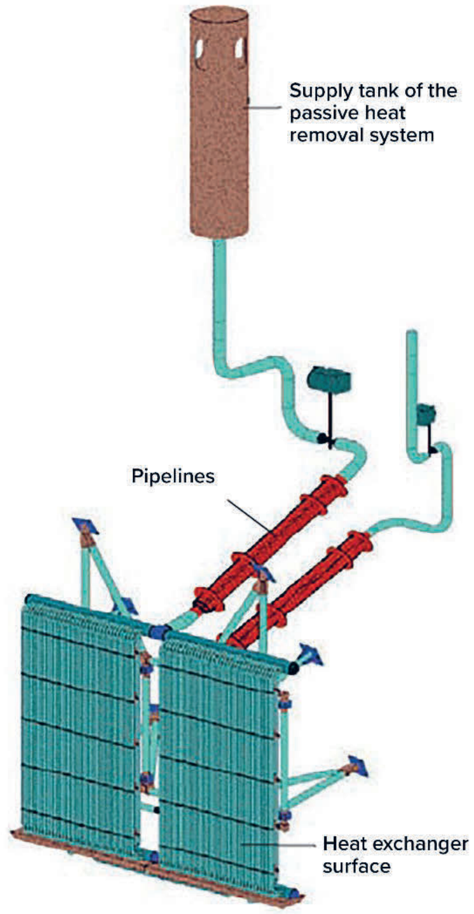
Residual heat removal

After the shutdown, a large amount of energy (so-called residual heat) is released in the reactor during some certain time period, due to the continuing decay of radioactive elements in the fuel assemblies. In order to ensure the removal of residual heat released in the reactor core, power units are equipped with passive heat removal systems. In the event of station blackout or loss of steam-generator feedwater supply – when the removal of residual heat to the Danube cannot be assured by active heat removal systems – this residual heat can be released into the atmosphere by means of passive heat removal systems. The process is implemented via a heat exchanging tube, preventing the escape of radioactive materials into the environment.

The containment is designed with a passive heat removal system aimed at the limitation of temperatures prevailing under accident circumstances in the case of unavailability of the water sprinkler system intended for the cooling of containment space.



Steam generator passive heat removal system



Containment passive heat removal system

Hydrogen control

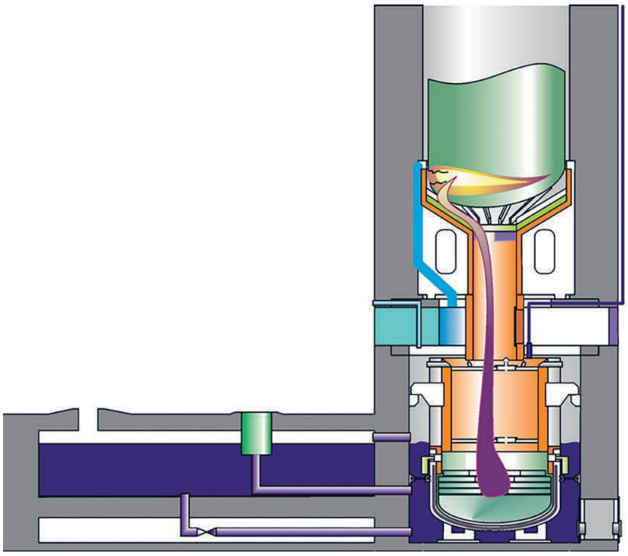
The chemical reaction being the result of interaction between fuel rod zirconium claddings and water steam under accident conditions may lead to the generation of hydrogen and pose the risk of fire and explosion. Catalytic oxidation of hydrogen and maintenance of low-level hydrogen concentration is assured by passive autocatalytic hydrogen recombiners. The recombiners are installed in all such points of the containment annular space that may pose the risk of hydrogen concentration increase based on the analysis results.

Hydrogen recombiner

Core catcher

The nuclear power plant to be built is also prepared to manage severe accidents occurring with very low probability. In order to mitigate the consequences of core meltdown, the reactor design implies the installation of a core catcher. The core catcher is aimed at preventing corium-concrete interactions and reducing the amount of hydrogen generation and environmental release of radioactive materials.

A core catcher is a special device installed underneath the reactor pressure vessel with the purpose to receive corium in case of reactor vessel damage. The core catcher device is filled with special aluminium- and iron oxide mixture with neutron-absorbing properties allowing to reduce the corium specific heat release. The above mixture is also supplemented with gadolinium preventing – due to its neutron absorbing characteristics – corium chain reaction. In case of severe accidents, the steel tank of the core catcher can be cooled from outside with the use of borated water.



Core catcher

Location of the new units

The two new nuclear power units will be constructed northwards from the presently operating units of Paks Nuclear Power Plant, on the so-called plant expansion area. The plant site will accommodate all other systems and buildings necessary for subsequent plant operation, as well as Construction and Erection Base buildings and areas used during the construction period.

Visualisation of the new nuclear power plant

Implementation Licensing

Rules for the application of nuclear energy in Hungary are laid down in the Act CXVI of 1996 on the Atomic Energy Govt. Decree No. 118/2011. (VII.11.) and its annexes presenting the Nuclear Safety Code contain specific regulatory rules and requirements for the licensing procedure, which are essential in terms of obtaining regulatory licenses and permits. The competent licensing authority in case of nuclear facilities is the Hungarian Atomic Energy Authority (HAEA); however, the regulator may involve further specialised authorities competent in other topics.

The implementation licensing procedure starts upon the submission of the implementation license application to the HAEA. General provisions concerning the content of implementation license application are specified in the Govt. Decree No. 118/2011. (VII.11.), while more detailed requirements to the application content and form are described in the respective HAEA Guide. The implementation license application consists of substantiating and supporting documents.

SUBSTANTIATING DOCUMENTS	
1.	Preliminary Safety Analysis Report
2.	Facility Model
3.	Preliminary Nuclear Accident Preparedness and Response Plan
4.	Long-term storage and disposal of radioactive waste originating during the operation of new power units in Paks and spent nuclear fuel
5.	Modernisation strategy for Instrumentation and Control systems
6.	Determination of safety area
7.	Demonstration of the implementation activities time-schedule and preliminary version of the general construction plan
8.	Proof of ownership or trust in real estate belonging to the safety area of the nuclear facility
9.	Effective local building code and regulatory plan for the plant site
10.	Independent safety analysis
11.	Documentation of the independent technical expert review
SUPPORTING DOCUMENTS	
12.	Deterministic safety analysis
13.	Probabilistic safety assessment
14.	Potential impact of the new builds and related activities on the safety of existing nuclear facilities
15.	Calculation and analysis results

The official administrative deadline for the implementation licensing procedure is 12 months (that may be extended by the authority by another 3 months). The Atomic Act provides for the involvement of the following special authorities for the review of certain professional issues:

- in respect of environmental and nature protection, mining supervision issues: Government Office of Baranya County
- in respect of fire protection and disaster management issues: Directorate General for Disaster Management of the Ministry of Interior

To support the regulatory decision-making process connected to the implementation license application, the HAEA involves international expertise. The authority has concluded an agreement with the International Atomic Energy Agency (IAEA) concerning the review and commenting of the Preliminary Safety Analysis Report.

In sense of Section § 11/A (4) of the Atomic Act, the HAEA will hold public consultations in course of the implementation licensing procedure; the concerned parties (organisations involved in the licensing process) and the general public shall be informed of the venue and date of these public consultations at least 15 days prior to the consultations. During these public consultations, the concerned parties and all the interested entities may get acquainted with the subject and progress of the licensing procedure, may express their standpoints and put their questions.

The primary document of the implementation license application is the Preliminary Safety Analysis Report, the purpose of which is to confirm that the implemented nuclear power plant – by means of using technical and process solutions and operation modes presented in the design documentation - fulfils the specified nuclear safety requirements and can be constructed and operated safely.

The Preliminary Safety Analysis Report comprises the result of a several-year-long, complex activity. The first step consisted in the development of the design basis determining nuclear safety-related and other requirements, as well as the design criteria applicable for the nuclear power plant, its systems and components that should be taken into consideration during the designing process. After finalisation of the design basis, the preparatory work was continued to develop the Basic Design documentation. In addition to the Basic Design of plant buildings, systems and components, various analyses and evaluations had to be conducted in order to confirm the adequacy of technical solutions contained in the Basic Design documents. Based on the Basic Design and other related evaluations, the Preliminary Safety Analysis Report provides an overview of nuclear safety characteristics of the constructed nuclear power plant and contains an item-wise confirmation of fulfilling relevant nuclear safety requirements.

Herein below, you will find summary descriptions of the Preliminary Safety Analysis Report chapters.

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 1 – INTRODUCTION AND GENERAL DESCRIPTION OF THE NUCLEAR POWER PLANT

Introduction

The chapter contains general description of the nuclear power plant and its characteristics.

In addition, this chapter provides an overview of the full content of the Preliminary Safety Analysis Report, including the titles of chapters, sub-chapters and a complete list of abbreviations. Moreover, it refers to the documents specifying the requirements to the content and form of the Preliminary Safety Analysis Report.

General description of the nuclear power plant

Conditions of project implementation

Prior to the submission of implementation license application, the Licensee shall obtain the following licenses and permits:

- Environmental License pursuant to Act LIII of 1995 on the general rules of environmental protection
- Site License pursuant to the provisions of Act CXVI of 1996 on atomic energy and requirements of the Govt. Decree No. 118/2011. (VII.11.)
- Principal Plant License in accordance with the requirements of Act LXXXVI of 2007 on electricity

References to other chapters containing the description of the above licenses confirm the fulfilment of pre-conditions for the project implementation, i.e. the implementation process is in full compliance with the Hungarian legislation and regulatory requirements.

Connection to the national electric grid

One of the cornerstones in the operational safety of nuclear power plants is the creation of connection to the national electric grid. In addition to the assurance of reliable transmission of the electricity generated by the nuclear power plant, in certain cases, this grid connection can be also used for the external power supply to the house load systems. This subsection presents high-voltage electrical systems (400 and 132 kV) assuring the connection of the nuclear power plant to the grid and provides an overview of the 400/132 kV existing (Paks) and new (to be constructed in Biritó) transmission substations for the connection of the plant's high voltage systems.

Operating modes of the nuclear power plant units

Basically the nuclear power plant is designed for the continuous generation of electricity; however, based on the customer demands and taking into account renewable sources involved in electric power generation, the nuclear power plant is also able to operate in load-following mode.

Overall safety concept of the nuclear power plant

At the current stage of the unit designing process, this subsection – at the concept level –- presents a comprehensive protection plan against the hazards of natural origin. The detailed description of certain measures belonging to the concept is presented in other relevant chapters of the Preliminary Safety Analysis Report.

Environmental impacts of the nuclear power plant

This sub-chapter presents the environmental impacts of the nuclear power plant, taking into consideration the conditions specified in the environmental license.

Comparison with other facilities

The chapter compares main technical characteristics of the Paks units with other power unit types.

Information related to project implementation

Organisational structure of the project implementation with defined spheres of authorities and tasks

The sub-chapter presents organisations involved in the project implementation, describes the project management system implemented by the Licensee (Owner), Contractor and General Designer. This chapter shows the organisational structure applied for the project implementation, contains the description of functions performed by main organisational units and identification of key positions - these functions and structure will change and transform as the project progresses.

The sub-chapter provides information on the Licensee's preliminary organisational structure and operational processes typical for the project implementation phase. Main operational processes include licensing, quality management, design process review, implementation supervision, work safety and fire protection, quality control, program support, preliminary preparation for operation, human resources, legal support, financial issues, property and data security, physical protection and communication. In addition, this subsection contains links and references to the Preliminary Safety Analysis Report documents related to the implementation of the Licensee's construction organization.

As far as the Contractor and General Designer are concerned, the subsection describes their project management system related to project implementation, organisational structure, duty regulations, change management program concerning the organisation, as well as the Project Management Manual and connected documents of the Preliminary Safety Analysis Report.

Effects of the project implementation on nearby nuclear facilities

It is necessary that effects of the new nuclear power units on nearby nuclear facilities should be described in respect of Paks Nuclear Power Plant and Interim Spent Fuel Storage Facility. This subsection provides information on the effects and hazards potentially affecting the existing nuclear facilities (e.g. spread of dust, vibration, subsidence, effects of handling hazardous chemical substances), design parameters related to certain hazards, relevant design criteria, standards and computer codes, as well as the methodology of review and evaluation.

The subsection outlines general design measures and solutions aimed at maintaining negative effects of hazards within the limits of acceptance criteria, preventing any effects originating from the construction and operation of new power units from affecting the site of existing nuclear facilities.

Lessons learnt from earlier project implementation activities, and the safety non-conformance

This sub-chapter provides an overview of the methods intended for the use of operating experiences gained during the designing, construction and operation of nuclear power plants, as well as the general structure and operation of the information system for the collection and utilisation of operating experiences important in terms of the development of nuclear power plant designs. In addition to the overview of events effecting nuclear safety and identified during the construction and commissioning of projects implemented earlier, local cooperation experiences gained during the implementation of other nuclear power plant projects, this subsection describes the experiences related to the construction and commissioning of nuclear power plant units currently operating on the Paks site.

Identification of agents and contractors

The subsection lists the tasks and responsibilities of the companies, who, on one side, participated in the designing process and preparation of the design documents and on the other side, presents those participants who will be involved at later stages of the project implementation, like manufacturing, construction, erection and commissioning. This subsection includes a matrix illustrating interconnections between the activities performed by various companies.

Additional documents considered as part of the safety analysis report, references

This sub-chapter lists all the documents taken into consideration during the preparation of Preliminary Safety Analysis Report, references to which are contained in the report.

Unified Identification System

The chapter describes the unified identification system for nuclear power plant systems and components based on the German standard Kraftwerk-Kennzeichen System (KKS). This subsection provides the codification structure and main principles, as well as references to the procedure containing details and relevant annexes.

Drawings and other graphical information

The subsection lists drawings presented in the Preliminary Safety Analysis Report, including drawing numbers, descriptions, version numbers and dates. Furthermore, the subsection contains graphic presentation of the site arrangement.

Interfaces

The subsection provides a brief overview of the documents containing relevant information in terms of the Preliminary Safety Analysis Report that do not make a part of the implementation license application but – independent on this fact – have been prepared and approved. These are the Environmental License, Preliminary Safety Information, Site License and Basic Design.

Licensing of long lead equipment and priority buildings

The use of complicated manufacturing technologies and special materials is required for certain equipment of the nuclear power plant. A several-year long manufacturing process requires the application of a special licensing procedure and preparation for the procedure performance by the regulator. This subsection lists long lead equipment, the list of which had already been agreed upon by the regulatory authority.

In addition, this subsection contains the list of buildings agreed upon by the regulatory authority, the construction licensing and construction of which would be started immediately following the issue of the implementation license.

CHAPTER 2 – SITE CHARACTERISTICS

Regarding the site description, experts took into consideration the fact that site investigation and site-licensing procedures had already been completed. As an integral part of these procedures, the so-called Site Safety Report (SSR) was drawn up. Based on the site investigation and evaluation results, the SSR, on one side, confirmed the site suitability, approved by the HAEA through the issue of the site license. On the other side, the report showed site characteristics necessary for designing proper protection of the nuclear power plant against external hazards.

Complex investigations of the site suitability were completed by the preparation of the SSR report. Following the issue of the site license, further clarifications and modifications were made in respect of several elements of external hazards relevant to the plant design – partly due to the necessity of providing special design data in the scope differing from that of site investigations and partly due to the fact that final interpretation of hazard effects can be made in connection with the basic design.

As a part of chapter preparation, we examined and evaluated which data – out of those considered in the scope of SSR – should be updated. In general, it can be stated that updating of data was necessary only in a very limited amount of cases. It can be explained by the fact that a few years of new data related to the design basis events occurring once every 100,000 years cannot lead to significant differences in statistics. However, there are some characteristics, like nearby industrial facilities, in case of which the pace of changes justifies the performance of more frequent reviews.

In addition to the above-mentioned aspects, while preparing this chapter, we also took into account information received during the fulfilment of conditions specified in the site license in all cases relevant to the chapter content.

It should be mentioned that the evaluation of fulfilment in respect of some requirements could be done only in part or in a limited scope at the phase of SSR preparation since the knowledge of selected technology was necessary to evaluate the requirement fulfilment in full; this knowledge, however, was not available then (e.g. spatial separation of redundant trains of the safety systems). The adequacy of the plant systems' protection was evaluated during the preparation of PSAR and – in a great part during the Basic Design preparation, as well – based on the tests specified in the SSR and performed by the designer. Chapter 3 concerns issues related to the assurance of proper protection against potential external hazards. Chapters 2 and 3 contain the confirmation of fulfilling the requirements belonging to the scope of site investigation and evaluation and relevant at the phase of implementation license substantiation.

Geography and demography

The subsection focuses on the site location including site and nearby areas managed by the Licensee. The information received in respect of the activities carried out in the plant vicinity includes data on the population distribution and density, as well as on the location of state-owned and private facilities. This subsection describes demographic characteristics within a radius of 30 km required for the preparation of emergency plans pursuant to the legislative requirements.

Site-specific external man-made hazards

The subsection summarises site-specific external man-made hazards important to the safety of new power units, providing a summary description of input data, assumptions, methods and evaluation results used for the presentation of site characteristics connected with external man-made hazards.

The study of external hazards covered industrial, military, mining, transport and transportation activities, aviation, and other hazards such as upstream and downstream facilities, objects moved by wind, effects of potential fires and electromagnetic interference.

The subsection also contains the evaluation of co-occurrence of certain external hazards. As compared to the data contained in the SSR, no new facilities were identified within a 10 km radius that should be taken into consideration as potential hazard.

Meteorology

Based on the SSR, this subsection summarises site-specific meteorological characteristics, describes climate modelling and results thereof, provides information on the site relevant regional measurement program and its results; the subsection specifies meteorological characteristics affecting safety analysis, including the description of extreme weather conditions. These data are used as the basis for the calculations related to the spread of radioactive materials. Data required for climate analysis were provided by authentic data sets stored in the electronic database of the National Meteorological Service.

Hydrology

Site-specific hydrological characteristics were determined taking into consideration the information supplemented in the scope of site characteristics discussed in the SSR and finalised upon the completion of further examinations and evaluations specified in the SSR. For the determination of the Danube high water levels, the experts – in addition to the data used in the

SSR – relied on the information on the Danube water levels until 2018 inclusive. The evaluation of flash flood caused by extremely heavy precipitation started in the scope of the SSR was completed. The evaluation determined maximum water levels occurring based on the assumptions. For the determination of the Danube low water levels, the experts – in addition to the data used in the SSR – relied on the information on the Danube water levels until 2018 inclusive.

For the determination of the Danube extreme water temperatures, in addition to the information presented in the SSR, the experts determined the extreme Danube water temperatures with the frequency of occurrence of 100,000 years in respect of various durations. These extreme temperature values were finalised with consideration to the predictions outlined in the SSR. The subsection also deals with the issue of grainy materials that may get into the Danube water in the result of bunding collapse and landslides in the upstream stretch of the Danube potentially leading to equipment clogging.

Geological structure

The subsection describes geological structure in the order of structural development history moving from older formations to the younger ones dividing those into three large formation groups (Paleo-Mesozoic bedrock; Neogene formations; Quaternary formations). The description of geological structure includes a 30-km radius, within which targeted investigations were conducted to clarify geological conditions.

Basic geological and seismological information

The regional seismicity of the Pannonian Basin and the local seismicity of Paks were determined on the basis of historical earthquakes and current measurements, archival sources and data received with the help of micro seismic monitoring network operated by Paks Nuclear Power Plant and Paks II. Nuclear Power Plant Ltd. It can be stated that seismic activity in the vicinity of Paks is definitely lower than the average Hungarian value.

The experts determined site-specific seismic risks in three steps with the use of innovative scientific technologies:

1. determination of seismic risk in bedrock
2. determination of local modifying variable; determination of the surface seismic risk;
3. secondary effects resulting from earthquakes – determination of soil liquefaction risks

Geotechnical characteristics of the site

Previously, the experts had explored and determined the general geotechnical conditions of the area, as well as the main geotechnical characteristics of the soil layers within the framework of the Geological Research Program (GRP) forming the basis of the SSR. However, due to the difference of purposes pursued by the GRP, several issues remained unclarified at that stage (e.g. pre-load of the Pannonian strata, location of Pannonian strata in the southern part of the area). In order to clarify the soil stratification and soil characteristics required for the design, a large number of additional drillings, on-site geotechnical and geophysical surveys, and laboratory experiments were carried out in that area. Surveys and tests were performed in several stages. Firstly, geological hazards were identified, based on which it can be stated that:

- no problems related to the slope were recorded – the sited is designed for the level of 97 m above the sea level and the terrain surface is flat
- no signs of karst presence were detected; there are no cavities, caves, mines, cellars or any other unrecultivated structures of natural or artificial origin in this area
- the Danube banks are covered with forest and are in a stable condition; banks of the inlet channel are well-arranged; the inspection and maintenance are performed on a regular basis
- there are no formations of natural origin or man-made objects that could lead to the surface collapse, subsidence or rise within the site area
- due to the increase of groundwater levels in case of high water, technical protection should be assured for deep foundations
- there are no fragile, swelling, saline or eluvial soils in this area
- there are risks related to soil liquefaction on the site area; soils prone to liquefaction occur evenly throughout the site at a depth of 6–16.8 m - the effect of soil liquefaction can be eliminated by proven geotechnical methods, such as proper building foundation or soil stabilization

Other hazards

These hazards include all such impacts that may endanger the site of the new units by water, by road or by air (rodents, insects, leaves, etc.) or may impede the assurance of permanent cooling, in extreme cases making it impossible (mussels, snails, algae, microorganisms). No populations or natural conditions have been identified in case of the atmospheric or terrestrial invasion that would somehow endanger the facility. The risks posed by biological factors impeding the possibility of permanent cooling (e.g. mass fish mortality in the Danube, intrusion into the Danube, inlet channel or water intake structure of aquatic plants that may cause clogging of systems) do not affect the construction of new power units as the above risk factors can be properly managed during the construction.

Radiological impact

Data on the background radiation for Paks site and its vicinity were summarised based on the SSR and supplemented for the period of 2018-2018 with the measurement results received by the Regulatory Environmental Radiation Monitoring System (HAKSER), National Environmental Radiation Monitoring System (OKSER) and Operational Environmental Radiation Monitoring System (ÜKSER) in respect of the following environmental components:

- gamma-radiation exposure dose rate
- aerosols and fall-out
- surface water (the Danube, fish-ponds and canals) and sediments
- fish
- groundwater
- soil-, grass- and feed samples
- cow milk

Data on background radiation presented in the SSR were supplemented with the results of special site investigations conducted in 2015, in respect of the following environmental characteristics and components:

- radon exhalation
- gamma dose rate
- deep soil samples
- planned inlet and outlet channels

The study summarised the data and characteristics effecting the spread of radioactive contamination and public radiation exposure.

Monitoring program

Pursuant to the recommendations provided in N3a.34 Guide of the Hungarian Atomic Energy Authority, this subsection provides an overview of measures implemented for the purpose of site monitoring. The monitoring program focuses on the identification and examination of parameters that can be affected by seismic, atmospheric, aquatic and groundwater factors and impacts caused by industrial facilities and transport. The collected data are used to prepare the nuclear emergency response plan, to substantiate findings made during periodical safety reviews, to develop the model for the spread of radioactive contamination and to evaluate the completeness of the list of site-specific hazards. The long-term monitoring program collects data measured and recorded both by on-site measuring devices and provided by specialised national institutions, in order to identify essential deviations from the design basis. This subsection contains information on the methods used for the documentation and storage of data collected in the course of monitoring activities.

CHAPTER 3 – DESIGN OF SYSTEMS AND SYSTEM COMPONENTS

The chapter summarises all potential impacts and stresses taken into consideration during the plant designing process and describes how they were managed in the Basic Design. The complex of static and dynamic loads with different magnitudes and frequencies forming the basis of the Basic Design is a part of the design basis. These effects may originate during the normal operation of the nuclear power plant, failures of certain structural elements or may be the consequence of natural hazards, e.g. earthquake.

One of the fundamental principles of the nuclear power plant design is the defense-in-depth principle. The defense-in-depth principle is based on a multi-level system of measures and technical decisions aimed at preventing, handling and minimising the consequences of accident situations. Events occurring during the plant lifetime due to the potential failures of systems and system components caused by external or internal hazards or being the result of human errors are categorised based on the event severity and occurrence frequency; these categories are linked to different levels of defense-in-depth. Respective actions and interventions are determined with respect to these levels. This is the result of an iteration process, during which the design is reworked until the results of safety analyses demonstrate full compliance with the compliance criteria stipulated by the Nuclear Safety Code.

In view of the above-said, Chapter 3 is focused on the presentation of the plant design basis, identification of systems designed for managing certain events and confirmation of the fulfilment of requirements by the given system on the basis of safety analysis, paying special attention to the fulfilment of legislative requirements, in particular in respect of the environmental and public radiation exposure. All these aspects are discussed in detail in other chapters of the Preliminary Safety Analysis Report. Site characteristics, being sources of external hazards, are shown in Chapter 2. The detailed technology description is provided in Chapters 4–10. Chapter 15 contains the results of deterministic analysis, while Chapter 19 presents those of probabilistic safety assessment. Chapter 3 provides us with an opportunity to have a global insight into the plant design process and design compliance.

Fulfilment of requirements

This subsection provides information on the fulfilment of Nuclear Safety Code requirements relevant in terms of the implementation license application, determined in accordance with the HAEA Guide N3a.34 Safety reports of new nuclear power plants, based on the level of detail applicable at the current design stage.

Application of design and analysis tools

This sub-chapter is aimed at demonstrating the applied design and analysis tools, as well as at confirming the compliance of input data used for calculations. This compliance is examined from two different aspects: validation and verification. In terms of a program or a model, validation means that the physical quantities calculated by the program match the measured results within a given uncertainty. The comparison with experimental results can be done either directly or indirectly, for example by means of comparing with the results of such a program that has already been validated for the experimental data. Verification confirms that certain models operate as appropriate, and data management modules perform their tasks contained in the code description. All software programs used for safety analysis are verified as soon as they are purchased or at the time of a new version release. Control of compliance is conducted also by a special independent team or and expert, in accordance with legislative requirements.

In addition to this chapter, relevant supporting documents are aimed at confirming the compliance of applied design and analysis tools with the above-mentioned aspects.

Categorisation of plant conditions and events

As regards nuclear safety, special importance is attributed to operational occurrences resulting from the so-called postulated initiating events and reflecting some failure or abnormal operation of the nuclear power plant. Identification of postulated initiating events forms a part of the nuclear power plant design basis; these initiating events are categorised based on the frequencies of occurrence. This subsection contains the list of postulated initiating events with respective frequencies of occurrence; the identified initiating events are consistent with Chapter 15. Furthermore, summary evaluation of external and internal hazards is provided in this sub-chapter.

Safety functions

This subsection identifies and provides a general overview of safety-related features, assigning their performance to the respective levels of defence-in-depth. In regard of each safety function, this subsection contains a summary description specifying systems performing the given function and indicating the method of function performance.

Safety systems descriptions

This subsection provides a brief general description of engineered safety systems intended for handling postulated initiating events; the detailed description of safety systems is available in system-related chapters of the Preliminary Safety Analysis Report.

Classification

One of the most important tools of nuclear safety assurance is the system of safety and environmental classifications. The applied classification concept ensures that the importance of nuclear power plant systems, system elements and structures to nuclear safety is determined in a systematic way. The practical purpose of this procedure is to ensure that classified equipment and structures will be designed, manufactured, commissioned, operated and maintained in accordance with uniform technical requirements differentiated pursuant to their impact on safety. As regards the current phase of the unit designing process, this chapter presents the applied classification methodology, as well as safety classes and environmental qualification of key systems, system elements and buildings of the nuclear power plant.

Internal and external hazards

With respect to the current phase of the unit designing process, the chapter summarises external (e.g. earthquake, heavy aircraft crash) and internal (internal fire, explosion, flooding, etc.) hazards taken into consideration during the designing process, design parameters of certain hazards, relevant design criteria, standards, computer codes used for the analysis, review and evaluation methodology.

In addition, this chapter presents general design measures and solutions providing sufficient protection against the harmful effects of external and internal hazards taken into consideration during the design of systems and system components important to nuclear safety.

Strength calculations

With respect to the current phase of the unit designing process, the chapter provides the description of codes and standards containing the requirements to system components, girders and supports including calculation and acceptance criteria, designed transients with the analysis of typical loads and load combinations, static and dynamic loads and structural integrity, as well as software and codes used for analysis performance. General description of the leak-before-break concept is provided here as well.

Materials

With respect to the current phase of the unit designing process, the chapter provides a general overview of structural materials used for the system elements, general methodology for their selection and applied standards, including metallic, non-metallic and concrete structural materials. Most important physical, chemical, corrosion, mechanical and other material properties necessary for the designing process, as well as ageing and degradation characteristics are shown herein. The chapter outlines manufacturing and inspection processes related to structural materials, welding, heat-treatment, material testing and associated certifications.

Maintenance, testing, inspection and ageing management

At the current phase of the unit designing process, the applicable maintenance strategic elements are presented, which basically consist in preventive maintenance and repair. Preventive maintenance consists of the time-based and condition-based maintenance. At the following stage, all system components – independent on their safety classification – will be subjected to the reliability centred maintenance analysis (RCM) and as a result thereof will become a part of the given maintenance strategy.

The subsection includes the description of general test and inspection methods, listing the tests preliminarily calculated at the current phase. These will be finalised on the basis of RCM analysis.

The subsection outlines the systems of continuous and periodical monitoring, a part of which will be used during the condition-based maintenance.

The chapter concerns the preliminary ageing management methodology developed in respect of passive system elements. This methodology includes degradations that may affect certain elements in different operating states, with the indication of phenomena causing these degradations. The subsection defines basic design principles required for the prevention of the above phenomena. Furthermore, the document touches upon ageing management requirements related to the later stages, i.e. manufacturing, installation and operation.

Electrical systems and components

The chapter provides a summary description of the structure and operation of the plant's house load electrical systems. In addition, this chapter focuses on the requirements to house load electrical systems determined in the Nuclear Safety Code and demonstration of their fulfilment. Further detailed descriptions of certain electrical systems are available in Chapter 8.

Instrumentation and control systems and components

The chapter describes general principles of instrumentation and control system design, including but not limited to the defense-in-depth, human-machine interface, single failure criterion, redundancy, diversity, separation, testability and cyber security. System descriptions provided in Chapter 7 were prepared with respect to these design principles.

Buildings and civil engineering structures

The chapter focuses on the general principles of architectural design, based primarily on national norms and regulations. In addition, in this chapter one may find specific information and design principles related to building foundations, containment (reinforced concrete structure used for the placement of nuclear reactor and connected systems) and buildings located nearby the containment.

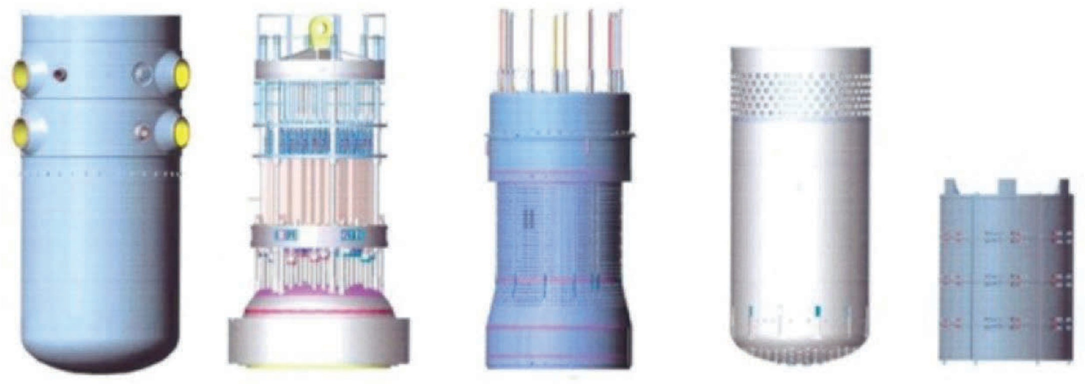
CHAPTER 4 – REACTOR

General description

The generation of heat in a nuclear power plant takes place in the reactor core as a result of controlled chain reaction. The process is based on the splitting of heavy uranium nuclei leading to energy release. Neutrons appearing during splitting are used for fission. The generated heat is removed from the reactor core with the use of water medium functioning as coolant. An important element of the above process is the assurance of control by means of control rod movements and dilution of boric acid – characterised by neutron absorbing properties – in water.

Chapter 4 describes design solutions applied for the uranium-based nuclear fuel functioning as energy source, absorber rods participating in the reactivity control and coolant circulation assuring the removal of generated heat. A scientific breakthrough of the modern age was to harness the enormous inherent energy hidden in nuclei to the betterment of mankind. This chapter provides information on the fuel rods containing nuclear fuel, fuel bundles made of fuel rods and structural components used in fuel bundles. In addition to the description of design elements, the chapter evaluates neutron physics and thermal hydraulics characteristics of the reactor core, confirming the compliance of design with the applicable requirements.

The chapter describes the reactor pressure vessel accommodating the reactor core and reactor internals intended for the arrangement of reactor core made up of fuel bundles and regulation of coolant circulation in the reactor vessel. Relevant details are provided in Chapter 5.



Reactor

Fuel system

One of the important tasks of this chapter is demonstration of safe operation of the fuel system. The fuel system consists of fuel assemblies, fuel rods arranged into fuel assemblies, absorber rods and respective structural elements.

Fuel rods are made up of fuel pellets and fuel rod claddings. Fuel rods contain two types of fuel pellets. One of them is presented by uranium dioxide pellets, the other one by fuel pellets manufactured from uranium dioxide and gadolinium mixture. Fuel rods are arranged in hexagonally shaped fuel bundles. In addition to fuel rods, the fuel system includes so-called absorber rods. The task of these absorber rods is to assure the control of nuclear fission reaction in the core. The material of absorber rods is boron carbide and dysprosium titanate with good neutron absorbing properties.

Safety of the reactor core is assured by the integrity of so-called physical barriers. Physical barriers are aimed at preventing the escape of fission products from the fuel rods. These physical barriers include fuel matrix (pellet), fuel rod cladding, reactor coolant system boundary and containment. The first two ones belong to the fuel system.

The chapter confirms the compliance of fuel bundles, reactor core and structural components thereof to the requirements specified in the Nuclear Safety Code. The fuel system was designed with the use of validated and verified design tools and software codes. Properties of structural materials applied in the fuel system were tested and substantiated with the use of an experimental reactor; in addition, several decades of VVER reactor operating experiences were used in course of the design process.

Nuclear design

Fuel assemblies loaded into the reactor core according to the given loading pattern are called fuel load. The unit campaign shall be understood as the period until the next re-arrangement of fuel assemblies in the core, viz. the removal of spent fuel assemblies and loading of fresh ones. The isotope composition of the reactor core changes throughout the campaign. When designing the core reload, nuclear characteristics of the reactor core for the given reload, power density distribution and its time-wise changes should be taken into consideration for calculations.

Neutron physics characteristics of core reloads shall meet various requirements. Along with the load capacity of the reactor vessel and reactor internals we should take into account the controllability of neutron physics and thermal hydraulic processes in the reactor core and fuel behaviour.

Effects emerging in the reactor and reactor core during operation are analysed in order to determine frame parameters of reactor physics, viz. to define minimum and / or maximum initial values of the quantities substantive for the analysis results. The purpose pursued by the determination of reactor physics frame parameters is to ensure that calculations do not have to be repeated every time prior to each campaign and will remain valid for several subsequent reloads in the future. Compliance of the newly re-arranged core with reactor physics frame parameters is confirmed prior to the given campaign by means of core reload design calculations.

Thus, based on the above principles we do not only assure and check the operational compliance of those core reloads, which will be loaded into the reactor at the beginning of the plant lifetime but also specify the limits assuring sufficient margins for the fuel operation.

Thermal and hydraulic design

Energy released during the fission of U-235 atomic nuclei heat up the fuel matrix. By means of thermal conductivity, this energy is transferred from the fuel to the fuel cladding containing the fuel. After that this energy - via heat transfer - comes into the coolant circulating around the fuel assembly. The coolant - i.e. water in our case - transports this thermal energy released by the fuel from the reactor to the steam generator to produce steam on the steam generator secondary side. Water circulation through the reactor is provided by four high capacity pumps. The reactor, steam generator and connected pipelines form the so-called reactor coolant system. The primary coolant water circulates under high pressure, so there is no boiling even if the coolant temperature rises up to 300° C.

This subsection describes thermal hydraulics - i.e. thermal and flow characteristics of the reactor under normal operating conditions. The subsection demonstrates that the reactor cooling can be assured safely and reliably - in other words, fuel assemblies containing nuclear fuel will not be damaged and radioactive material will not escape into the environment. It is shown that under normal operating conditions the released heat is safely removed and transferred from the reactor to the secondary circuit.

Reactivity control

This subsection is focused on the reactivity control. The design basis of nuclear reactors assures the performance of three fundamental safety functions:

a) reactivity control
b) removal of heat from irradiated fuel assemblies
c) retention of radioactive materials

Safety-related reactivity control function ensures the controlled chain reaction and allows to keep reactor power at the controlled level. The control of core reactivity in VVER-1200 reactors can be performed by two diverse - i.e. based on different operating principles - methods. These two methods include the insertion of absorber rods into the core, placed above the reactor core and containing neutron absorbing materials (boron and dysprosium) and injection of concentrated boric acid solution into the coolant. The above functions assure well-controlled and safe reactor shutdown.

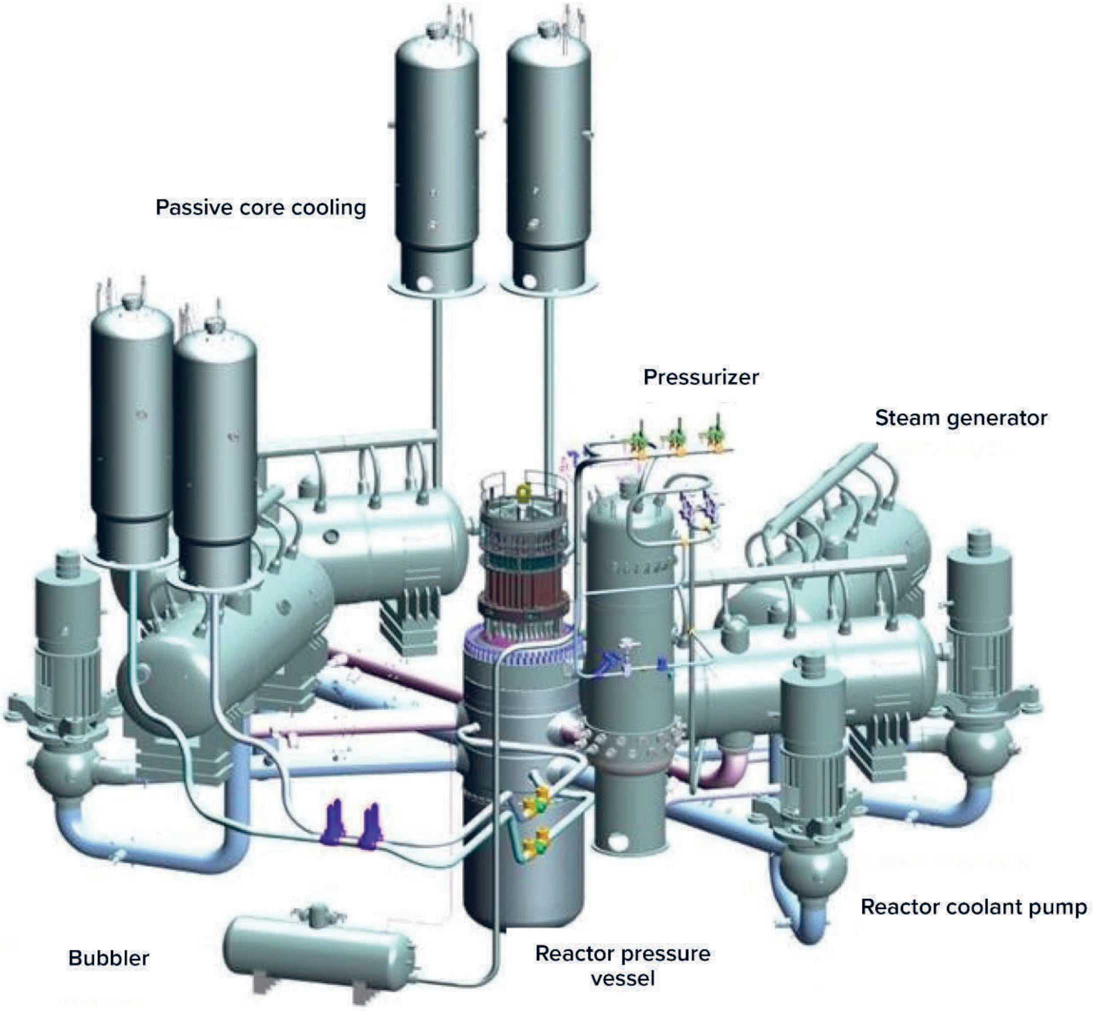
An important part of the reactivity control function is the use of reactivity coefficients (coolant-, moderator- and fuel temperature reactivity coefficients) depending on the reactor design and reactor core composition. VVER-1200 reactors are inherently safe, which means that the above-mentioned reactivity coefficients have an impact on safety, i.e. they have a negative effect in case of power increase (reducing the rate of power increase).

This subsection provides the description of design basis, as well as the characteristics and operation of systems involved into reactivity control.

CHAPTER 5 – REACTOR COOLANT AND CONNECTED SYSTEMS

The main tasks performed by the reactor coolant system are removal and transfer of heat released in the process of nuclear fission to the secondary circuit, maintenance of operating pressure and temperature, as well as the prevention of coolant escape into the containment. In case of VVER-1200 power units, the main reactor coolant equipment is as follows:

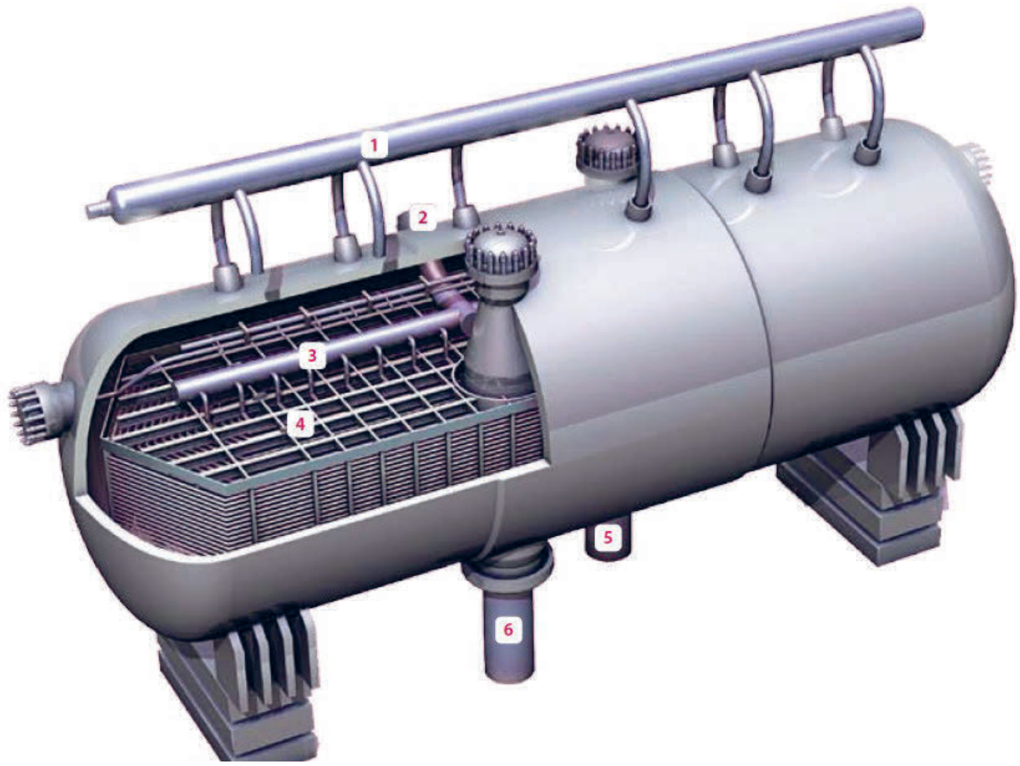
- reactor
- 4 reactor coolant circulation loops
- 4 steam generators
- 4 reactor coolant pumps
- pipelines connecting the loop equipment with the reactor
- reactor pressure control system
- control rods for reactor core reactivity management



Reactor coolant and connected systems

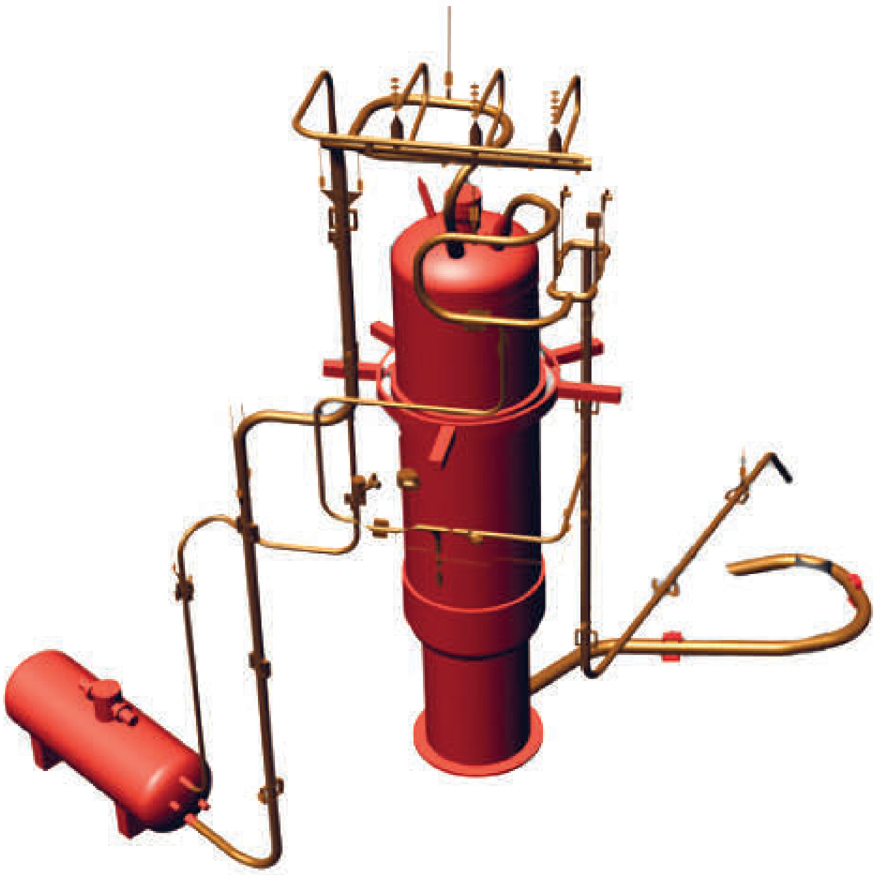
Each reactor loop is connected to the horizontally arranged steam generator via cold- and hot leg pipelines; 10858 horizontal heat-exchanging tubes transfer the heat removed from the reactor to the secondary side coolant. Cooling water supplied to the steam generator secondary side heats up and generates high temperature and high pressure steam that is then transferred towards the turbines via the steam header. At an operating unit, each steam generator produces 1600 tons of steam per hour.

Steam generator



1. Steam generator main steam header
2. Feedwater supply
3. Feedwater distribution unit
4. Heat-exchanging tubes
5. Hot leg collector
6. Cold leg collector

An important element of the reactor coolant system is a 55 m^3 reactor pressure control system – the so called pressurizer – being an essential means of the reactor pressure control in pressurised water reactors.

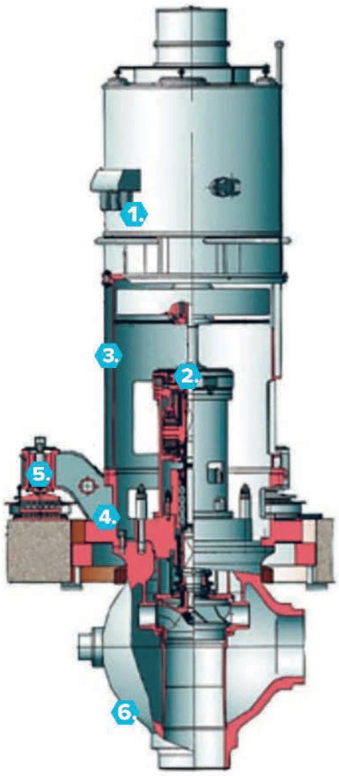


Reactor pressure control system

Coolant circulation in four reactor loops is assured by four reactor coolant pumps, one of which is installed in each loop. No oil is required for the operation of VVER-1200 reactor coolant pumps - lubrication and cooling are assured by water.

Reactor coolant pump

1. electric motor
2. pump shaft bushing
3. upper part
4. lower part
5. support structure
6. pump housing



The reactor coolant system of VVER-1200 power units is located in a special protective building (containment). In order to minimise and prevent damages caused by internal hazards, the equipment and pipelines of reactor coolant system are physically separated from each other.

Equipment belonging to the reactor coolant system is designed for operation under normal operating conditions, including various power levels, load changes, refuelling, shutdown for outage and transients connected with the testing of equipment and systems.

In case of abnormal or accident situations, the system - depending on the situation severity - limits the reactor power or trips the reactor with the help of control rods.

Fundamental safety functions performed by the reactor coolant system are:

- core reactivity control
- heat removal from fuel assemblies
- localisation of radioactive materials

Non safety-related functions performed by the reactor coolant system are:

- reactor start-up and heat up
- continuous operation at the given power level, management of transients necessary for power level changes
- reactor shutdown for outage or refuelling

Reactor coolant system and connected system valves

Along with main equipment belonging to the reactor coolant system, like reactor and reactor internals, reactor coolant pumps, steam generators, primary pipelines and reactor pressure control system, this subsection focuses on auxiliary systems connected with the reactor coolant system and required for its operation.

The most important ones are:

- make-up water and boron control systems
- residual heat removal system
- primary coolant purification system
- primary controlled leakage system

The task of make-up water and boron control system is to maintain water balance in the reactor coolant system in all operating states, to maintain necessary boric acid concentrations and change them as appropriate, as well as to control specified water chemistry parameters of the coolant.

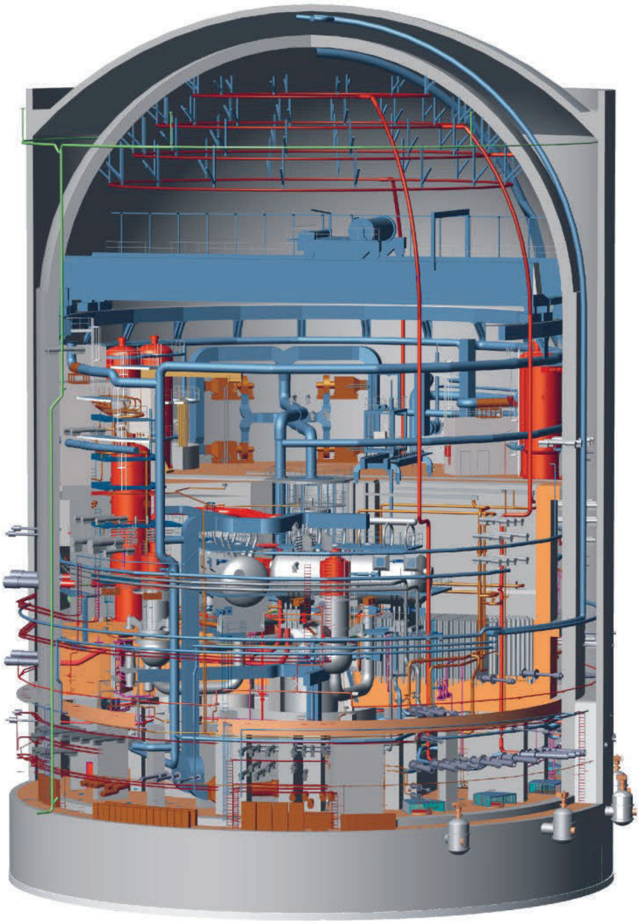
The task of residual heat removal system is to assure cooling of the reactor unit and removal of residual heat released in the core during the reactor shutdown.

The task of primary coolant purification system is to remove dissolved additives, corrosion and fission products from the primary coolant, assuring coolant purity.

The task of primary controlled leakage system is to check the integrity of joint seals on the main equipment and to control leakage on priority valves important to safety. The above system is responsible for collecting leakages and drains and transferring them into a control tank for the further re-use of these drains.

CHAPTER 6 – SAFETY SYSTEMS

The reactor coolant system of VVER-1200 power units is located in a special protective building made of reinforced concrete (containment). Secondary containment with the diameter of 50 m protects the equipment against external hazards; the primary cylindrically shaped containment with the inner diameter of 44 m is closed by dome on its top. The cylindrical part is 44,6 m high. Wall thickness in the cylindrical part is 1,2 m, while the containment dome is 1 m thick. Containment walls made of pre-stressed reinforced concrete are covered with a 6 mm thick steel lining.



Primary and secondary containment

The primary containment hermetically isolates the primary circuit containing radioactive materials from the environment; ventilation and exhaust from the space located between the primary and secondary wall is assured via filters. Doors leading to the containment function as a lock and are hermetically closed.

Safety systems actuated in case of certain accidents are located outside the containment in the safety building; their primary task consists of assuring core cooling. The design of VVER-1200 power units implies the use of several active safety systems requiring electric power supply. Most of these systems are comprised of four independent trains, physically separated from each other. It should be emphasised that in most cases, even one of the four trains is able to perform system safety functions in a full scope.

The most important active safety systems are:

- high pressure safety injection system
- low pressure safety injection system
- borated water storage system
- emergency feedwater system
- emergency boron injection system
- emergency gas removal system
- spray system
- emergency pressure reduction system
- reactor emergency cool down system

In case of accidents, electric power supply to active safety systems is ensured by four diesel generators, one diesel generator for each safety train.

In addition to active systems, VVER-1200 reactors are also equipped with several passive safety systems. A common feature of these systems is that they do not require human intervention and external power supply sources for their operation; performance of their functions is ensured by simple physics processes. Should the operation of active systems fail, passive systems come into action to ensure emergency core cooling and containment integrity. The passive part of emergency core cooling systems is designed for the removal of residual heat from the reactor core. Under the effect of gravity and nitrogen blanket located above borated water in 4 hydroaccumulators belonging to this system, borated water will flow into the reactor as soon as the reactor pressure becomes lower than that of hydroaccumulators.

The most important passive safety systems are:

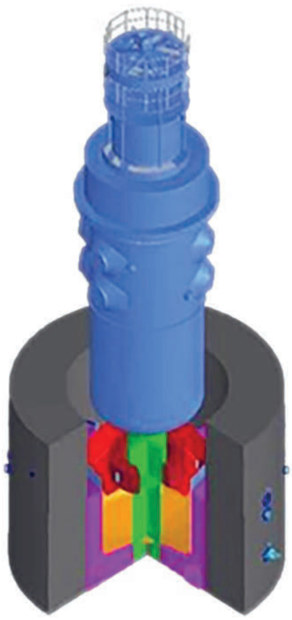
- emergency core cooling system passive part
- steam generator passive heat removal system
- containment passive heat removal system
- containment hydrogen removal system
- core catcher

There are two passive systems for residual heat removal designed for severe accident management. One of them removes heat from steam generators, while the other one assures heat removal from the containment - the flow during their operation is assured by natural circulation. Steam generator passive heat removal may become necessary in case of unavailability of the active cooling systems. In case of unavailability of the active sprinkler (spray water) system designed for the cooling of containment space, passive heat removal systems will ensure that the internal pressure in the protective building will not reach the value that could endanger the building integrity.

These passive systems have been proven to be able to prevent core damage during at least 72 hours without any external intervention.

After potential core damage, hydrogen appearing in the result of zirconium-water steam reaction would endanger the containment integrity. Passive autocatalytic recombiners installed inside the primary containment of VVER-1200 power units will prevent the development of explosion-hazardous states.

Core catcher is designed for mitigating the consequences of the core meltdown. Core catcher is a special device located beneath the reactor pressure vessel and designed for receiving corium in case of core damage and potential subsequent damage of the reactor pressure vessel. The device is filled with aluminium- and iron oxide charge able to mix with the core melt. As a consequence, the material properties of the corium change, the core melt dilutes resulting in the decrease of residual heat generation per volume unit. The core catcher charge contains gadolinium absorbing neutrons and assuring the corium sub-criticality. The steel tank of the core catcher can be cooled by water from outside. This so-called dry catcher prevents interaction between the corium and concrete foundation slab. The use of core catchers allows to reduce hydrogen generation and escape of radioactive fission products from the core debris.



Reactor and core catcher

CHAPTER 7 – INSTRUMENTATION AND CONTROL

The description of instrumentation and control systems is provided in Chapter 7; however, this chapter is closely connected with Chapter 3. One of subsections in Chapter 3 summarises design principles for instrumentation and control in the form of a catalogue. Such a structure provides the possibility for involving the given applied design principles from Chapter 3 in subsection of Chapter 7 dealing with system descriptions.

I&C system architecture, functional allocation

This chapter focuses on the overall architecture, summary of safety classes of certain instrumentation and control subsystems, as well as on the allocation of instrumentation and control functions and certain instrumentation and control subsystems.

This part contains summary descriptions of type solutions applied for subsystems, with the following breakdown:

- type solutions for measurements and primary signal processing
- main solutions for mechanisms control and regulation
- type solutions for system and equipment protection
- organisation of external and internal interfaces
- type solutions for electric power supply, grounding and interface immunity of I&C hardware

Safety I&C systems

The subsection describes instrumentation and control systems belonging to the highest safety class - safety class 2 - with the following breakdown:

- reactor trip system and engineered safety feature actuation system
- reactor trip breakers
- priority logic equipment
- neutron flux monitoring system
- seismic protection system
- emergency power supply system
- instrumentation and control system of diesel generators
- equipment of the FFSF transport gateway (access gates and FFSF hatch), package I&C

Safety-related I&C systems

The subsection describes instrumentation and control systems belonging to safety class 3 with the following breakdown:

- Reactor Limitation System
- Reactor Rod Control System
- Safety-Related Process Automation System
- In-core instrumentation system
- Seismic activity registration system
- Diverse protection system
- Severe accident management system
- Reactor vessel level indication system
- Primary Circuit Integrated Diagnostic System
- Primary Circuit Leakage Detection System
- Boric Acid Concentration Monitoring System
- Hydrogen Monitoring System
- Fire Protection Instrumentation and Control System
- Fuel Transportation Instrumentation and Control System
- Personnel Lock Instrumentation and Control System
- Electric bridge polar crane I&C
- Trestle crane I&C
- Instrumentation and Control System of Safety-related normal operation reliable power supply system
- Instrumentation and Control System of Reliable power supply system DEC1
- Instrumentation and Control System of Reliable power supply system DEC2

- Instrumentation and Control System of Safety-related heating, ventilating and air conditioning systems
- Instrumentation and Control System of refuelling machine
- Instrumentation and Control System of the crane of the outdoor refuelling station
- Instrumentation and Control System of electric cranes belonging to the fresh- and spent fuel storages, radioactive waste storage

Instrumentation and Control Systems not related to safety

This subsection describes systems not related to safety. Although these systems are not subject to licensing, it is still necessary to confirm that they do not have adverse effect on the operation of systems categorised into higher safety classes. The subsection presents these systems with the following breakdown:

- Reactor Power Control System
- Process Automation System Not Related to Safety
- Instrumentation and Control System of Turbine Plant
- Power unit electrical equipment data acquisition system
- Reactor Vibration Monitoring System
- Loose parts detection system
- Fatigue Monitoring System
- Rotary Equipment Vibration Diagnostic System
- Water Chemistry Instrumentation and Control System
- Instrumentation and Control System of Generator Auxiliary Systems
- Instrumentation and Control System of electric cranes belonging to the turbine building, fresh- and spent fuel storages, radioactive waste storage
- Instrumentation and Control System of Solid Radioactive Waste Treatment and Storage
- Failed Fuel Detection System
- Instrumentation and Control System of Fuel Inspection and Repair Bench
- Turbine supervisory instrumentation controller

The subsection confirms that systems not related to safety do not have any safety-reducing effect on systems important to safety.

Supervision level systems

The subsection describes supervision level systems with the following breakdown:

- Unit Upper Level System
- Mosaic panels and boards
- Large Screen Display
- Operator Support System

Control rooms and posts

The subsection describes different types of control rooms with the following breakdown:

- Main Control Room
- Emergency Control Room
- Technical Support Room
- Local Control Posts

CHAPTER 8 – ELECTRIC POWER

The chapter provides details related to the plant's electric power supply systems. These systems and equipment can be divided into two large groups: equipment necessary for the plant's normal operation and equipment for mitigating the consequences of accident situations.

The equipment designed for normal power supply assures electric power supply to the consumers necessary for the plant's normal operation (e.g. pumps, special plant equipment, computers, lighting, etc.) and transmits generated electricity to the power grid.

The equipment designed for managing accidents and events (emergency systems) assures electric power supply to the consumers necessary for the safe reactor shutdown in case of events and accidents at the nuclear power plant. These emergency power supply systems and equipment include, for example, diesel generators and accumulator batteries assuring the operation of essential consumers in the case of loss of offsite power. Each type of safety equipment is comprised of several safety trains that will be installed on the principle of physical separation from each other. Thus, it can be guaranteed that even in case of unavailability of several safety trains, the other similar equipment will continue supplying electric power to essential consumers. 10 diesel generators are installed at each of the nuclear power units to support the most important process systems and equipment. In contrast to general industrial practices, more stringent technical specifications and quality requirements were determined for these types of equipment. For instance, these requirements include significant safety margins and robustness (e.g. climatic and seismic resistance), additional requirements of special international guidelines and standards accepted in the nuclear industry (e.g. procurement and product purchase only from the suppliers possessing international nuclear experiences and special qualifications, from the companies recognised in the nuclear industry). Application of these requirements guarantees that reliable power supply will be assured for the essential consumers even in the case of events and accidents with very low probability.

Cables are of priority importance in terms of power supply since cable failures lead to unavailability of the equipment on the whole. In order to exclude cable failures, the respective requirements to cables are as stringent as those to the equipment. The chapter describes cables connecting various pieces of equipment, paying special attention to the rules of safe cable arrangement. The cables are also categorised as those for normal continuous operation and those participating in the accident management.

In addition, the chapter provides information on the lightning and over-voltage protection for the entire power plant, which is also designed in accordance with more stringent technical and quality requirements, as compared to the general industrial practice. The task of these lightning and over-voltage protection devices is to prevent damage to the equipment due to atmospheric and other voltage surges or electromagnetic interference.

CHAPTER 9 – AUXILIARY SYSTEMS AND CIVIL STRUCTURES

CHAPTER 9.A.1 – FUEL STORAGE AND HANDLING SYSTEMS

This system is intended for use during the replacement of spent fuel assemblies and absorber rods of the Reactor Control and Protection System with fresh ones.

Under normal operating conditions, fuel storage and handling system is responsible for the performance of the following functions:

- loading of fresh fuel assemblies and absorber rods from fuel transport containers into the reactor or spent fuel pool
- transportation and loading of leak-tight bottles (used for the storage of leaking fuel assemblies) from leak-tight bottles installed in the reactor cavity into the spent fuel pool
- unloading of fresh fuel assemblies and absorber rods from the reactor into the spent fuel pool
- re-arrangement of fuel assemblies and absorber rods in the reactor core
- loading of fresh fuel assemblies and absorber rods from the spent fuel pool into the reactor
- leak-testing of spent fuel assemblies prior to refuelling: first, in the refuelling machine telescopic cylinder and then in the failed fuel detection system
- repair of leaking fuel assemblies on the fuel assembly inspection and repair bench
- reloading of spent fuel assemblies from the spent fuel pool into the transportation package sets

Leak-detection system of the spent fuel pool

The system is designed for the leak-testing of the spent fuel pool, refuelling pool and lining of the pit used for the inspection of reactors internals. The system detects leaks, checks their volume and removes water detected under the lining, to the water treatment plant.

Failed fuel detection system

The system is used for the identification of supposedly leaking fuel rods (on a shutdown reactor) and includes the following types of testing:

- leak testing of fuel rods with the use of refuelling machine telescopic cylinder
- failed fuel detection system
- inspection and repair of leaking fuel assemblies

On-site fuel transportation system

The system is used for the on-site transportation of nuclear fuel assemblies. The nuclear power plant design includes the outdoor fuel handling station for fuel acceptance and transportation, equipped with all necessary devices.

CHAPTER 9.A.2 – WATER SYSTEMS

Essential service water system

The task of the essential service water system (ESWS) is to assure cooling water supply for the primary circuit and other priority technologies and removal of waste heat to the ultimate heat sink. Similarly to the operating nuclear power plant units in Paks, the ESWS system of new power units will be based on the freshwater technology; however, in case of loss of the primary ultimate heat sink – River Danube – or loss of power in the same sub-system, there is a possibility of using an alternative cooling cell-based sub-system.

The essential service water system removes heat from the primary consumers via the intermediate cooling circuit for essential consumers. Pressure in the systems - depending on the operating state of the nuclear power plant - functions as a physical barrier preventing both the contamination of the process cooling loops with raw water and escape of potential radioactive contamination into the environment. Pursuant to the requirements specified for nuclear power plants with pressurised water reactors of generation 3+, N+2 redundancy level of the ESWS system matches the 4x50% safety system configuration of the new power units based on the design basis. It means that safety systems are capable of performing their functions if one of the safety trains is unavailable due to maintenance and the other one fails at the very same moment.

Intermediate cooling circuit for essential consumers

The primary task of systems described in this chapter is to assure cooling of the primary consumers. These systems form an intermediate cooling circuit between the essential service water system and primary consumers and as such, function as the physical barrier between presumably radioactive media and essential service water, preventing the escape of radioactive medium into the essential service water system.

Demineralised water supply system

The task of this system is to assure primary circuit consumers and diesel generator systems with demineralised water in the reactor building, safety building, nuclear technology service building, diesel generator buildings and auxiliary building.

Ultimate heat sink

In terms of the nuclear power plant, ultimate heat sink should be understood as all such media and any related system absorbing waste heat (in the view of nuclear safety, first of all, decay heat) originating from various sources. Pressurised water reactors belonging to 3+ generation are expected to have – in addition to the primary ultimate heat sink – an alternative solution allowing to perform the above task.

Similarly to the operating VVER-440 units of Paks Nuclear Power Plant, the primary ultimate heat sink in case of new VVER-1200 units is the Danube; however, in case of ultimate heat sink loss, safe shutdown, transition and maintenance of the nuclear power plant in the safe controlled condition is also assured by several other technologies. The table below includes – but is not limited to – the most important systems, functions and necessity of their operation in different operating conditions of the nuclear power plant. As one can see from this table, several diverse cooling solutions are available in different operating states of the nuclear power plant to assure residual heat removal and to prevent core meltdown.

System	Function	Ultimate heat sink	Plant operating conditions				
			Normal operation	Anticipated operational occurrences	Design basis accidents	Complex accidents	Severe accident
Main circulating water system	condenser cooling: support of normal generation	Danube (primary)	X				
Auxiliary cooling water system	cooling of secondary auxiliary consumers: support of normal generation	Danube (primary)	X				
Essential service water system (primary)	cooling of primary and essential consumers / residual heat removal via normal route: normal operation and accident	Danube (primary)	X	X	X		
Essential service water system (redundant)	residual heat removal via normal route: certain complex accidents	air (alternative)				X	
Essential service water system (redundant)	residual heat removal via redundant route: certain complex accidents	air (alternative)				X	
Steam generator passive heat removal system	passive residual heat removal: certain complex accidents	air (alternative)				X	
Containment passive heat removal system	passive residual heat removal: severe accidents	air (alternative)					X

Make-up water system

The task of make-up water system is to assure storage and supply of demineralised water for the operation, filling and, whenever necessary, replenishment of primary and secondary circuit systems and auxiliary equipment. The make-up water system is located in the technology service building. The system consists of 5 independent demineralised water tanks with the volume of 1000 m^3, two normal operation and two emergency demineralised water pumps.

Domestic drinking water supply system

The task of this system is to satisfy the demand for drinking water in the nuclear power plant facilities. The system serves both communal and process needs. The drinking water network is built from the pre-defined connection point.

Domestic sewerage systems

The task of this system consists in the collection and removal of sewage water generated in the nuclear power plant facilities. Handling of waste water originating in the controlled area is carried out in separate designated sub-systems.

CHAPTER 9.A.3 – PROCESS AUXILIARY SYSTEMS

Process and post-accident sampling systems

In order to assure optimal water chemistry conditions, real-time, reliable and precise measurements of chemical parameters are required for monitoring. In all operating states and conditions, efforts should be made to obtain on-line information and data in order to check the compliance of the unit operation with relevant technical requirements. Another important task of these systems is to assure that the unit operations during transients are in accordance with the operating instructions and to allow early identification of long-term abnormal trends causing operational problems. Thus, it is very important to ensure that information on the primary water chemistry during transient modes will be received in due time, and respective measurements will be conducted with high frequency in a prompt, precise and reliable manner.

Fulfilment of the above task is supported by the following four primary sampling and measuring systems:

1. Primary circuit automated chemistry monitoring system

   The task of this system is the performance of automated control of the primary coolant parameters.

2. Sampling system used at water treatment plants and auxiliary reactor systems

   The task of this system is to assure manual sampling of the water media and primary coolant from auxiliary plant systems for further laboratory analysis of their quality parameters.

3. Post-accident sampling system

   The task of this system is the performance of the primary coolant sampling after design basis accidents for laboratory analysis of its chemical and radiochemical parameters, as well as sampling from emergency borated water storage tanks in case of design basis accidents.

4. Sampling system for severe accident conditions

   The task of this system is the assurance of sampling under severe accident conditions followed by laboratory analyses with the purpose to qualify the accident state and support strategic decision-making.

Equipment and floor drainage systems

The chapter describes equipment and floor drainage systems used for the treatment of radioactively contaminated and presumably radioactively contaminated drains generated in the reactor building, safety building and nuclear technology service building. The task of these systems is the assurance of selective collection of drains in controlled conditions based on their supposed radioactive contamination and transfer of these drains for further treatment. The above-mentioned drains may appear in the result of drainage, leakage, water from fire-fighting and other technology-related processes.

Steam generator blowdown system

The steam generator blowdown system assures maintenance of secondary water chemistry parameters within the specified limits. The system is responsible for continuous and periodical removal of the secondary coolant for cleaning and its return to the secondary circuit after cleaning.

Receiving system of radioactive drainages

The system is responsible for the collection of boron-bearing drains to assure the return of boric acid-containing media and clean condensate to the technology-related processes.

Primary coolant treatment system

The task of this system is the processing of collected boric acid-containing primary drains to bring them into a state allowing their re-use in the technology-related processes. The processing technology includes purification of boron-containing media to specified boric acid concentration, treatment of clean condensate appearing during and purification of boron media with specified concentrations.

Decontamination system

The decontamination system is responsible for the decontamination of equipment and plant rooms prior to the planned preventive maintenance and inspections, in order to reduce the personnel radiation exposure.

Decontamination procedures and tools are available for the removal of surface radioactive contamination of certain primary systems, system components and rooms located within the controlled area during normal operation. In addition, this system can be used to reduce the level of radioactive contamination during operational events and accident situations.

A separate decontamination system is available for the decontamination of control rod drives belonging to the reactor protection and control system, to reduce the level of radiation exposure resulting from maintenance and inspection works performed on the drives.

Group of equipment for the special laundry water collection

This group of equipment is responsible for the organised collection of special laundry water appearing after the washing of primary protective clothing, control of water parameters and water transfer for further processing.

Blowdown system for NI instrumentation lines

The system is responsible for filling impulse pipelines belonging to the primary circuit instrumentation and control measurements with the medium corresponding to process parameters, blowdown of pipelines in case of unreliable measurement results and removal of precipitating boric acid crystals from the impulse piping and remote control transmitters.

Steam supply system of auxiliary building

The systems supplies steam for auxiliary building systems and consumers. Among others, these consumers include evaporation equipment, make-up water deaerator tank and equipment belonging to the system for liquid radioactive waste solidification. Based on contamination level, condensing water is either returned to the secondary circuit or transferred for processing to the primary side.

CHAPTER 9.A.4 – AIR AND GAS SYSTEMS

Process air supply system

Similarly to most of industrial facilities, compressed air supply plays an important role in the nuclear power plant as well. Compressed air supply is divided into the process air supply system, instrument air supply system and compressor station.

The process air system and its associated compressor station supply standard industrial-grade air with the pressure of 8 ± 0.5 bar to pneumatic tools used during operation and maintenance, certain technologies (e.g. nitrogen system purge air, loosening of ion exchange resin column charge, etc.) and periodically to the containment pressure test system. Flow rate in the system varies between 500-6000 m^3/h during normal operation, 2000-8000 m^3/h during planned outages and makes 12000 m^3/h for the containment pressure testing.

Consumers requiring higher purity air (e.g. pneumatic valves, certain chemical technologies, etc.) are supplied by the instrument air system. Depending on the plant operation mode, this system supplies compressed air to the consumers with the flow-rate of 20-500 m^3/h.

Service air supply system

In case of certain technologies and applications, it is necessary that chemically neutral (inert) gases should be supplied to the concerned equipment or system. In terms of the new power units, these applications include creation and maintenance of gas blankets (e.g. hydroaccumulators) or use as purge gas (e.g. displacement of explosive media in case of hydrogen-using systems for technologies like generator gas cooling system or primary circuit bubbler) with nitrogen used for this purpose. The required gas is produced at the nitrogen station from liquid nitrogen at a pressure of 17.9 bars at two pressure stages.

Nitrogen at the pressure of 58 bars is delivered to the consumers of the containment by the high-pressure nitrogen system. Liquid nitrogen is supplied to the evaporators by reciprocating cryogenic pumps at the pressure of 200 bars; maximum continuous capacity of the system is 400 m^3/h. In order to compensate irregularities of the consumption, 4 groups of gas cylinders with the pressure of 200 bars and volume of 7.2 m^3 each were designed as gas buffers. The operating pressure is set on the pressure relief valve groups after the buffers.

The low-pressure nitrogen system supplies consumers at the pressure of 9.8 bar with maximum flow rate of 100 m^3/h. Liquid nitrogen flows from the cryogenic tanks directly to the evaporators, proportionally to the withdrawn volume. Similarly to the high-pressure nitrogen system, the operating pressure is set on pressure relief valve groups; however, no buffers are provided in the low-pressure system.

Others

This chapter describes other gas systems related to nuclear safety and not mentioned earlier in other subsections. The only one of those not discussed previously is the oxygen supply system, which is responsible for supplying oxygen to the hydrogen burning system connected to the primary circuit. Together, these systems are responsible for the controlled burning of hydrogen released via radiolysis in the primary coolant, preventing the appearance of explosive concentrations and large amounts of gas.

The oxygen supply system supplies oxygen stored in two groups of gas cylinders at the pressure of 150 bars to the hydrogen burner with pressure reduction in the flow range of 0.13 m^3/h and 1.6 m^3/h.

CHAPTER 9.A.5 – HEATING, VENTILATION AND AIR CONDITIONING SYSTEMS (HVAC)

MCR and ECR HVAC

These systems are equipped with isolating and filtering components allowing the personnel in the main- and emergency control rooms to work without personnel protective equipment during anticipated operational occurrences, design-basis and beyond-design basis accidents. Ventilation and air-conditioning systems supporting main- and emergency control rooms of the power units are independent from each other - to the extent reasonably achievable - and are protected against a single failure.

Reactor building HVAC

These systems are responsible for assuring proper working environment for the operating personnel working in the rooms accessible during normal operation. Ventilation systems installed in the reactor building maintain conditions corresponding to the environmental qualification of the systems and system components located in the building rooms, limit the spread of gaseous radioactive materials and ensure compliance with the limits established for air activity concentration. As the containment function is fulfilled by two separate containment structures, annular space between the primary and secondary containment is equipped with the ventilation system maintaining negative pressure and filters to remove any radioactive material that may leak from the containment.

HVAC of controlled and supervised areas

Ventilation systems installed in the controlled area are designed to prevent the spread of gaseous radioactive materials and their entry into the uncontrolled environment. The use of ventilation and air-conditioning systems limiting the spread of radioactive materials ensures that the volume of air exchange in the given room is proportional to the concentration of radioactive materials moving in the air, and the airflow is directed from less contaminated rooms to potentially more contaminated ones. The number and arrangement of systems ensure the separation of ventilation of potentially more and less contaminated rooms.

As the loss of ventilation and air conditioning systems affects the operation of systems and system components performing safety functions, these ventilation and air conditioning systems were categorised into a relevant safety class and developed with appropriate redundancy and on the basis of respected differentiated requirements to fulfil reliability requirements.

Turbine building HVAC

Heating-, ventilation- and air-conditioning systems of the turbine building are designed to support the operation of equipment and to maintain air parameters necessary for the comfortable stay of the personnel. The operation of ventilation systems located in the turbine building is expected only under normal operating conditions.

CHAPTER 9.A.6 – FIRE PROTECTION SYSTEMS

Fire-fighting water supply system

Fire-fighting water supply system of the nuclear power plant shall provide the amount of water necessary for fire-extinguishing:

- to the outdoor surface hydrants located on-site
- to the wall-mounted hydrants inside the plant buildings located on-site
- to the built-in automatic water-based fire-fighting systems
- to mobile fire-fighting devices of the fire brigade

System of automatic fire-fighting units for main buildings and structures

These automatic fire-fighting units are installed in main buildings and structures located on the nuclear power plant site and are capable of extinguishing detected fires without human interventions.

System of automatic fire-fighting units for auxiliary buildings and structures

These automatic fire-fighting units are installed in auxiliary buildings and structures located on the nuclear power plant site and are capable of extinguishing detected fires without human interventions.

System of automatic gas fire-fighting units

These automatic gas fire-fighting units are installed in the rooms of buildings located on the nuclear power plant site and are capable of extinguishing detected fires without human interventions with the use of gaseous extinguishing agents.

Modular units system for automatic sprayed water fire-fighting

These automatic sprayed water fire-fighting units are installed in the rooms of buildings located on the nuclear power plant site and are capable of extinguishing detected fires without human interventions with the use of water mist.

Passive fire protection systems

These systems represent the complex of architectural fire protection solutions aimed at preventing the propagation of fire within the plant site by means of assuring appropriate fire distances between adjacent buildings, risk-related fire sections and fire compartments, as well as by the installation of fire-retardant building structures (fire barriers). The use of building structures with relevant performance-based fire protection parameters and application of fire-retardant coatings (paints) allows to ensure efficient fire protection of the concerned buildings. The use of passive fire protection of systems and system components important to nuclear safety allows to ensure that the frequency and effects of fires will be minimised, the nuclear power plant can be safely shut down during and after a fire, residual heat can be removed and the escape of radioactive materials into the environment can be prevented.

Fire alarm, communication and fire warning

Rooms of the buildings located on the nuclear power plant site are equipped with automatic fire alert and fire warning systems. The task of these systems is to detect the fire at an early stage and to alert the workers staying in the concerned room and the personnel participating in fire response operations. All fire alarms are displayed in the main- and emergency control rooms of the nuclear power plant.

Fire Protection Instrumentation and Control System

The system is designed for fire detection, fire alarm and automatic fire extinguishing in the technology-related premises and control rooms of the nuclear power plant in a manner that would not disturb the operation of safety systems in case of either normal or spurious actuation. The system is designed to maintain integrity and operation in all operating conditions of the nuclear power plant. The fire protection instrumentation and control system is connected to such unit processes as ventilation systems, fire-fighting water supply systems, unit fire-extinguishing systems, fire-fighting and emergency response organizations, ensuring the operability of fire-fighting and fire alarm conditions in all operating states of the power unit.

Systems for heat and smoke removal and overpressure in case of fire

The task of these systems is to carry off smoke and heat generated in the buildings of nuclear power plant in case of a fire and to prevent the entry of combustion gases into the building space and areas used as evacuation routes. The system for heat and smoke removal will be installed only in such places of the premises in technology-related facilities where there is no chance of radioactive material release into the environment.

Fire safety assurance in terms of site layout

This subsection describes distances between buildings and structures to be constructed on the plant site, as well as the main transport and access routes.

Fire prevention measures for the equipment containing flammable liquids and gases

The chapter describes fire prevention measures for the equipment containing flammable liquids and gases.

CHAPTER 9.A.7 – EMERGENCY DIESEL GENERATOR AND SUPPORTING SYSTEMS

The chapter provides information on diesel generators belonging to the emergency power supply system, which assure - in an automatic mode - the emergency power supply necessary for the safe shutdown, cooling and maintenance of safe shutdown state of the power units in case of loss of normal power supply of house load consumers or loss of power supply in the emergency power supply system. Each of 4 independent emergency power supply systems is equipped with one emergency diesel generator.

Emergency diesel generators and supporting systems are described in the following subsections:

- The first subsection focuses on the motors of emergency diesel generators and motor-driven generators.
- The second subsection describes independent fuel supply and storage systems of emergency diesel generators, which are able to ensure continuous 72-hour fuel supply to diesel generators without refilling.
- The third subsection describes cooling systems of emergency diesel generators. Each diesel generator has its own cooling system performing two functions: maintenance of the necessary temperature of the diesel generator and oil lubrication system in standby mode to assure quick diesel generator start-up, as well as the removal and transfer of heat generated by the diesel engine during operation, to the environment via water-to-air heat exchangers.
- The fourth subsection presents compressed air supply systems required for a quick start-up of diesel generators within 15 seconds. All diesel generators are equipped with starting- and control air system providing sufficient amount of compressed air for five successful successive diesel generator start-ups and for the operation of pneumatic control valves.
- The fifth subsection describes independent oil lubrication systems of emergency diesel generators. In addition to the provision of lubrication for bearings and friction components during diesel generator start-up, operation and shutdown, these systems play an important role in the removal and transfer of heat generated in the result of friction, to cooling systems.
- The sixth subsection focuses on the systems for combustion air supply to diesel generators and systems for the removal of combustion products. This subsection also contains the description of diesel generator turbochargers improving performance characteristics and efficiency of diesel generators.

CHAPTER 9.A.8 – OTHER DIESEL GENERATORS AND SUPPORTING SYSTEMS

The chapter describes primary circuit diesel generators of the new power units, as well as diesel generators used in case of complex- and severe accidents. Diesel generators for primary circuit and complex accidents provide - in an automatic manner - electric power supply to the essential preventive power supply systems of the primary circuit and to essential severe accident power supply systems in case of loss of power supply in the above-mentioned systems. Each power unit has two independent primary preventive power supply systems and two essential severe accident power supply systems; each of them is equipped with its own independent diesel generator. In addition, each power unit is equipped with a mobile diesel generator for severe accident management that can be put in operation, whenever necessary.

The description of primary circuit diesel generators, diesel generators used in case of complex accidents and their supporting systems is provided in the structure identical to that used in the subsection dealing with emergency diesel generators.

CHAPTER 9.A.9 – MISCELLANEOUS AUXILIARY SYSTEMS

Communication systems

The hardware complex of communication and alert systems of the nuclear power plant consists of two interconnected parts: internal communication complex and external communication complex.

The internal communication complex comprises a set of individual systems enabling the performance of operational and maintenance functions in normal operating states of power units, as well as in case of an emergency or an accident. The internal communication network consists of six voice communication systems.

The external communication complex ensures the access to external communication network during normal operation and maintenance of the nuclear power plant, as well as in case of an emergency or an accident.

Lighting systems

Lighting systems (normal and emergency lighting) are intended to provide effective lighting for the personnel working inside the rooms and outdoors on the site of the nuclear power plant, depending on the room function and operating state of the nuclear power plant, as well as to provide adequate lighting for evacuation routes.

Containment separation system

The containment separation system was designed to ensure the operation of ventilation systems installed in the reactor building by separating the internal space of the reactor building into two compartments (zones).

CHAPTER 9.A.10 – OVERHEAD HEAVY-LOAD HANDLING SYSTEM

The chapter focuses on overhead heavy-load handling systems of new power units. Transportation and handling of nuclear fuel (fresh and spent fuel assemblies) is an important activity in terms of nuclear safety. The design of mechanical, electrical, instrumentation and control parts of the heavy-load handling machinery shall be highly reliable to exclude the possibility of fuel damage.

The first chapter describes the crane installed in the reactor building (better known as a polar crane). The crane is intended for the performance of load handling and movement operations required in the reactor building, including the support of construction works at the unit, nuclear fuel transportation and reactor maintenance during the unit operation.

The second subsection describes the crane installed in the fresh fuel storage area. The crane is a part of fresh fuel storage and handling system; its task is the performance of load-handling operations related to the receipt and preparation of nuclear fuel for use.

In terms of safe electricity generation, an important role is played by the crane installed in the turbine building and described in the third subsection. Maintenance of the turbine, generator and auxiliary systems requires handling of heavy parts and components. This crane with the load-bearing capacity of 350 tons and operating radius of 56 meters is the largest overhead handling machine at the power unit.

Movement of materials into and out of the containment is carried out with the help of a cargo lock. The fourth subsection describes the crane connected to the containment external wall. The crane is responsible for transporting the arriving nuclear fuel, auxiliaries, spare parts and for removing waste from the containment between the cargo lock and ground level.

The fifth subsection shows the crane involved in operations on the receipt and handling of fresh fuel assemblies arriving at the plant site and transportation of spent fuel assemblies.

CHAPTER 9.A.11 – CHEMISTRY

Primary- and secondary water chemistry of the nuclear power plant is designed with consideration to the following requirements:

- nuclear safety
- radiation safety norms
- operational safety

During the unit operation, the primary and secondary coolant can be contaminated by materials that may induce harmful processes.

One of the most important tasks in terms of reducing the level of coolant contamination is minimisation of external contaminants. The spread of external contaminants entering the coolant depends on their physical and chemical characteristics and concerned systems. The contamination can deposit in certain parts of the equipment reducing operational safety and efficiency or can increase the level of primary dose rate, leading to the deterioration of operational characteristics and hindering maintenance. Deposits on the fuel assemblies are especially dangerous since they do not only reduce heat transfer but can also cause damages in fuel rods.

The other important task is to limit the corrosion rate of structural materials that can be ensured by the proper adjustment of water chemistry parameters and optimisation of interactions between the metal and coolant.

The task of the secondary side water chemistry is to minimise the risk of stress corrosion processes in steam generators. To fulfil the above task, minimum technically achievable concentrations of contaminants in the secondary side medium should be ensured.

CHAPTER 9.B – BUILDINGS AND CIVIL STRUCTURES

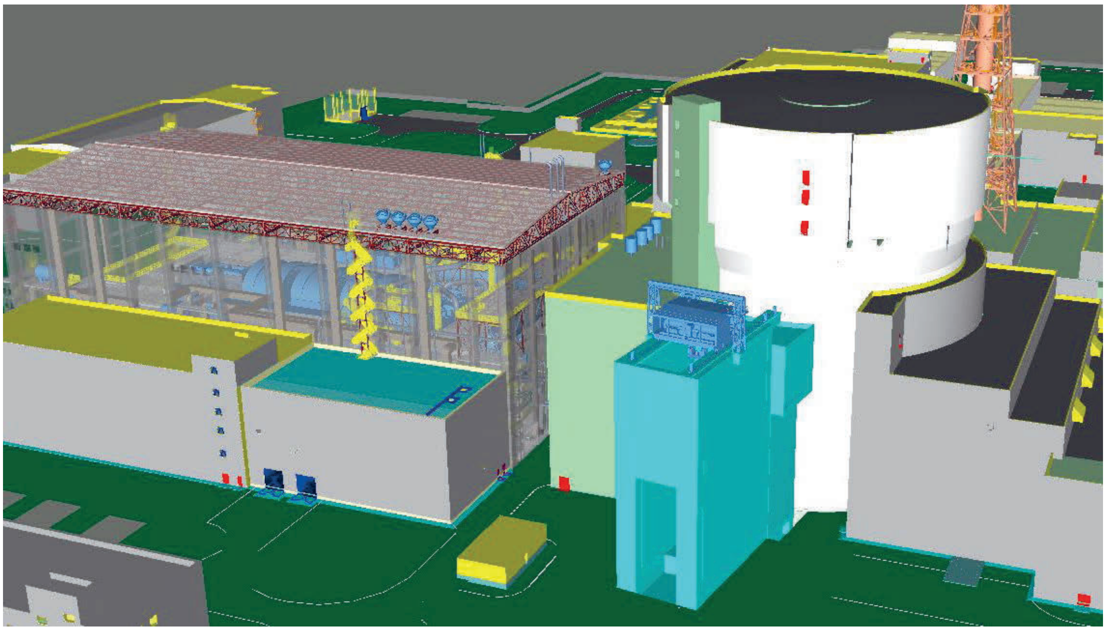
The chapter provides information on the buildings and civil structures located on the nuclear facility site and necessary for the nuclear power plant operation. These buildings and civil structures were designed, firstly, in accordance with Hungarian and, secondly, international requirements described in Chapter 3.

The chapter contains a general description of the nuclear facility listing all the buildings and civil structures. The chapter describes the so-called nuclear island and its operation. The nuclear island includes the reactor building and group of buildings surrounding it.

The chapter divides building and civil structures necessary for the nuclear power plant operation into two groups:

1) those directly or indirectly connected to the nuclear technology,
2) other buildings and civil structures not connected directly to the nuclear technology but necessary for the nuclear power plant operation.

In these subsections, you can get information on functions of the buildings and civil structures, their connection with other structures, classification of building into different classes, basic design data taken into consideration during the design of the given structure, as well as descriptions of structural and applied materials, floor plans and cross-sectional views.



Containment and connected buildings

In case of buildings and civil structures belonging to group 1), the detailed description of monitoring, test and maintenance activities is provided as well. In addition to the above, this subsection reviews the specific features of lifetime design - including ageing management and radiation protection requirements - and, last but not least, focuses on the safety-related design aspects (design with respect to external and internal hazards). The fulfilment of the Nuclear Safety Code requirements for the given building is evaluated in this subsection.

In many cases, buildings and civil structures of Units 5 and 6 are identical; however, some differences still can be identified. Though the report covers mainly the details pertaining to the buildings of Unit 5, it describes also the differences specific for Unit 6. Furthermore, at the current stage of the unit design process, the subsection presents the strength calculations of support structures connected with the buildings belonging to safety classes 1,2 and 3, within the framework of a background document.

Among others, group 1) includes reactor building, auxiliary building, control building, fresh fuel storage and civil structures connected with essential service water system. Among others, group 2) includes turbine building, other civil structures connected with water cooling systems of the turbine building consumers, electricity network buildings, common plant structures, water preparation buildings, various oil storages and other infrastructural facilities (utilities, roads and railways).

CHAPTER 10 – STEAM AND POWER CONVERSION SYSTEMS

Role and general description

The steam and power conversion systems are made up of the main steam supply system, turbine, auxiliary systems and equipment, main condensate and feedwater system. The turbine generator equipment converts thermal energy of high temperature and high pressure steam arriving from the plant steam generators, into electrical power.

Main steam supply system

The task of the main steam supply system is to transfer the reheated steam with the pressure of 6,9 MPa and temperature of 286 °C generated by steam generators (4x1600 t/h) to the steam turbines. Each steam generator has its own main steam header.

The system is equipped with isolating valves assuring immediate cut-off and isolation of the steam generator in the event of a steam and / or feedwater pipeline rupture or primary-to-secondary leakage.

The main steam supply system is equipped with atmospheric discharge valves to discharge main steam arriving from steam generators into the atmosphere, for the purpose of heat removal during normal operation if the turbine bypass system is disabled or whenever the emergency unit cool-down is necessary.

Feedwater systems

Feedwater systems are intended for storage, deaeration and transfer of feedwater to steam generators (with the pressure of 8 MPa, temperature of 225 °C, and flow-rate of 6400 t/h) via high-pressure heaters. The system consists of a feedwater tank, a feedwater pump connected with four booster pumps and associated auxiliary systems.

Each steam generator has its own feedwater collector.

High pressure feedwater heaters system
The system is intended for feedwater heating aimed at improving the electric efficiency of power units. Heating of feedwater is carried out in two stages and two parallel rows; the outlet temperature of feedwater is 225 °C.

Auxiliary feedwater system

The auxiliary feedwater system supports the operation of the main feedwater system. The system performs start-up and supplementary functions (filling, heating, deaeration), supplies feedwater from the deaerator to the steam generator during the unit initial start-up and unit shutdown. The system consists of two auxiliary feedwater pumps and connected auxiliary systems.

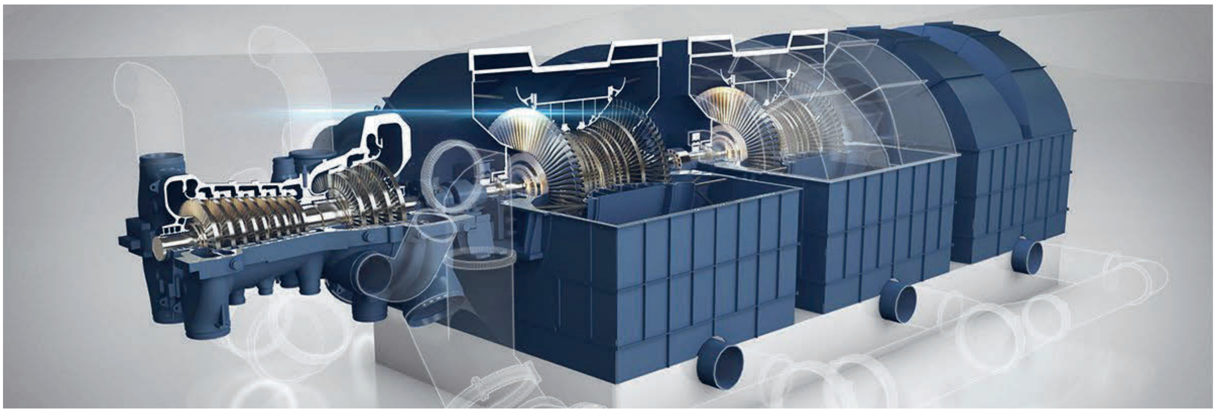
Turbine generator

The task of the turbine is to convert the thermal energy of steam coming from the steam generators, supplied by the main steam system, into the rotational energy of the generator and finally into electrical power.

The half-speed (1500 1/min) ARABELLE™ steam turbine consists of one low-intermediate pressure cylinder and three low-pressure cylinders.

The turbogenerator is a four-pole half-speed (1500 1/min), three-phase synchronous machine (with rated active power of 1270 MW and rated terminal voltage of 24 kV) with direct hydrogen -cooled rotor and stator housing and water-cooled stator winding. The primary task of the turbogenerator is to convert - operating in parallel with the electrical grid - the mechanical energy coming from the turbine through rigid shaft clutch connection into electricity.

The turbogenerator is connected to the rotating diode excitation system, the function of which is the excitation and de-excitation of the turbogenerator, as well as the regulation of turbogenerator terminal voltage and reactive power with the help of excitation controllers.



ARABELLE™ steam turbine

Turbine and condenser systems

Moisture separator and re-heater system

The purpose of this system is the reduction of moisture content of the steam leaving the high-pressure turbine (moisture separation) and steam re-heating (in two stages), followed by the transfer of this superheated steam to the intermediate pressure cylinder of the turbine to improve the unit energy efficiency.

Main condenser

The function of the condenser is to receive, cool and condense the steam discharged from the turbine and turbine bypass system, as well as to receive condensate and drains from several other systems located in the turbine building, in order to ensure maximum turbine efficiency.

Main condenser vacuum system

This system is responsible for creating and maintaining vacuum in the condenser.

Main circulating water system

The task of the main circulating water system consists in the supply of cooling water to the generator of the steam turbine and removal of waste heat from the cycle. The main circulating water system of the power units uses fresh water; the cooling water is delivered from the Danube through the inlet channel, and is then returned to the Danube via the outlet channel. The use of fresh-water cooling is very advantageous as compared to the solution based on the use of cooling towers since in the case of fresh-water cooling the units operate with the highest efficiency at all times, resulting in additional 15-20 MW of electrical power.

The volume of cooling water withdrawn from the Danube can vary between approximately 43 and 61 m^3/s per unit, the nominal water intake is 58 m^3/s. In order to guarantee that the operation of the nuclear power plant will not be limited on extremely hot days, an additional cooling system is developed to assure the compliance with the Danube water temperature limits; this system will transfer a part of heat removed from the condensers to the atmosphere using small forced draft cooling towers.

Main condensate system

The main condensate system is responsible for the transfer of main condensate from the condenser to the feedwater tank equipped with a de-aerator.

Low pressure feedwater heaters system

The task of the low pressure feedwater heaters system is the heating of main condensate in the low pressure feedwater heater to improve the system efficiency.

Turbine auxiliary systems

Turbine lubrication system

The system is used for the lubrication of the turbine and generator bearings and removal of heat generated due to friction from the bearings. The required amount for filling the lubrication oil tank is 141 m^3. The main oil pump is driven from the turbine shaft using the drive mechanism. The oil during turbine start-up and shutdown is supplied by auxiliary oil pumps. In case of a failure of auxiliary oil pumps, the oil supply is assured by an emergency oil pump. The oil supply line is equipped with protective pipe (pipe-in-pipe design) preventing oil from escaping into the turbine hall.

Turbine control oil system

The system is responsible for the supply of high-pressure control oil for the operation of the turbine quick-release and control valve servomotors, as well as for the operation of electro-hydraulic unit of the turbine protection system. The system operates with 1,400 litres of control oil; a refill of 10-12 litres is required annually.

Turbine gland sealing system

The turbine gland sealing system is designed for the prevention of steam escape from the turbine (into the turbine hall), as well as for the prevention of air intake into the turbine vacuum systems during normal operation.

Generator auxiliary systems

The turbogenerator should operate with high operational safety and high availability. To fulfil the above task, the turbogenerator operation should be supported by auxiliary systems assuring proper cooling and lubrication oil supply.

Oil supply system for generator shaft sealings

In case of hydrogen-cooled turbogenerators, the oil supply to shaft sealings is provided by a separate system connected to the bearing lubrication oil system. The task of this oil supply system consists in the supply of oil to shaft sealings installed at both ends of the turbogenerator, prevention of hydrogen leakage from the generator housing, oil cooling and degassing.

Generator gas supply system

The task of the turbogenerator gas supply system is filling - depending on the operation mode - the generator with gas (hydrogen, nitrogen or air), maintenance of the specified gas pressure (assuring proper gas purity and humidity), assurance of the possibility of checks after filling the generator with gas and necessary interventions, cooling and gas venting from the generator. The generator gas supply system includes an emergency hydrogen discharge system intended for the quick reduction of hydrogen pressure via a special pipeline installed on the roof of the turbine building.

Cooling water system for generator stator winding

The task of the cooling water system for generator stator winding is to cool the stator winding of the turbogenerator through the circulation of desalinated water and to ensure continuous monitoring of the desalinated water temperature and conductivity.

Turbine bypass system

In case of accidents affecting the turbine and generator, the steam from the main steam supply system is transferred to the condenser using the turbine bypass system.

House load steam supply system

The system includes the auxiliary steam header, reducing valve of the auxiliary steam system, safety valves, connected pipelines and valves. The task of the house load steam supply system is the supply of steam to the turbine hall consumers, to the district heating system, to the facility heating system, to the operating steam consumers and, if necessary, to the adjacent power unit. During normal operation, the house load steam supply system is fed from steam bleeder 5 of the turbine (10.3 bars); however, the supply from the main steam system can be also assured with the use of two reducing valves. Through the auxiliary steam line connecting the units, it is possible to provide the steam required during the start-up of the adjacent power unit that can be supplied from an electrically heated auxiliary boiler.

CHAPTER 11 – RADIOACTIVE WASTE MANAGEMENT

This chapter describes technical solutions applied for the management and disposal of radioactive waste generated during normal operating conditions and anticipated operational occurrences, taking into consideration the provisions of the effective national policy and program for radioactive waste management in Hungary.

The chapter outlines the solutions aimed at safe processing of the radioactive wastes originated during the plant lifetime and confirms the compliance of developed measures. Sources, quantities and time of generation are determined for solid, liquid and gaseous radioactive wastes in accordance with the design requirements. The chapter provides information on the quality, form, processing, storage and transportation methods to be applied in case of radioactive waste with different activity levels, taking into account all operating states and modes of the nuclear power plant. The chapter specifies solutions aimed at minimising the total volume of radioactive wastes generated throughout the plant lifetime.

Methods and steps of the designed processing of solid radioactive wastes appearing during the plant lifetime are as follows:

- Radioactive and non-radioactive wastes are separated already in the place of their generation in order to assure that only those requiring special treatment will be transferred for processing.
- Radioactive wastes are classified into two categories, i.e. compactable (e.g. contaminated working clothes, personal protective equipment, spent filter cartridges of ventilation systems) and non-compactable wastes (e.g. dismantled equipment with radioactively contaminated surface not subject to repair).
- The volume of compactable radioactive wastes will be reduced by a high-pressure compactor with a compressive force of 20000 kN, allowing to reduce the waste amount to one-tenth of its original volume. In other words, after the compaction of 10 drums containing radioactive wastes one drum of radioactive wastes will be received for further final disposal. It should be emphasised that almost 80-85% of the generated solid radioactive wastes will belong to the compactable category, thus, based on the above-said, the volume of wastes to be transferred for final disposal will be significantly reduced.
- Non-compactable wastes will be placed into steel containers. In order to assure the most efficient utilisation of the storage volume available inside the container, "empty" spaces between the waste debris are filled by cemented liquid radioactive waste added to the container volume.

The purpose of liquid radioactive waste processing system is the selective collection, storage and processing of radioactively contaminated waste water, small amounts of contaminated organic solvents and oils appearing during the plant operation. The complicated technical challenge is managed in the technical design documents by providing the following functions for the treatment of liquid radioactive wastes:

- collective selection of waste water,
- storage of liquid radioactive wastes,
- processing of liquid radioactive wastes by means of volume-reducing evaporation and ion-exchange purification,

- solidification of liquid radioactive wastes in reinforced concrete containers by cementation.

As a result of waste processing by efficient modern technologies, several thousand of cubic meters of diluted, liquid radioactive waste generated annually are transferred in 25 pcs. of 1,5 m^3 containers for final disposal in the National Radioactive Waste Repository in Bátaapáti.

The annual volume of processed solid radioactive wastes with low and medium activity levels is 60 m^3 per power unit; this volume is transferred to the National Radioactive Waste Repository, as well.

CHAPTER 12 – RADIATION PROTECTION

This chapter deals with the issues of radiation protection design. The operation of nuclear power plants is accompanied by the origination of radioactive materials leading to the personnel radiation exposure by emitting ionising radiation. Protection against ionizing radiation is one of the most important pillars of safe operation, thus, a strong emphasis is placed on assuring radiation protection of the plant personnel and population.

As a result of technical measures implemented at the units, radiation exposure of the plant personnel and the public remains far below the regulatory limits. The predicted public dose contribution of 60 nSv / year resulting from the normal operation of the nuclear power plant is an extremely low value; its magnitude is comparable to the natural dose contribution equal to a one-hour stay outdoors, so it can be considered negligible in terms of health risks.

The chapter outlines general principles of radiation protection, radiation protection policy, strategy, methods and technology-based solutions. The chapter demonstrates radiation protection measures in respect of the plant personnel and the public, in connection with different operating states of the power unit.

Information contained in this chapter includes brief descriptions of radiation protection solutions taken into consideration in the plant design, and shows how the basic radiation protection measures were considered at the present design stage of project implementation. This document confirms that we have developed such operational and administrative solutions based on continuous monitoring of radioactive sources, which guarantee As Low As Reasonably Achievable (ALARA principle) radiation exposure on the plant personnel and the public.

The chapter consists of several sub-chapters. The first sub-chapter deals with the assurance of as low as reasonable achievable radiation exposure levels, with the description of:

- general radiation protection directives
- leadership commitment
- design solutions
- operational solutions

The next sub-chapter describes the sources of ionising radiation. Here we describe both fixed and mobile radiation sources, as well as ionising radiation caused by potentially available gaseous radioactive materials. Potential ways of irradiation are mentioned here as well. Certain sub-chapters outline:

- ionising radiation caused by solid and liquid radioactive materials,
- ionising radiation caused by gaseous radioactive materials

One of sub-chapters describes the details of radiation protection design, with special attention to:

- specifics of the facility design
- development of shielding
- ventilation
- fixed radiation- and aerosol monitoring system
- organisational and administrative actions related to radiation protection.

Similarly, one of sub-chapters presents dose calculations with the detailed identification of radiation-affected areas under different operating conditions of the nuclear power plant and describes potential radiation exposure doses effecting the plant personnel and the public. One of the purposes pursued by this chapter is to describe regulatory dose limits, as well as administrative and practical measures applied for dose control.

The sub-chapter dealing with the administrative organisation implementing radiation protection program, equipment, measuring devices, facilities and connected procedures provides a detailed overview of the following elements:

- set up and structure of the organisation implementing radiation protection program
- installation of fixed radiation monitoring systems
- operation of radiation monitoring and sampling system
- implementation of radiation protection procedures and methods

The last sub-chapter presents analysis and evaluations confirming the fulfilment - by means of systems, technical and administrative measures described in this chapter - of design objectives related to radiation protection and regulatory requirements.

CHAPTER 13 – CONDUCT OF OPERATIONS

This chapter describes how the current organisational structure of Paks II. Ltd. will be transformed in a controlled manner into a nuclear plant operator organisation meeting nuclear safety requirements connected to various phases of the project implementation process. The current organisational structure shall be considered as the initial state, while the final state shall presume the operation of both nuclear power units with the organisation performing its tasks in accordance with the Standard Nuclear Performance Model (SNPM).

Organisational structure, personnel headcount and operator capabilities will be modified in accordance with the tasks arising at various project stages. At the current stage of project implementation - i.e. submission of implementation license application - the personnel headcount and qualifications are lined up with the performance of licensing and preparation tasks. The personnel headcount will change along with the project progress and will eventually achieve the designed operating personnel headcount of 1060 employees (0,42 MW/person), required for safe and economically efficient operation of the power plant units. In terms of the operating personnel and respective personnel qualifications, the project is divided into well-separated time-periods. Instead of indicating names of organisational units existing at subsequent project stages, it would be reasonable to describe their functional tasks: Stages and required personnel headcount are as follows:

- *Preparation for implementation license application* (preceding period)
  The number of the project headcount is fixed at this stage; the task consists in the submission of implementation license application by specified deadline. In order to do so, all organisational units shall demonstrate efficient cooperation with the Contractor, within the scope of own competence.

- *Period following the submission of implementation license application (current stage)*
  The project shall get prepared for employing the required number of personnel, taking into consideration properly qualified young specialists who are willing to do this work. One should also keep in mind the employee age structure at least for the next 10 years. The tasks of already available project staff include acquisition of manufacturing, procurement, construction and water right licenses, approval of received documents, on-the-job training, and acquisition of basic and special knowledge.

- *Construction and installation* (intermediate stage)
  The main tasks at this stage cover on-site acceptance, quality control and supervision of work and activities performed by subcontractors. No exact personnel figures exist in this regard, thus, a WANO recommendation to determine an intermediate reference point – "the first concrete" in this case - was taken into consideration to determine the necessary headcount. During the construction stage, theoretical and practical training of the future operating, maintenance and technical support personnel will be carried out, include practical training in the simulator facility.

- *Preparation for commissioning* (initial preparation stage)
  Training of all the operating and maintenance personnel for Unit 5 is completed at this stage. Technical specialists participate in the on-site acceptance of equipment and systems.

- *Inactive commissioning (individual tests, flushing, blow-down)*
  This stage includes individual testing of the equipment and systems in accordance with the commissioning work program and preliminary operating instructions. The work implies gaining of practical experiences by the personnel, start-up and shutdown of the equipment, conduct of individual tests, preparation and performance of maintenance activities. By the end of this stage, all the personnel employed in safety-related positions shall have obtained necessary licenses and permits.

- *Active commissioning (operational tests, hot functional testing, HZP, i.e. testing at minimum controlled power level, operational load)*
  This stage includes loading of the first nuclear fuel into the reactor core and connected inspections, first criticality, unit power increase based on the performed operational tests and performance of other pre-determined tests.

- *Unit 5 in operation, commissioning works at Unit 6 (operation and commissioning stage)*
  At this stage, Unit 5 is operating at full power, while the personnel of Unit 6 are participating in on-the-job and practical training, as described herein above. The personnel headcount at this moment is similar to that at the last stage with the only difference that a part of the technical support personnel has already joined the operations, while the other part is continuing the commissioning activities.

- *Both power units are in operation* (operation stage)
  At this stage, similar to other nuclear power plants, the operations and organisational structure are in accordance with the so-called INPO model (Institute of Nuclear Power Operations).

In accordance with the requirements of the Nuclear Safety Code, this chapter of the Preliminary Safety Analysis Report touches upon the following issues:

- Personnel training and training forms. Methods and recommendations aimed at checking and improving the level of knowledge. The purpose of developing personnel training and methods is to assure the availability of sufficient number of highly qualified specialists at Paks II. Ltd.
- Training centre, including the installation of a full scale simulator; requirements, procedure of validation and verification, development of applied practices. Training of normal and abnormal operations performance.
- Development of maintenance and repair strategy. Equipment reliability and condition-oriented maintenance planning: advantages, disadvantages and areas of application.
- Ageing management, change management of systems and system components during the plant lifetime. Preparation of programs for equipment replacement and stockpiling.
- Storage, handling, transportation and disposal of fresh- and spent fuel assemblies.
- Instructions and procedures related to core re-arrangement, loading and unloading, safety discipline, provisions concerning subcontractor personnel and requirements to “clean” assembly.
- Utilisation of internal and external operating experiences, documentation of events and follow-up of corrective actions.
- Reviews and audits conducted by the independent organisation of the Licensee appointed for the performance of safety oversight. Rights and obligations of the appointed organisation, tasks of the reviewed areas.
- Occupational health and work safety instructions, other procedures (radiation protection, document management procedures).
- Operating instructions. Description of the procedure related to the management of normal operating states, anticipated operational occurrences, design-basis and beyond-design-basis accidents (very low probability events). These operating instructions shall contain the detailed description of tasks performed by the main control room operators, technical support personnel and emergency preparedness organisation, as well as applied documents and control systems.

CHAPTER 14 – PLANT COMMISSIONING

This chapter has been prepared based on the commissioning experiences of VVER-1000 type reactors and Leningrad II Nuclear Power Plant being the reference unit for new power units of Paks II. project.

The chapter provides an overview of the entire commissioning process with the indication of commissioning phases and sub-phases. The chapter specifies tasks and responsibilities belonging to the scope of Paks II. Nuclear Power Plant Ltd. and Contractor for each individual phase during the commissioning.

The first phase during the commissioning of new power units implies the performance of tests on equipment and systems without nuclear fuel.

Commissioning works and activities to be carried out on the nuclear fuel-free reactor were divided into the following successive phases:

- Phase A – Preparatory works (activities performed prior to commissioning);
- Phase B – Functional testing of individual systems and equipment;
- Phase C – Testing prior to operation.

Nuclear fuel-based testing of systems and equipment can be commenced only after the completion of nuclear fuel-free tests and examinations, and confirmation of the equipment availability and unit systems compliance with the specified design parameters.

During the commissioning phases performed on the reactor containing nuclear fuel, the requirements of the Nuclear Safety Code pertaining to operation shall be applied, viz. the personnel, processes, procedures and supporting infrastructure of Paks II. Nuclear Power Plant Ltd. shall reflect the condition required for safe operation of the nuclear power plant:

First criticality related phases are as follows:

- Phase D – Loading of nuclear fuel and tests performed in subcritical conditions
- Phase E – First criticality and tests performed at low power levels

Reactor power start-up, including the following phase:

- Phase F – Tests of power operations

Operation of the power unit at rated power:

- Phase G – Trial operation
- Phase H - Guarantee tests

The chapter describes rules, procedures, test and examination programs necessary for the performance of commissioning works, including tasks and responsibilities of the process participants. The chapter provides information on provisional materials and fixtures necessary for the work performance, such as blind flanges, filters, measuring devices and structures.

The chapter provides the description of procedures related to the preparation and approval of documentation supporting the testing process, contains overview of change management and repair processes necessary in the course of testing, indicating the obligations of organisations responsible for the given process.

The chapter presents the Contractor's experiences gained during the construction and commissioning of other nuclear power plants and possibilities for the utilisation of these experiences in the commissioning activities at Units 5 and 6.

In addition, the chapter focuses on the description of organisation responsible for providing guidance during the commissioning, as well as on the description of teams, committees and authorities participating in the process, with the indication of their responsibility scopes. Operation and maintenance activities on equipment and systems are carried out by the Licensee's operating and, partly, maintenance personnel under guidance, supervision and responsibility of the Contractor, allowing the Licensee's personnel to gain skills and practical experiences for the subsequent operation of power units.

CHAPTER 15 – SAFETY ANALYSIS

Purpose of safety analysis

The primary objective of this chapter is to confirm the safe operation of the nuclear power plant, as well as the compliance of its systems and system elements to the requirements, by means of deterministic safety analysis. The tools for this analysis are model-based calculations carried out with the help of computer codes capable of modelling the nuclear power plant or certain parts thereof. According to the basic requirement, input data used for the creation of these plant models shall be consistent with the plant design, as the safety analysis has to provide information on the design compliance. In this case, the design compliance means that physical characteristics of the plant's systems and system components, as well as their operation modes and operating parameters are reproduced by analytical models with the highest level of accuracy. It is also important that these model-based calculations can be repeated either with the use of independent methods or alternative computer codes. Despite the fact that this analysis is conducted with the use of validated and verified software, i.e. it is ensured that reliable and reproducible result can be received in terms of the analysis objectives, the analysis still contains certain approximations, and the models assume certain designer's considerations due to the complexity of the nuclear power plant and its processes. Performance of independent analysis – in addition to the review of designer analysis by the Licensee and independent experts - provide further evidence of the result compliance. To assure the above, the chapter – in addition to the presentation of designer's analysis and their results – contains references to the information necessary for the independent replication of this analysis and check of compliance with the plant design.

Scope of safety analysis

Various changes in the operating states take place even during the normal operation of the nuclear power plant, and the maintenance of the adequate nuclear safety level is necessary. These changes in the operating states may be connected with the unit power increase and decrease, during which it should be confirmed by means of the above-mentioned model-based calculations that the behaviour of the nuclear power plant will be within the specified operating limits and conditions. These analyses pertaining to the normal operation are presented in a separate subsection of this chapter. At the same time, in terms of nuclear safety, special importance is given to the conditions and states different from those of the normal operation that occur in the course of the co-called postulated initiating events and represent failures or abnormal operation of the nuclear power plant. These states are particularly important since an improper response to these situations may lead to radioactive releases exceeding the allowed values. The chapter focuses mainly on the analysis of these postulated initiating events and presentation of the above-mentioned analysis data and considerations.

Whenever the nuclear power plant – under the effect of a postulated initiating event – enters the state of anticipated operational occurrences, it is necessary that the primary safety functions, i.e. reactivity control, heat removal and retention of radioactive materials, should be maintained at a proper level. Depending on the nature and severity of the initiating event, certain parameters of the nuclear power plant change in a manner triggering processes affecting nuclear safety, and in such a case, it is required that the situation should be handled as soon as possible to bring the power unit into the controlled safe state. After the initiating event, this requirement is fulfilled if reactivity control, heat removal from the reactor core and spent fuel pool, as well as the retention of radioactive materials are assured. If it is not possible to restore the normal operating state of the nuclear power plant, then the power unit should be brought into and maintained in the safe shut down condition during the required period.

Analyses described in this chapterdemonstrate that after all the initiating events, the nuclear power plant - following the developed initial state -is able to achieve the above-mentioned controlled condition and safe shutdown state along with the assurance of necessary sub-criticality, long-term and stable removal of residual heat, guaranteeing that the level of radioactive release will not exceed the limits established by the regulatory authority. The plant's response to the initiating event is determined by the characteristics of designed passive and active systems, conditions for their actuation and necessary operator interventions. In terms of the latter, it should be emphasised that during the performance of safety analyses, operator interventions can be taken into consideration only upon expiry of a certain time interval. Among others, interventions of the main control room personnel may not be taken into account within 30 minutes following the initiating event. Handling of initiating events should be done without operator interventions, thus, reducing the possibility of human errors and confirming the capability of the nuclear power plant to provide proper response in an autonomous manner with the use of design solutions.

The chapter examines a finite number of postulated initiating events. The list of these events is based on the Hungarian regulatory requirements, international recommendations, as well as on the design and operating experiences gained at similar nuclear power plants. The postulated initiating events can be caused by various reasons: they may occur due to internal or external hazards of natural or man-made origin arising on the plant site or in its vicinity, due to improper operation of the nuclear power plant caused by human errors or spontaneous failures of the plant equipment or systems. In practice, certain triggering causes may lead to the same initiating state: for example, a failure in case of a certain system can be caused by an internal fire or improper maintenance, thus, if the impact of these initiating events on the nuclear power plant is identical, then – based on the so-called enveloping principle – they are not handled as separate initiating events in this chapter. Anyway, this chapter considers over several hundreds of initiating events on more than five thousand pages.

In terms of analyses, the determining characteristic of initiating events is the frequency of their occurrence – it is also important to determine design and analysis considerations to be applied in respect thereof. Should the frequency of a certain initiating event exceed 10^{-6}/year, i.e. the frequency of occurrence is minimum once in one million years, then this initiating event should be handled as a part of the design basis. As the frequency of occurrence increases, the confirmation of nuclear safety in the analyses is carried out in accordance with more and more stringent requirements. Initiating events with lower frequencies of occurrence are covered by design extension conditions, in respect of which less stringent requirements and, correspondingly, different analysis methods can be applied; however, transition of the nuclear power plant into the controlled and safe shutdown state is assured and confirmed by calculations in their case, as well. In this latter case, the exception is connected with severe accident analyses examining special states with very low probability of occurrence, which lead to the nuclear fuel meltdown due to the damage of basic safety functions. It is important and confirmed in this case as well that melt fuel and other radioactive materials remain inside the containment and the environmental impact will not exceed the specified values. It should be noted that in respect of several initiating events, more stringent requirements are used in this chapter during the confirmation of safety than it would be required based on the calculated frequency of event occurrence.

Certain initiating events – based on their impact on the nuclear power plant – can be organised in the so-called representative groups, though in some cases there can be overlaps due to the enveloping principle mentioned earlier. The chapter presents certain analyses in the structure based on the categories of design basis- and design extension events and representative event groups (e.g. reactivity-initiated events, loss-of-primary-coolant events, events occurring during handling and transportation of fuel assemblies).

Compliance of safety analyses

An important and tangible objective of safety analysis is the confirmation of compliance of the final analysis results with acceptance criteria. These acceptance criteria are limiting values assigned to well-defined physical parameters, compliance with which assures the plant's safe response anticipated during the given operating state and maintenance of the environmental impact below the specified limit. Primarily, the acceptance criteria are connected to the protection of physical barriers – their integrity should be maintained by limiting maximum pressure and assuring cooling since – according to the defense-in-depth principle – these physical barriers are responsible for the prevention of radioactive material escape into the environment. The most stringent acceptance criteria should be used in case of events with the highest frequency of occurrence; however, the limitation of radioactive releases and compliance with relevant acceptance criteria guaranteeing the public safety is necessary under all operating states and conditions. For certain initial events, anticipated public exposure doses are determined with the use of specific propagation models, taking into account the spread of radioactive contamination not only within the containment and at potential emission points but in the environment, as well. The calculation of public exposure is described in a separate subsection confirming the retention of public exposure within the allowed limits for each operating state and condition of the nuclear power plant.

Special attention during the confirmation of compliance with the acceptance criteria is paid to conservative analysis considerations aimed at compensating potential analysis-related uncertainties and confirming safety for the most unfavourable initiating and frame conditions, assuring at the same time the highest reasonably achievable margins. Such analytical considerations may include selection of the most unfavourable parameters among those allowed in the normal operating state of the nuclear power plant or assumption of further failures of the systems necessary for handling the initiating events, not connected with the initiating event. These considerations may be identical for the analysis of all initiating events, e.g. through conservatism built into the computer model; however, specific considerations are needed in case of certain initiating events and acceptance criteria since things conservative in one case may work as less stringent conditions in the other one. The applied conservatism is described in subsections related to general analytical considerations and subsections connected with the analysis of initiating events.

During the analysis of very low probability initiating events covered by design extension conditions, the best estimate method can be used in accordance with the domestic and international experience in this regard; in this case, initiating and frame conditions pertaining to the analysis, as well as the availability of systems can be taken into consideration with the use of more realistic approximation. Analyses related to the design extension conditions demonstrate that meltdown of nuclear fuel in the reactor core can be prevented by the operation of safety systems designated for use during complex accidents and by relevant operator interventions. Despite the above, we should be prepared for managing severe accidents connected with core meltdown and serious reactor damages described in a separate subsection of this chapter. The analysis of severe accidents covered by the design extension conditions demonstrate that even in the case of reactor core meltdown, only certain limited precautionary measures are required in terms of the public, due to the design solution consisting in the availability of the last physical barrier - containment - assuring localisation and cooling of the melt fuel and retention of radioactive release, providing for the transition of power unit into safe shutdown state after the severe accident, long-term cooling of the melt core and prevention of further releases.

Based on the confirmation of the fulfilment of acceptance criteria connected with design basis-, design extension- or severe accident conditions within the framework of safety analyses, we can make an important conclusion in terms of the entire nuclear power plant. The results not exceeding limiting values of the safety analyses conducted in accordance with relevant analysis requirements provide evidence and confirm the compliance of safety systems with the established requirements and proper application of the defense-in-depth principle.

CHAPTER 16 – LIMITS AND CONDITIONS (TECHNICAL SPECIFICATIONS)

Pursuant to the Nuclear Safety Code, it is necessary that limits and conditions should be determined for systems and system components during the design process, the compliance with which will assure that the nuclear power plant can be operated in accordance with design objectives and nuclear safety requirements.

The operating limits and conditions should be determined in a manner that would guarantee that the compliance with these limits and conditions would prevent the occurrence of situations leading to design basis accidents or complex accidents and mitigate the consequences in case of potential accidents. Conservative approach should be followed for the determination of safety limits due to the necessity of taking into consideration the uncertainty of safety analysis.

In sense of the above-mentioned requirements, the chapter concerning Operating Limits and Conditions (OLC) summarises all such operating limits and conditions for systems and system components, the compliance with will guarantee that the nuclear power plant can be operated in accordance with design objectives and nuclear safety requirements.

The chapter consists of five subsections:

1. *Application*

The subsection describes the methodology and scope for the development of operating limits and conditions, rules for their modification and application scope. In addition, this subsection outlines the method applied for the determination of those limitations related to systems and system components that should be included into the OLC. The subsection contains the list of abbreviations and definitions used in the OLC, as well as the rules for the document application.

Since the OLC should contain clear and unambiguous instructions for the operators, a special format is chosen for the document development. The description of this format is provided in this subsection as well.

2. *Safety limits*

The subsection contains numerical values for safety limits presenting the upper limits of the relevant limiting parameters.

3. *Limiting conditions for operation, protection thresholds, personnel's actions and requirements for monitoring*

The subsection contains numerical values of limiting parameters and operating conditions for systems and system components. It also specifies requirements pertaining to inspection, maintenance and repairs assuring that the above parameters will remain within the acceptable limits guaranteeing safe operation of systems and system components.

The subsection unambiguously defines actions and measures to be taken in case of failure to comply with specified operating limits and conditions, indicating the moment to commence relevant interventions and time available for the action performance.

4. *Administrative requirements*

The subsection contains administrative requirements concerning the use of OLC. The subsection identifies the main control room personnel as primary users of the OLC, paying special attention to the importance of nuclear safety commitment during the unit operation.

At the same time, it is emphasised that all the personnel involved in the application of safe operating limits and conditions should be properly trained and prepared for their use.

5. *Bases*

The subsection presents substantiations and safety analysis that formed the base for the development of operating limits and conditions. All operating limits and conditions, as well as operator interventions should be substantiated by the presentation of relevant background supporting information. The subsection shall also describe operating modes and states relevant for the given operating limit or condition. At the current design stage, the scope of bases and substantiations cannot be full yet – details and parts missing at the current stage will be completed at the second stage of OLC development, i.e. at the stage of preparing the commissioning licensee application.

The OLC shall be applied for the following normal operating states of the nuclear power plant:

- power operations,
- power increase,
- hot standby condition,
- hot shutdown state,
- cold shutdown state,
- refuelling,
- tests and test conditions.

The development of the OLC is done in three steps; these steps are connected to the so-called facility level nuclear safety licensing procedure:

1. *Implementation License – Preliminary Safety Analysis Report*
   Preliminary demonstration of the operating limits and conditions, taking into consideration the Basic Design and Preliminary Safety Analysis Report.
2. *Commissioning License – Preliminary Final Safety Analysis Report*
   Determination and presentation of the operating limits and conditions (updated and modified in accordance with the project progress), taking into consideration the detailed design and analysis prepared for the Preliminary Safety Analysis Report.
   The document on the Operating Limits and Conditions to be provided for use by the operating personnel shall be prepared prior to the beginning of commissioning activities.
3. *Operation License – Final Safety Analysis Report*
   Detailed description and presentation of the reviewed and updated operating limits and conditions based on the commissioning experiences.

CHAPTER 17 – MANAGEMENT SYSTEM

It is required that a properly developed, reviewed and approved management system clearly describing the activities to be conducted at the given stage of the nuclear facility life cycle should be available throughout the entire life cycle of the concerned nuclear facility (in case of new nuclear power units these include site investigation- and analysis, design, construction and erection, operation, decommissioning).

It is necessary that relevant legislative requirements, in particular those specified in the Act CXVI of 1996 on Atomic Energy and Govt. Decree No. 118/2011. (VII.11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities (hereinafter referred to as Nuclear Safety Code) should be taken into consideration during the elaboration, operation and further development of the management system.

Prior to the transition to another life cycle, the Licensee shall demonstrate – in the form of an application for obtaining the next license – the awareness of requirements pertaining to the next life cycle and implementation of measures required for the fulfilment thereof.

In order to obtain the implementation license, the Licensee – in terms of the management system presentation - shall describe its project management system, addressing project-specific features. The presentation shall include both the management system of Paks II. Nuclear Power Plant Ltd. and management system developed jointly with the Contractor.

The chapter shall outline the developed regulatory structure and show how the relevant requirements of the Nuclear Safety Code are implemented and fulfilled within this structure.

Regarding the demonstration of the management system, all the life cycle stages shall be mentioned, including:

- design stage – as the activity preceding and substantiating the implementation period – with the demonstration of applicable requirements and their fulfilment,
- implementation stage – as the most decisive activity in terms of obtaining the license – with the demonstration of relevant requirements, tasks and preparedness for the task fulfilment.

CHAPTER 18 – HUMAN FACTORS ENGINEERING

HFE program objectives and scope

The subsection describes the objectives and scope of the Human Factors Engineering (HFE) program, as well as the organisational structure, tasks and responsibilities of the team responsible for implementing the HFE program, HFE program related processes and procedures.

Review of NPP operating experience

The subsection describes how the periodical review and utilisation of HFE-related deviations and events identified during the operation of nuclear power plants similar to the new power units to be constructed in Paks and reference units contribute to the enhancement of facility safety level and minimisation of risks connected with human errors. The subsection outlines the utilisation of international and domestic experiences covering safety questions related to human factors engineering. In addition to the review of already available operating experiences, the subsection provides information on the preparation for the periodical review of the plant's own operating experiences to be gained in the future.

Functional requirements analysis and function allocation

Functional analysis is conducted in order to assure the achievement of safety- and electricity generation objectives by the nuclear power plant. The analysis of functional requirements includes the review of operating functions, as well as the examination of potential and necessary automatic control functions at the nuclear power plant, which should be executed in order to assure the proper performance of operational functions. Automatic control functions are allocated between the "human" (main control room operators) and "system" (instrumentation and control system). Functional requirements analysis and function allocation are carried out in respect of all operating states and conditions of the nuclear power plant.

Task analysis

The purpose of task analysis consists in the evaluation of workloads of the main control room personnel, identification of complex tasks characterised by high human error risks and determination of control equipment, information and support necessary for the performance of tasks by the control room personnel.

Actions, information-, control- and task support requirements were defined in course of the task analysis process. The analysis scope covered representative and important tasks related to operation, maintenance, testing, supervision and communication, task-support requirements and operating conditions considered by the analysis. The task analysis identifies the activities to be carried out by the personnel during the performance of their tasks. The results of task analysis provide the basis for the design of human-system interface and define the requirements related to the performance of main control room personnel, taking into consideration human capabilities.

CHAPTER 19 – PROBABILISTIC SAFETY ASSESSMENT

In order to assure the compliance with radiation protection limits specified in respect of the population and plant's personnel, the Nuclear Safety Code – in addition to other principles and analysis – requires the performance of probabilistic safety assessment (PSA – Probabilistic Safety Assessment).

Description of probabilistic safety assessment

The Nuclear Safety Code contains numerical values in respect of various parameters forming the result of PSA analysis. Correspondingly, the main objective of PSA analysis is to examine the fulfilment /non-fulfilment of these criteria based on the basic design, to make conclusions based on the received results and to provide feedback for the design process. The PSA shall:

- define the unfavourable final state at the actual phase of an accident that should be avoided in terms of nuclear safety. In accordance with radiation protection objectives, final states should be understood as the damage of the last physical barrier retaining radioactive materials from affecting the population and personnel, first of all, as the damage of fuel rod cladding and as the damage of containment or its functions (and associated releases) protecting the environment from the radioactive consequences of occurred fuel rod cladding damage;
- identify the events leading to final states in case the designed equipment or systems do not perform their functions or human interventions are not made when necessary. These events can occur in the technology-related process (e.g. tank leakage) or these may be on-site or off-site hazards of nature or man-made origin (e.g. explosion of a fuel tank-car on site or tornado-like extreme winds). If we deal with a technology-related event – the so-called initiating event – then it is analysed as it is; however, if we identify hazards (e.g. fire in reactor building), then we should examine which complex of initiating events occurring concurrently and what kind of degradation in response systems can be caused by this primary event. Consequently, in this latter case, instead of a single initial event, we deal with a certain set of initial events;

- determine - on the basis of deterministic analysis modelling - the actual time-wise behaviour of physical processes – the chain of accident events (event trees) that may result from the occurrence of initial events capable of leading to final states. The analysis also identifies the conditions (success criteria) that – at a certain point - may turn the chain of events in a favourable or an unfavourable direction (e.g. whether the operation of all three designated pumps is necessary in an emergency situation or the operation of one is sufficient for the fulfilment of success criteria);
- specify equipment faults necessary for the fulfilment of the above success criteria, non-performance of necessary human interventions and create the so-called fault logic model (fault trees) containing logic connections allowing to predict the probability of failure to fulfil certain success criteria in case of the availability of relevant input fault data. Modelling of failures is based on the analysis of comprehensive component failure modes and consequences resulting from the non-fulfilment of anticipated system functions;
- create input databases necessary for the numerical evaluation (frequency of initial events, parameters of common-cause failures, deterministic system-related information, etc.);
- perform calculations for the full scope of all operating states and initial events (sets of initial events connected with hazards), quantify their results and examine them as to their sensitivity and importance;
- compare the received results with limits specified in the Nuclear Safety Code, evaluate the fulfilment of requirements, provide feedback to the designers in order to reduce risks and assure proper balance.

Types of analyses carried out by PSA

PSA Analysis Level 1 – refers to the analysis, in the course of which success is considered as bringing the power unit into a stable controlled condition as the outcome of an accident chain of events, without the violation of conservative criteria applied to the fuel rod cladding. In case of even a short-time violation of the criteria applied to fuel rod claddings, we deal with fuel rod damage, fuel assembly damage, core damage and core meltdown. Quantification of their frequencies is one of key tasks during PSA Analysis Level 1.

PSA Analysis Level 2 – refers to the analysis of event trees resulting in fuel rod damages in the chain of events within PSA level 1. These event trees accompanied with fuel rod damages are called severe accidents; the purpose of their analysis is quantification of the unit safety level (among others, estimation of frequency of off-site radioactive releases connected with severe accidents). This calculation starts from the fuel rod damage and plant vulnerability detected at the same moment of time. The analysis is completed with the determination of frequency values related to various running and volume of releases and supplemented by the comparison with probabilistic safety criteria specified in the Nuclear Safety Code, making conclusions and provision of feedback for the design process.

Reliability analysis of functions – analysis of the system function reliability. This type of analyses includes, for example, the analysis of failure to provide residual heat removal to the ultimate sink, the frequency of which should not exceed the value defined in the Nuclear Safety Code or analysis of the reliability of electrical or instrumentation and control systems, in respect of which such analysis is required by the regulation.

Conclusions on the fulfilment of the different probabilistic safety criteria

Based on the Preliminary Safety Analysis Report for Units 5 and 6, the following conclusions can be made in respect of the safety of the units, taking into consideration the comparison on the international scale:

1. The probabilistic safety assessment was conducted with the use of methods and data defined in the Nuclear Safety Code to determine the overall risk posed by the nuclear power plant, to confirm the fulfilment of respective risk-related objectives and acceptance criteria, to evaluate the design robustness and adequacy of design extension conditions. The probabilistic safety assessment confirmed:
    a. absence of such factors, initiating events, large potential sources of releases, such accident processes, plant systems, equipment or human interventions that would make a disproportionately large contribution to the overall risk;
    b. availability of a necessary margin to avoid the Cliff edge effect, i.e. there is no such a component related to safety, a small change in the parameters of which would lead to a sudden degradation of safety level.
2. The cumulative frequency of partial or full core melt accidents for all event sequences resulting from the postulated initiating events does not exceed the value of 10^{-5}/reactor-year defined as the acceptance criteria by the regulation; moreover, it accounts for only 2 % of the above value.
3. The frequency of losing the function of residual heat removal to the ultimate heat sink is less than 10^{-7}/reactor-year. Thus, the removal of residual heat to the ultimate heat sink is ensured properly.
4. In fact, we can exclude the probability of events accompanied by large or early radioactive releases. It means that there are no such event sequences leading to large or early radioactive releases and connected with postulated initiating events (set of initiating events) that would be more frequent than once in 60 million years – this value is significantly above the requirement specified in the Nuclear Safety Code, according to which the average frequency should be less than 10 million years.
5. The cumulative frequency of exceeding the criterion of limited environmental impact, taking into consideration all postulated initiating events and operating states is:

    $$2.20 \times 10^{-7} \text{ 1/reactor-year}$$

    The above value means that the frequency of inadmissible – in terms of the environmental impact - radioactive releases as consequence of an accident is as low (rare) as the frequency of any other event occurring once in 4,5 million years. This frequency value is lower than that of 10^{-6} 1/reactor-year specified in the Nuclear Safety Code.

To sum up, it can be concluded that at the stage of Preliminary Safety Analysis Report, PSA analyses confirm the fulfilment of safety objectives, in several cases – even with significant margins.

CHAPTER 20 – EMERGENCY PREPAREDNESS

The chapter describes compliance with nuclear safety requirements pertaining to emergency management in case of a nuclear accident that may potentially occur at the nuclear power plant. The emergency preparedness system of Paks II. Nuclear Power Plant Ltd. should be developed in accordance with the requirements of the Nuclear Safety Code and with respect to the provisions of the National Nuclear Disaster Management System.

The chapter provides the regulatory authority with detailed information to evaluate the emergency preparedness system of the nuclear power plant. Along with the project progress, the information and details concerning emergency preparedness will be further clarified both in respect of the tools and procedures to be used and emergency preparedness organisation.

One of the primary nuclear safety requirements concerning nuclear facilities is the creation of emergency preparedness organisation. The task of emergency preparedness organisation consists in the organisation and coordination of emergency preparedness activities, assurance of the availability of protection facilities and means to ensure the overall emergency management and protection of the operating personnel in emergency situations. The task of emergency preparedness organisation during the emergency is to assure the protection management and performance of on-site emergency activities.

The purpose of nuclear emergency preparedness in case of very low probability emergencies is to assure in terms of both the operating personnel and the public that deterministic health effects caused by ionising radiation can be prevented and stochastic health effects can be maintained at the level as low as reasonable achievable. Another purpose of emergency preparedness is the minimisation of social and economic impacts. In order to achieve the above objectives, the organisation should be prepared to implement the following tasks:

- emergency warning and alert;
- prevention and mitigation of damages caused by ionising radiation;
- prevention of deterministic health effects, limitation of stochastic health effects to the level as low as reasonably achievable;
- radiation protection control of the individuals and plant site, removal of radioactive contamination;
- timely implementation of on-site precautionary measures (isolation, iodine prophylaxis, evacuation, use of protective equipment), search for missing persons;
- technical interventions aimed at restoring the safe condition;
- protection of environment and material assets;
- first aid, care of radiation injuries;
- guidance of recovery works.

It is planned that the emergency preparedness organisation performing the above-listed tasks will be established and operated jointly by the neighbouring nuclear facilities (Paks II. Nuclear Power Plant, Paks Nuclear Power Plant and Interim Spent Fuel Storage Facility).

In case of an emergency, emergency preparedness organisation provides guidance from the Emergency Control Centre. The Emergency Control Centre will be equipped with computer workstations, communication tools, displays, data collection units for remote monitoring and measurements of the meteorological, radiation and radioactive contamination conditions.

In case of an emergency, the Emergency Control Centre is used also as an assembly point for the employees assigned to the emergency preparedness organisation. The on-site location of the Emergency Control Centre assures the possibility of an easy access in case of an emergency. In case of unavailability – due to any reason – of the Emergency Control Centre for the control of emergency response operations, its functions are transferred to the off-site Supplementary Emergency Control Centre located in the town of Paks.

Both the Emergency Control Centre and Supplementary Emergency Control Centre shall assure the fulfilment of the following functions:

- provision of information on the safety parameters of the nuclear power plant units;
- provision of necessary information for the identification and classification of the emergency;
- possibility of the examination of potential scenarios for emergency occurrence and further development, development of recommendations concerning potential emergency response actions;
- possibility of analysing long-term archives containing safety-related parameters of the nuclear power plant unit and databases containing data on radiation;
- provision of information for the radiation monitoring necessary in terms of the evaluation of actual and anticipated impact of on-site radiation and radioactive materials on the individuals and on the environment;
- provision of information on the decisions aimed at preparing and substantiating the recovery of the unit controlled conditions, at reducing, localising and mitigating the consequences of emergencies;
- provision of information to the personnel on the performance of activities aimed at emergency evacuation from the plant site;
- monitoring of emergency response activities;
- recording of important operating parameters.

Radiation monitoring system is installed in order to monitor the radiation situation being the consequence of emergencies occurring in the nuclear power plant; in order to monitor important process parameters during a severe accident, the plant is equipped with a severe accident measuring system. Radiation- and severe accident parameters can be traced in the operating control centre and emergency control centre of the nuclear power plant.

The basic document concerning emergency preparedness and emergency response at the stage of implementation licensing is the Nuclear Disaster Management and Response Plan (NBEIT). The Nuclear Disaster Management and Response Plant defines the operational procedures of the emergency preparedness organisation, means and requirements of warning and alert, developed precautionary measures, organisational structure and tasks of the emergency preparedness organisation, protection measures applied in case of emergency response participants and tasks related to emergency preparedness.

The employees assigned to the emergency preparedness organisation shall participate in regular trainings, special training courses and emergency response drills corresponding to their professional area. The employees of the nuclear power plant not assigned to the emergency preparedness organisation shall also attend trainings in order to be aware of personal protective measures, obligations and rules to be followed in case of an emergency.

CHAPTER 21 – PRELIMINARY DECOMMISSIONING PLAN OF THE NUCLEAR POWER PLANT AND ITS UNITS

This chapter provides the description of preliminary compliance with nuclear safety requirements pertaining to the period after the completion of a nuclear power plant operation and related to the nuclear power plant decommissioning. The chapter specifies basic principles related to decommissioning and describes the implementation of those in the preliminary decommissioning plan, which should be prepared at the stage of the nuclear power plant design and should be submitted as a part of the implementation license application. The preliminary decommissioning plan shall be re-evaluated and reviewed in order to assure that the decommissioning of the nuclear safety facility will be carried out in accordance with state-of-the art technologies taking into consideration up-to-date technical and scientific achievements and changes in the decommissioning strategies.

The main objective of decommissioning is to demolish the facility to assure the removal of radioactive materials, wastes, components and structures and to restore the original condition of the site prior to construction ("greenfield" decommissioning) or make the site suitable for industrial use ("brownfield" decommissioning).

The decommissioning plan is prepared by the Licensee and submitted to the competent regulatory authority for licensing. Actual demolition is preceded by a five-year long preparatory phase, during which all equipment and auxiliary systems required for continuous oversight, as well as those necessary for the disassembly of the reactor and reactor coolant system equipment (e.g. power supply, ventilation, air filtration) shall be maintained in operation.

In parallel to the above, the demolition of non-nuclear areas and facilities (e.g. turbine hall and other auxiliary systems) begins; debris generated through the crushing of concrete elements can be recycled (e.g. to be used as road pavement).

The next step implies demolition, cutting, melting and re-cycling of the inactive equipment (pipelines, tanks, valves) without radioactive surface contamination located within the containment and in other nuclear technology related buildings.

The next step is the decommissioning of equipment with radioactive surface contamination requiring special preparation and specific process operations. The removal/reduction of level of the radioactive contamination (decontamination) will be carried out with the help of remote control tools and manipulators.

The last step will cover the removal of highly contaminated equipment subjected to the impact of neutron radiation (e.g. reactor pressure vessel and reinforced concrete reactor cavity).

A greater part of wastes generated during the decommissioning belongs to the category of non-radioactive wastes and will be treated in a manner identical to that of standard industrial wastes. The final disposal of low-, medium- and high activity radioactive wastes is possible in the existing or newly built radioactive waste repositories.

The decommissioning process is completed with recovery of the original environmental condition or creation of conditions for industrial use and – following the control of environmental parameters – is exempted for further use.

ABBREVIATIONS

ALARA – As Low As Reasonably Achievable

ESWS –Essential Service Water System

GRP – Geological Research Program

HAEA – Hungarian Atomic Energy Authority

HAKSER – Regulatory Environmental Radiation Monitoring System

HFE – Human Factors Engineering

IAEA – International Atomic Energy Agency

INPO – Institute of Nuclear Power Operations

KKS – Kraftwerk-Kennzeichen System

kV – kilovolt

NSC – Nuclear Safety Code

OERP – Overall Emergency Response Plan

OKSER – National Environmental Radiation Monitoring System

OLC –Operating Limits and Conditions

PSA – Probabilistic Safety Assessment

PSAR – Preliminary Safety Analysis Report

RCM – Reliability Centred Maintenance

RPS – Reactor Protection System

SSR – Site Safety Analysis Report

ÜKSER – Operational Environmental Radiation Monitoring System

VVER – Water moderated water cooled power reactor

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