6. ENGINEERED SAFETY FUTURES

This Section provides an overview of the important safety related systems of the plant. This includes the reactivity control systems, the plant radiation protection system and the instrumentation employed to monitor neutronic and radiological parameters. The largest and most extensive system protecting the plant personnel and the environment from radiation in the case of an accident, is the Accident Confinement System (ACS). Also very important are the Emergency Core Cooling System (ECCS), the ECCS ensures protection of the core from overheating if a loss-of-coolant accident occurs in the primary circuit. The water Purification and Cooling System (PCS) provides not only emergency core cooling, but also continuous purification of the primary circuit water from chemical and radioactive contaminants during normal operation. These important systems are described in more detail in the following Sections.

6.1 MEASUREMENT OF REACTOR PARAMETERS

Effective reactor control, both in normal operation and in accident situations, requires comprehensive and timely information. This includes visual, electronically recorded and printed information about the operating mode of the reactor and the status of its components (fuel, control, and reflector cooling channels, graphite stack, metal structures, forced circulation circuit, etc.). This information is gathered and recorded by the parameter measuring systems.

Reactor parameters are measured by the following systems:

A) Fuel and control channel coolant-flow monitoring system. The system monitoring coolant flow through fuel and control rod channels consists of tachometric detectors and frequency-to-analog signal converters.
B) Graphite and metal structure temperature instrumentation system. The graphite stack and metal structure temperature measuring system utilizes thermocouples.
C) Fuel channel-integrity monitoring system (monitoring the temperature and humidity of the surrounding inert gas). The channel integrity monitoring system consists of gas temperature detectors, reactivity control system channel drainage temperature detectors, relative humidity detectors, and equipment to pump inert gas through the reactor core block (in case of increasing of gas temperature).
D) Reactor power density-distribution monitoring system. The reactor power (neutron flux) distribution system includes radial ionization chambers, fission chambers and in-core sensor of neutron flux density.
E) Fuel cladding-integrity monitoring system. The fuel cladding integrity monitoring system includes gamma scintillation spectrometers and equipment to move the spectrometers in the spaces between the steam pipes.
F) Main forced-circulation circuit parameter monitoring system. Operational parameters are displayed on the Main Control Room (MCR) using visual and recording instruments, annunciation windows plus mimic diagrams and printers.
G) Information computing system (ICS) for data acquisition, processing and graphical display. The information computing system is configured in a three-level hierarchy, with SM-1M and SM-2M computers and interface facilities.

The technical monitoring system is schematically illustrated in Fig. 6.1, and the main parameters which are monitored are listed in Table 6.1 [38]

Table 6.1 Main measured parameters [38]

Parameter	Number of measurement points
Graphite temperature	85 (17*)
Temperatures of top and bottom channel guide tubes	15
Temperature of top biological shield	11
Temperatures of roller supports, compensators, and rigidity plates of the top and bottom biological shields	22
Temperature of bottom biological shield	10
Outlet water temperature of the CPS channels	6
Drainage temperature of the CPS channels	235
Water temperature in radial reflector cooling channels	156
Temperature of the central hall floor, supporting metal structure, and steel shell	19
Temperature of reactor vessel	16
Temperature of gas mixture in the lines of the fuel channel integrity system	2052
Fuel element temperatures	64 (16*)
Temperature of the control rod servo drives	211
Gas mixture sampling, gas pressure in reactor space	4
Pressure of gas in the reactor space	4
Coolant flow rate in fuel channels	1661
Coolant flow rate in CPS channels	235
Humidity of gas mixture in the fuel channel integrity monitoring system distribution header	26
Axial neutron flux distribution	160 (20*)
Radial neutron flux distribution	252
Control rod position	211
Fuel channel activity	1661 (8*)
Pressure difference between reactor space and reactor vessel	1

* Number of instrumented channels

6.2 PLANT RADIATION PROTECTION

Special components and systems are provided in the plant for the protection of the plant and its environment from radiation, both in normal operating mode of the reactor, and in accident situations. The reactor emergency protection and control systems guard the reactor from overheating and the release of contaminated coolant or inert gas into the environment. The following systems are used for radiation protection and monitoring:

- control and protection system,
- fuel cladding integrity monitoring system,
- reactor emergency core cooling system,
- accident confinement system,
- plant liquid radioactive waste or gas-aerosol waste cleaning, removal or storage system,
- plant and environment radiation protection monitoring system,
- gas-aerosol and liquid waste monitoring system.

The bulk of the radioactive gases emitted by a nuclear power plant, consists of inert gases and radioactive isotopes of argon (Ar), krypton (Kr) and xenon (Xe). These isotopes are formed in the uranium fission process and, since they do not form any chemical compounds, some make their way into the cooling circuit. The contribution of these isotopes to the general background radiation is minimal (in general bellow 0.1 % of the natural background in the vicinity of the plant). At the Ignalina NPP, these gases are released to the environment through a 150 m high ventilation stack.

To reduce the emissions of radioactive gases, the plant uses a two-stage purification process. The first stage consists of a holding chamber, in which the activity of the gases is reduced by natural radioactive decay. The second stage uses special filters to absorb radioactive aerosols present in the gaseous effluent.

The power plant uses a closed water supply circuit. Liquid radioactive waste undergoes special processing. An automatic radiation monitoring system provides continuous radiation monitoring of aerosol and liquid waste discharge. The external dosimeter service of the plant uses dosimeters and radiometric and spectrometric instruments for the assessment of the radiation conditions in the environment.

6.2.1 Habitability Requirements

One of the primary goals of nuclear safety systems is to maintain secure control of the nuclear chain reaction at all times and to prevent the possibility for the formation of a critical mass during reloading, transportation and storing of nuclear fuel. The Ignalina NPP design incorporates systems for the detection of failure in all barriers to radioactive release and of systems which confine radioactive releases if they occur and mitigate their consequences.

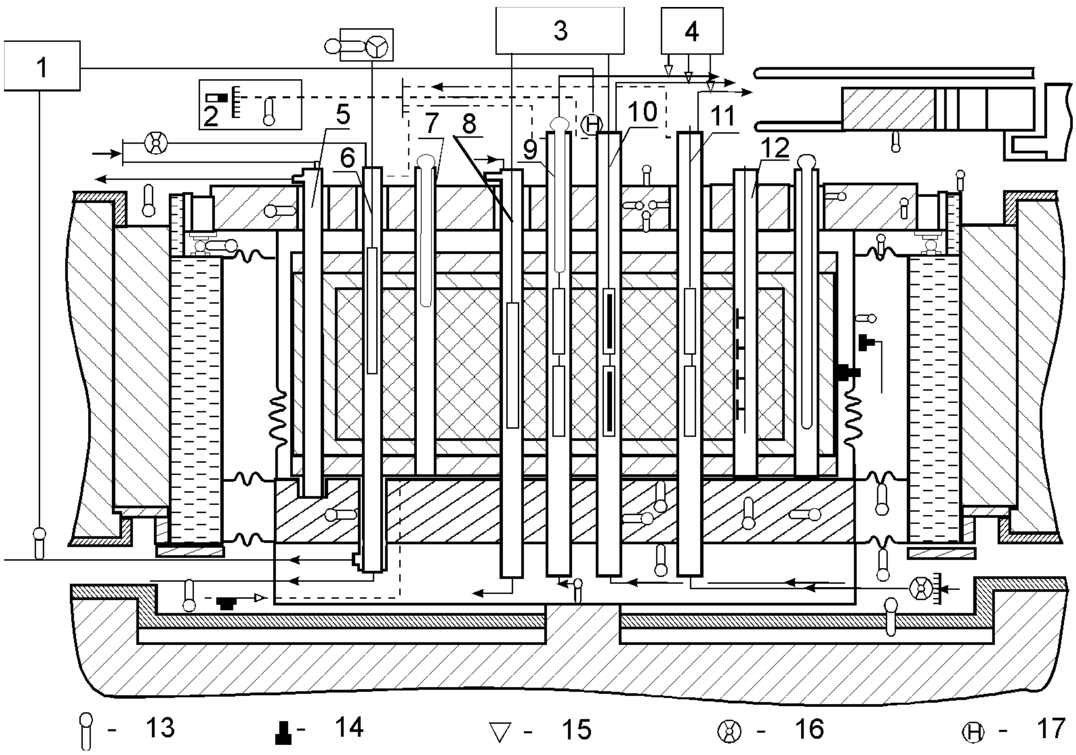


Fig. 6.1 Section of RBMK-1500 measurement parameters
1 - coolant leak monitoring system, 2 - fuel channel-integrity monitoring system, 3 - power density distribution monitoring system, 4 - fuel cladding-integrity monitoring system, 5 - radial reflector cooling channel, 6 - control rod, 7 - temperature channel, 8 - in-core sensor of the radial power density monitoring, 9 - temperature sensor, 10 - in-core sensor of axial power

density monitoring, 11 - fuel channel, 12 - fistula channel for gas sampling, 13 - temperature measurement, 14 - pressure measurement, 15 - radiation measurement, 16 - flow measurement, 17 - noise measurement

For the RBMK-1500 reactor of the Ignalina NPP the following barrier systems are provided:

- the innermost barrier is the fuel itself, and the leak-tight, corrosion-resistant cladding of the fuel,
- the next barriers are the piping of the reactor main coolant circuit, and the reactor biological shielding,
- finally, the RBMK-1500 is provided with a complicated partial containment system - the ACS.

Nuclear power plant states may be classified into operational states and abnormal events. The operational states include both normal operation and deviation from normal operation or anticipated operational occurrences. The normal operation is an operation of a nuclear power plant within specific operational limits and conditions including shutdown, power operation, shutting down, starting, maintenance, testing and refueling. Deviations are all operational processes departing from normal operation which are expected to occur once or several times during the operating life of the plant and which do not cause any significant damage to items important to safety nor lead to abnormal events.

Abnormal events refer to all fault conditions which lead to unplanned shutdown. Abnormal events relevant to safety, may be classified into incidents and accidents. Incident is an abnormal event, when the reactor safety systems are activated but allowing more or less immediate return to normal operation. Accident is a state defined under accident conditions or severe accident. Accident conditions are departure from operational states in which the releases of radioactive materials are kept to acceptable limits by appropriate design features.

Those departures of process parameters which can lead to an abnormal event during operation are considered as limits of safe operation. Table 6.2 presents the values of the RBMK-1500 reactor parameters which, if reached during operation on load, are considered as exceeding the operational safety of the plant.

Deviations are controlled by the reactor main operating and control systems. Necessary organizational measures are also established to prevent exceeding of the limit and violation of the requirements for safe operation of the plant. Automatic reactor protection devices against unallowable change of the parameters and failures of the power unit equipment are activated on signals and setpoints. When indications of deviation from process parameters occur, the necessary actions including reactor shutdown and power drop to a safe level are taken.

The main task of radiation safety systems is to minimize the exposure of operational personnel to ionizing radiation, and to assure that the release of radioactive contaminants to the environment is maintained within the authorized limits. Rules of radioactive safety [47]

Table 6.2 Limits of safe operation of Ignalina NPP with RBMK-1500 [36]

Parameter	Variation limit
Reactivity margin in manual control rods, number	< 30
Reactor power, MW(th)	> 4800
Fuel channel power, MW	> 4.25
Margin coefficient up to the heat transfer crisis in the fuel channel	< 1
Margin coefficient up to the linear load limit of fuel assembly	< 1
Calculated graphite cladding temperature, ^oC	> 760
Excess pressure in separator drum, MPa	> 7.95
Water flow in CRCC, m^3/s	< 0.256
Water flow in CPS channel with loaded rod, m^3/s	$< 8.3*10^{-4}$
Reactor and MCC warming rate, $^oC/h$	> 30
Reactor and MCC cooling rate, $^oC/h$	> 30
^{131}I activity level in MCC water, Ci/h	$> 2\cdot10^{-5}$

Table 6.3 Main dose limits [47]

Category	Group of sensitive organs		
	I	II	III
A - limit of NPP personnel to permissible dose, Sv/year	0.05	0.15	0.3
B - population dose limit, Sv/year	0.005	0.015	0.03

determine the dose limits and define accepted procedures for their evaluation. The following gradation of radiation safety principles are maintained:

- the allowed limiting doses are not to be exceeded,
- there must be a reason for any level of exposure,
- exposure doses are to be reduced to the lowest level practicable.

For allowable dose limits populations are divided into the following categories :

- category A - NPP personnel,
- category B - limited part of population, living near NPP,
- category C - general population.

Three groups of sensitive organs are defined in decreasing order of radiation sensitivity:

- group I - all body, gonads and red bone marrow,

- group II - thyroid gland, lungs, alimentary canal, liver, kidneys and other internal organs, which are not included in groups I and II,
- group III - skin integument, bone tissue, hand, forearms, ankles and feet.

The dose limits dependence on the group of sensitive organs for NPP personnel and the population living in the vicinity of the plant is shown in Table 6.3.
Measurements of radioactivity in the Ignalina NPP area and assessment of radio-nuclide content in food tested are performed by the Lithuanian Physical Institute, Vilnius, Lithuania and the National Hygiene Center of Lithuania. Past records show, that population doses arising from the operation of the Ignalina NPP are very low [15]. Based on the data of the laboratory for safety procedures at the Ignalina NPP, average exposure dose to the entire body of the plant personnel is about 0.004 Sv/year [15].

6.2.2 Examination Techniques and Procedures

The radiation monitoring system is used to ascertain that the radiation safety principles and norms are full filled. The Ignalina NPP radiation safety monitoring system includes the following monitoring types [2]:

- individual personnel monitoring,
- monitoring of radioactive material effluent in the environment,
- monitoring of environmental objects,
- monitoring of radiation level in the operational areas and in the Ignalina NPP territory,
- inspection of the radioactive waste collection and removal.

Personnel monitoring includes:

- control of internal and external body exposure dose,
- contamination control of hands, protective equipment, clothes, shoes and quality of their decontamination in the sanitary locks, workshops, laboratories and places for smoking.

The primary aim of personnel monitoring is the collection of information about exposure doses and prompt identification of exposure sources to assure that personnel over-exposure is prevented.

Monitoring of radiation levels in operational areas and in the Ignalina NPP territory includes control of radiation dose rates, noble gas and aerosol activity, activity and radionuclide content in gaseous and liquid release, radioactive contamination of equipment surfaces, etc. The limits of permissible release of radionuclides to the atmosphere and the annual allowable release of radioactive substances with the liquid non-radioactive wastes to lake Drūkšiai are established based on the conditions that the dose limits specified in sanitary regulations (OSP-76/87) [47] must not be exceeded. Values of allowable release of radioactive gases and aerosols for each 1000 MW power of the Ignalina NPP are given in Table 6.4.

6.2.3 Radiation Monitoring Instrumentation

The Ignalina NPP radiation monitoring system includes the following equipment [2]:

- automatic radiation intensity indicators,
- set of portable radiation monitors-radiometers for measuring X-ray, α, β and γ radiation within the following measurement ranges:
  α-radiation, part/(cm^2.min) $1\text{-}3\cdot10^4$
  β-radiation, part/(cm^2.min) $10\text{-}10^4$
  γ-radiation, R/h $10^{-6}\text{-}300$
- gamma-ray spectrometers,
- spectrometers for measuring body radiation doses,
- radiation spectrometers,
- radiometric and spectrometric equipment for determining the radio-nuclide content in various media and objects.

Control of plant personnel exposure doses is performed using individual thermoluminiscent indicators. Dosimeters are provided and serviced by the Radiation Protection and Industrial Safety Department Information regarding individual exposures are stored in computer data bases.

6.2.4 Scheduling of Inspections

Frequency of radiation monitoring inspections are specified in the monitoring schedules. They are approved by the Technical Director of the Ignalina NPP and signed by the local representative of the National Hygiene Center of Lithuania. Inspection of personnel radionuclide monitoring records at the Ignalina NPP personnel are conducted in the following cases [48]:

- beginning of employment,
- termination of employment,
- annually for each employee.

If accumulation over 0.05 x DS_A (DS_A is permissible contents of radionuclide in a sensitive organ) is found, the testing intervals are increased up to once per three months. Individual dosimetry inspection of operational and maintenance personnel are conducted once per month [49]. Personnel, who serve on preventive maintenance, are inspected before and after each preventive maintenance task.

Table 6.4 Allowable release of radioactive gases and aerosols from Ignalina NPP [36]

Nuclide	Allowable release
Radioactive noble gases, Ci/(day 1000 MW(e))	500

^{131}I, Ci/(day 1000 MW(e))	0.01
^{89}Sr, Ci/(month 1000 MW(e))	0.0015
^{90}Sr, Ci/(month 1000 MW(e))	0.0015
^{137}Cs, Ci/(month 1000 MW(e))	0.015
^{60}Co, Ci/(month 1000 MW(e))	0.015
^{54}Mr, Ci/(month 1000 MW(e))	0.015
^{51}Cr, Ci/(month 1000 MW(e))	0.015

The release of radioactive aerosols to the atmosphere, release of water to the environment, the concentration of radioactive gases and aerosols in the air of the operational premises, as well as the analysis of the γ-background and α-activity of steam of the separator drums is continuously monitored and is recorded once per day. Inspection of total α-activity in water of the MCC, water of the CPS circuit and water of the intermediate circuit are conducted once per day. Analysis of the contamination by radioactive substances of the Ignalina NPP territory, operational premises, special clothes and other safety related equipment is conducted according to a special schedule [49].

6.3 ACCIDENT CONFINEMENT SYSTEM

The reactors of the Ignalina NPP are protected by a pressure suppression type containment which, because of its specialized nature, is referred to as the Accident Confinement System (ACS). It differs from the standard pressure suppression containments in several respects. The major of which are :

- Not all of the primary cooling circuit is enclosed.
- It is made up of a number of semi-interconnected compartments.
- It utilizes ten separate pools of water for steam condensation.
- It includes a "clean air" venting system.

First three features are similar to the confinement system used in the VVR - 440/Mod213 Soviet built PWR's, the last feature seems to be unique. The original design documentation for this system was sparse, and a limited number of studies were available which analyzed the response of the ACS to accidental conditions. For this reason considerable effort was devoted to the determination of relevant ACS parameters and to the evaluation of ACS response to a broad range of DB and SB-LOCA events. Design parameters were determined from construction drawings and where uncertainties persisted, they were verified by direct measurement in the plant. The accident types evaluated include the entire spectrum of break sizes and break locations. Results of these studies are available in [86].

6.3.1 Purpose and Applicability

A characteristic feature of nuclear power plants built in the West is the containment, or protective shell. This is a large, especially strong, steel and reinforced concrete building, usually semi cylindrical in shape, which encloses the reactor and its cooling circuits. The Ignalina NPP does not have such a containment structure. Almost all components of the power plant are located in large, interconnected, traditional buildings. This external image of the power plant has contributed to the widespread opinion that the Ignalina NPP has no containment system. That is not true. In fact, both the reactor and a large portion of the cooling circuit are enclosed in reinforced protective shells. In addition, a complicated large scale system is provided whose purpose is to condense the steam erupting from the cooling circuit during a possible accident. This equipment takes up a significant fraction of the volume of the central power plant building.

The Ignalina nuclear power plants are cooled by high pressure water. While the reactor is on line, most of the thermal energy is found, not in the core, but in the cooling circuit. Therefore, if an accident occurs which damages the cooling circuit (pipe break, valve stuck open, etc.), the main function of the containment system is to prevent the released high pressure steam-water mixture from reaching the atmosphere. Generally, there are two containment types which achieve this result.

The most common type is called "dry containment". This is also the simplest type: a sufficiently large and strong building, which can withstand the pressure reached when the entire contents of the cooling circuit is suddenly lost from the circuit (within half a minute). In this case, it is possible for the pressure to reach 4 atmospheres. Containment buildings are correspondingly large and expensive.

The second, or "pressure suppression" containment type uses additional equipment which condenses a part of the released steam to reduce the peak pressures which can be reached during an accident. Internally, the containment structure is divided into at least two major volumes, which are interconnected by a flow path submerged under a permanent pool of water. When the steam enters the first section, the pressure rises, and in order to reach the second section, the steam-air mixture must "bubble" through the water. This condenses most of the steam, reducing the peak pressure attained.

The system used at the Ignalina NPP [50] belongs to the second category of containment. It differs from containment systems common in Western nuclear plants in several important aspects:

- Not all of the cooling circuit (only about 65 % by water volume) is enclosed in the containment.
- The containment is made up of a large number of compartments.
- Condensation of steam occurs in ten water pools which are separated into two groups of five each.

- A controlled venting system is present. This system is designed to remove part of the clean air after the start of an accident. It serves to reduce both the peak pressure and the long term plant-to-ambient pressure difference.

In many other respects the Ignalina NPP ACS has features similar to traditional pressure suppression systems used in the West. Spray nozzles are located in several regions where steam can accumulate, It has provisions for condensing steam released during operational transients (e.g. opening of main safety valves) and the spray water is cooled by the condenser tray cooling system.
The ACS performs its function of limiting pressure buildup and containment of potentially radioactive gases released from the primary system in the following way:

- most of the steam produced by flashing of the break flow is condensed in the condensing trays. This reduces the pressure and, thus reduces the release of radioactive material to the environment,
- radioactive species released during the accident are held in the enclosed compartments until they can be decontaminated,
- during the first stage of the accident, a part of the uncontaminated air is vented from the ACS,
- spray systems are used to condense the steam in the remaining ACS compartments.
- a condenser tray cooling system is employed to cool the pool water and condenser which is provided to the sprays.

The venting of uncontaminated air and the use of sprays makes it possible to lower the ACS pressure with respect to the atmospheric pressure. This reduces the probability of release of radioactive material to the environment. In addition the ACS serves as a water reservoir. The stored water can be used for emergency core cooling, as well as to condense steam released by the pressure relief valves.

6.3.2 Design Characteristics of the ACS

An overall schematic of the ACS representing a cross-section through the main power plant building is presented in Fig. 6.2. As shown in the diagram, a significant fraction of the total volume of this building is assigned to the accident confinement system. Its principal components are two interconnected ACS towers, each of which contain five vertically-positioned condensing trays. The water in the pools of these trays, is the medium which serves as the first barrier to the steam emitted during a LOCA event. The water of the pools forms a barrier which divides the entire system into two distinct parts:

1 - The compartments in front of the pools. They encompass the primary circuit piping where a potential break and the release of high pressure water is most likely.

2 - The compartments beyond the condensing pools. The non-condensibles present in the first part of the ACS are pushed into these compartments. Most of the released steam is condensed as the steam-air mixture bubbles through the condensing pools.

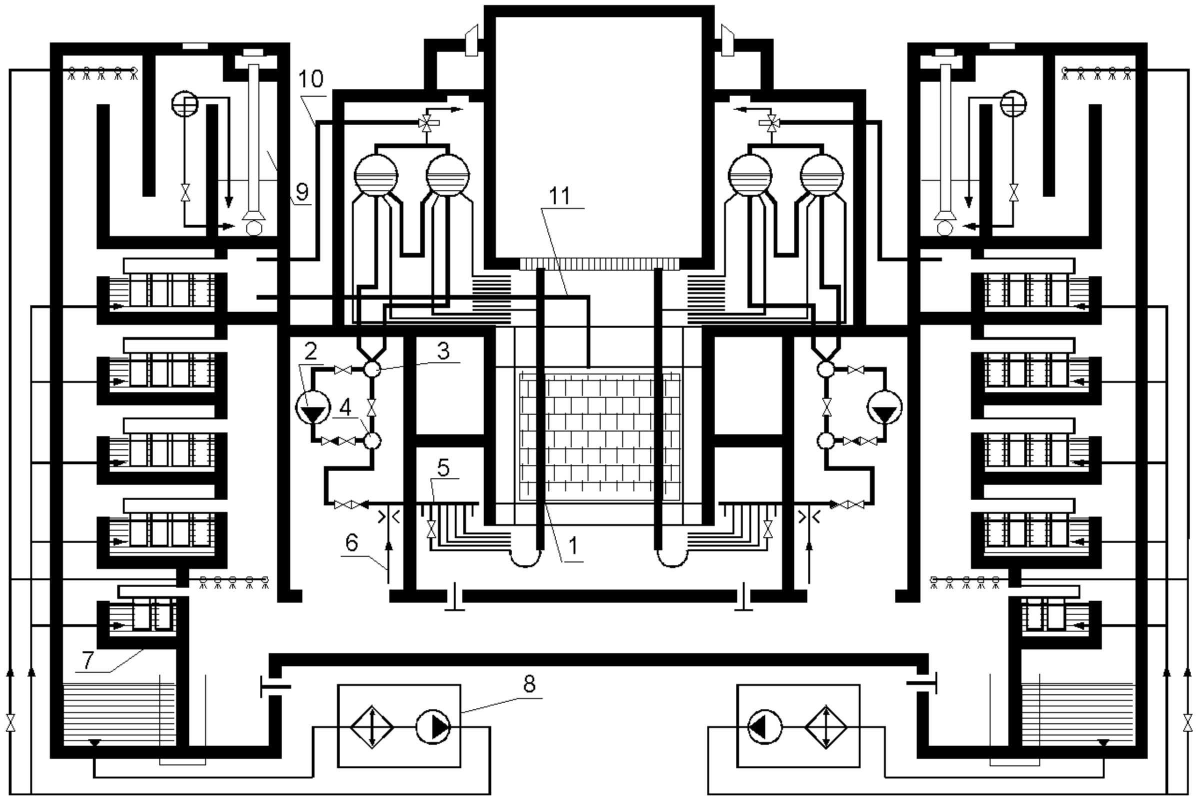


Fig. 6.2 Principal ACS schematic
1 - fuel channel, 2 - main circulation pump, 3 - suction header, 4 - pressure header, 5 - group distribution header, 6 - ECCS header, 7 - condensing pools, 8 - ACS condenser tray cooling system, 9 - air discharge pipe section, 10 - pipe for removal of contaminated steam from protection valve, 11 - pipe for removal of contaminated steam from broken fuel channel

The compartments which surround the reactor and the components of the primary circuit, are reinforced and capable of withstanding elevated pressures. Their design pressure is shown in Table 6.5. They can be divided into the following two subgroups:

- the compartments containing pipes which lead into the reactor, and the main circulation pumps. Steam released into these compartments is directed to the four bottom condensing pools of both ACS towers,
- the core block itself, and the pipes leading out of the reactor. Steam released into this zone is directed to the highest (fifth) condensing pool of the left hand side of the ACS tower.

The principal design features of the Accident Confinement System are presented in Table 6.6. The capacity of system is designed to isolate the steam escaping from the primary circuit in case of the Design Basis Loss of Coolant Accident (DB-LOCA) which in the case of an RBMK plant is taken to be the break of the pressure header. The design capacity of the ACS in this accident situation is shown in Table 6.5.

6.3.3 ACS Structural Characteristics

The structures housing the accident confinement system are divided into two parts, which are located in separate towers (Fig. 6.2). This feature was dictated by the plant layout requirements. However, both halves of the system function simultaneously, and analysis show that they receive an approximately equal portion of the released steam, except for one accident type, which will be discussed below.

The bottom section of the towers contain the steam reception chamber, the steam corridors, and the hot-condensation chamber. In case of an accident, the steam flows through these compartments and subsequently up to the steam distribution channels, which are positioned vertically on both sides of the tower. The lower section of the hot-condensation chamber stores the water reserves for the emergency core-cooling system and serves as a collection pool for the condensed steam during a LOCA event. The hot- condensation chamber is connected to the steam reception corridors by ten overflow

Table 6.5 Principal ACS design characteristics [50,51]

Parameter	Quantity
Maximum accident steam flow into the system, kg/s	14700
Condensed steam before starting the spray system, kg	30300
Discharged clean air, kg	25000
Maximum water temperature, ^oC:	
- pools 1 - 4	88
- pool 5	98
Maximum permissible excess pressure (MPa) in the:	

- reinforced leaktight compartments	0.3
- steam reception chamber	0.1
- bottom water pipes compartment	0.08
- gas-holding chamber	0.08

pipes. The purpose of these pipes is to direct part of the condensate to the steam-reception corridors. On the corridor side the pipes have hydrolocks to prevent steam from flowing into the hot condensate chamber.

The hot-condensate chamber is joined to an air removal channel to the gas-holding chamber, which is located in the upper part of the ACS tower. This channel conducts the air forced from the hot-condensate chamber and the condensation pools to the holding chamber. The spill-over water from the condensing trays falls through this channel to the bottom of the hot-condensate chamber sump.

Sprays are used in the top part of the reception chamber, and also in the gas-holding chamber, to condense the steam which remains in the ACS compartments. The over-pressure valve is built into the wall of the reception chamber, their purpose is to release air from the compartment of heat exchangers of condensate tray cooling system to the reception chamber, when the pressure there decreases below atmospheric levels.

Five concrete trays holding condensation pools are located above the hot-condensate chamber. The pools contain the water reserves needed for condensing the steam released during a LOCA event. A schematic diagram of the first four condensation pools (numbering from the bottom) is shown in Fig. 6.3. The space above the pools is connected by air passageways to he gas-holding chamber. The height of the overflow barrier (2.1 m) was chosen, by considering the expansion of the water caused by the bubbling of the steam. To maintain the initial water level of 1.1 m, there are 100 mm holes. Rectangular holes, distributed at a height of 1.2 m, allow the condensate and hot-water to overflow and spill into the hot- condensate chamber.

Table 6.6 ACS components [50,51,62]

Parameter	Quantity
Volume (m^3) of water in condensing pools:	
- pool 1 - 4	2860
- pool 5	745
Volume of cooling system water, m^3	1000
Initial water temperature in the pools, ^oC	30
Volumes (m^3) of ACS facility:	
- compartments containing main circulation pumps, suction and pressure headers, ECCS headers (reinforced leaktight compartments)	2 x 6000
- compartments up to the ACS towers (reinforced leaktight compartments)	4000
- compartment enclosing the pipes leading to the fuel channels and spaces below the reactor	4200
- summed volume of the steam receiving chambers in the ACS tower, and the steam distribution headers in the bottom four condensing pools (volume to the top of the barbotage)	2 x 2300
- total spaces above the water surface in the condensing pools, the air venting channel, and the channel to the first gas holding chamber	2 x 4935
- gas-holding chambers, not including the channel to the first chamber	2 x 10170

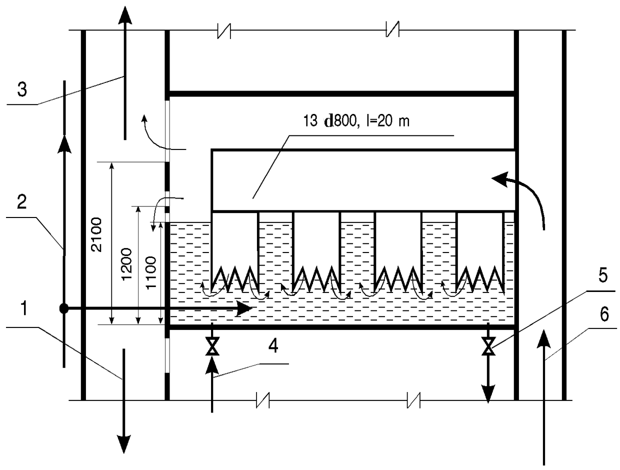
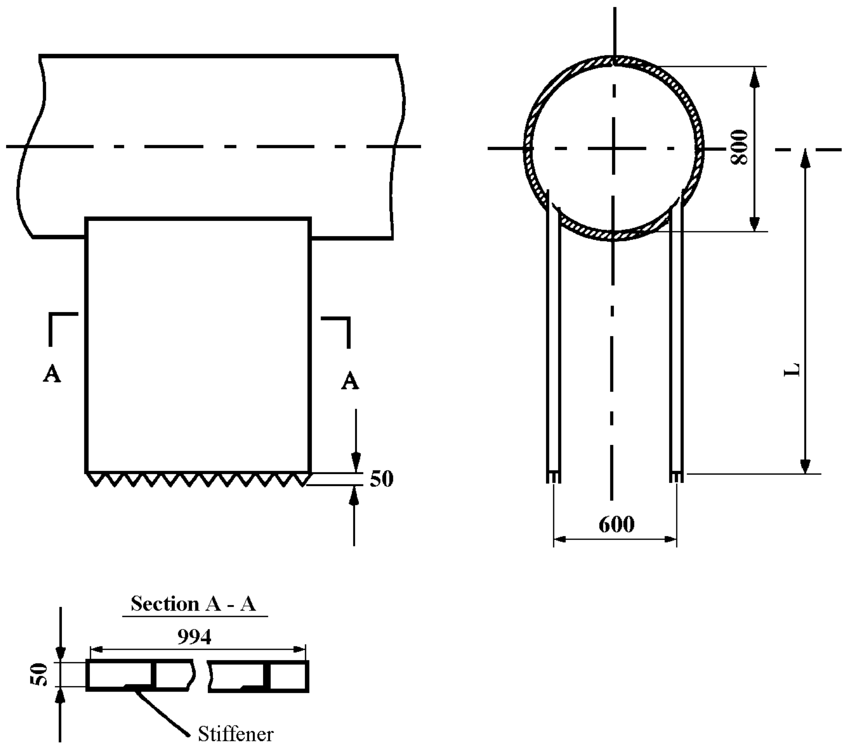


Fig. 6.3 Condensing compartment and pool
1 - condensate overflow to hot-condensate chamber, 2 - cooled water supply, 3 - air and non-condensing gas removal to gas holding chamber, 4 - water supply for filling first tray, 5 - water to purification system, 6 - contaminated steam from steam reception chamber

The bottom four condensation pools are intended for the condensation of steam from the compartments in which the main circulation pumps and the group distribution headers are located. They are al similar, except for the bottom pool which is somewhat smaller (see Fig. 6.3). Part of the cross-section of the bottom pool is taken up by the steam reception chamber. In each pool, there are 10 steam distribution devices, each about 20 m long. The bottom pool has 7 devices 20 m long, and 3 devices 10 m long. These are 800 mm diameter pipes connected to rectangular, sheet metal downcomers which have a total height of 1.2 m and are submerged to a depth of 0.9 - 1.05 m in the condensation pool water. A scaled representation of the vent and the rectangular channels including a summary of the geometrical characteristics of vents and downcomers is presented in Fig. 6.4. There are 10 headers per tray, this results in a total flow area of 11.93 m^2 for condensation trays 2 to 4. At the exiting end the rectangular downcomer channels are provided with a saw-tooth edge in order to better distribute the steam and to reduce condensation type oscillations. The steam distribution devices connect to the steam distribution channels, and through them to the steam reception chamber.



Condensation pool	Length of devices, m	Number of steam distribution devices in both ACT	Downcomers length L, m	Downcomers number
1	19.9	14	1.6	336
	9.95	6	1.6	72
2 - 4	19.9	60	1.6	1440
5	8.5	38	2.29	456
	7.5	2	2.10	20

Fig. 6.4 Sketch of steam distribution devices

The fifth (top) pool (Fig. 6.5) is intended for the condensation of steam from the pressure relief valves. 600 mm diameter pipes from the SDV-A and MSVs conduct steam into the rectangular chambers in the center of the pool. The structure of the steam distribution devices is analogous to those described earlier. The rectangular downcomers are connected to each other through holes in the columns.

The fifth pool of the left ACS tower is equipped with an additional 23 m long steam distribution device, which has downcomers submerged 1.5 m deep under water. This device is intended for the condensation of steam released during a potential rupture of a fuel channel. Steam is transported to this device from above, by a 600 mm diameter pipe. All of the condensing pools are equipped with cold water distribution headers. Water is provided to these headers from the condensate tray cooling system.

The compartment for confining potential "contaminated" gases is located above the condensation pools, Fig. 6.6. This compartment is divided by vertical partitions into seven channels, positioned in such a manner that air, or other non-condensing contaminated gases coming from the pools during an accident, would pass through them sequentially. A grating is installed in each channel in order to even out the flow along the cross-section.

Ten 800 mm diameter pipes, through which clean air is released into the atmosphere are located in the last compartment. The entrance end of these pipes reaches into a separate section of the compartment. Hollow metal spheres are placed at the bottom of each pipe. In their nominal position they lie on the bottom so that there is a 300 mm space between the ball and the pipe. The water level in the section is kept at about 300 mm. This is adequate for covering the ends of siphons coming from a 100 m^3 storage tank located in the top part of the gas-holding chamber. The volume of the storage tank is filled with water and it is connected by a 1500 mm diameter two pipes to the steam reception corridor. In the event of

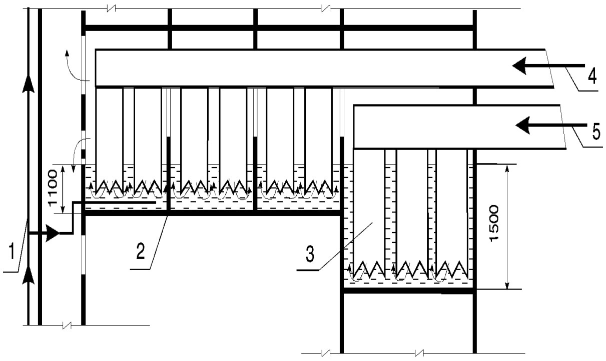


Fig. 6.5 Fifth steam-condensing pool (left ACS tower)
1 - cooling water, 2 - section for condensing steam from protection valve, 3 - section for condensing steam from fuel channels, 4 - steam pipe from protection valve, 5 - steam pipe from ruptured fuel channel

a LOCA which produces a pressure rise in the inner ACS compartments, a pressure pulse is transmitted through this pipe to initiate the flow of water in the siphon.

The top part of the gas-holding chamber is provided with a set of sprays. In the event of an accident which produces moderate steam releases over a long time period, their purpose is to reduce the partial pressure of the steam in the gas holding chamber and to cool the compartment. Water is provided to these sprays from the condensate tray cooling system.

6.3.4 Condenser Tray Cooling System

The Condenser Tray Cooling System (CTCS) is designed to remove the energy deposited in the condenser tray pools by condensation of LOCA generated steam, and to maintain the prescribed water reserves in the condensing pools during normal and off-normal plant operation. During long-term transients the CTCS becomes the major sink for the removal of the decay energy released as steam into the ACS.

The CTCS is located in enclosures beyond the re-enforced compartments of the ACS. A representation of the CTCS serving one of the ACS towers is shown in Fig. 6.7. The tower itself is shown schematically in the upper portion of the figure, a diagram of the main components of the CTCS and their inter-connections is shown below. The main CTCS components are:

- a set of eight shell-and-tube heat exchangers (four on one side) of the type designated as 1200 TNG-1-6-M1/20-4-1. While the reactor is on line, one heat exchanger can be taken out of operation for maintenance. Heat transfer surface area measures 426 m^2,
- six type D 1250-65 electric pump units with a capacity of 1250 m^3/h delivering a hydrostatic pressure of 6.5 kgf/cm^2 each,
- a complement of piping and valves.

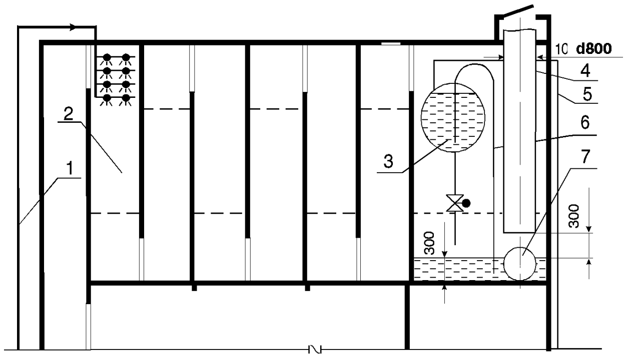


Fig. 6.6 Gas-delay chamber
1 - cooling water, 2 - section of gas-holding chamber, 3 - 100 m^3 water tank, 4 - air discharge to environment, 5 - pipe transmitting the pressure pulse, 6 - siphon, 7 - floating ball

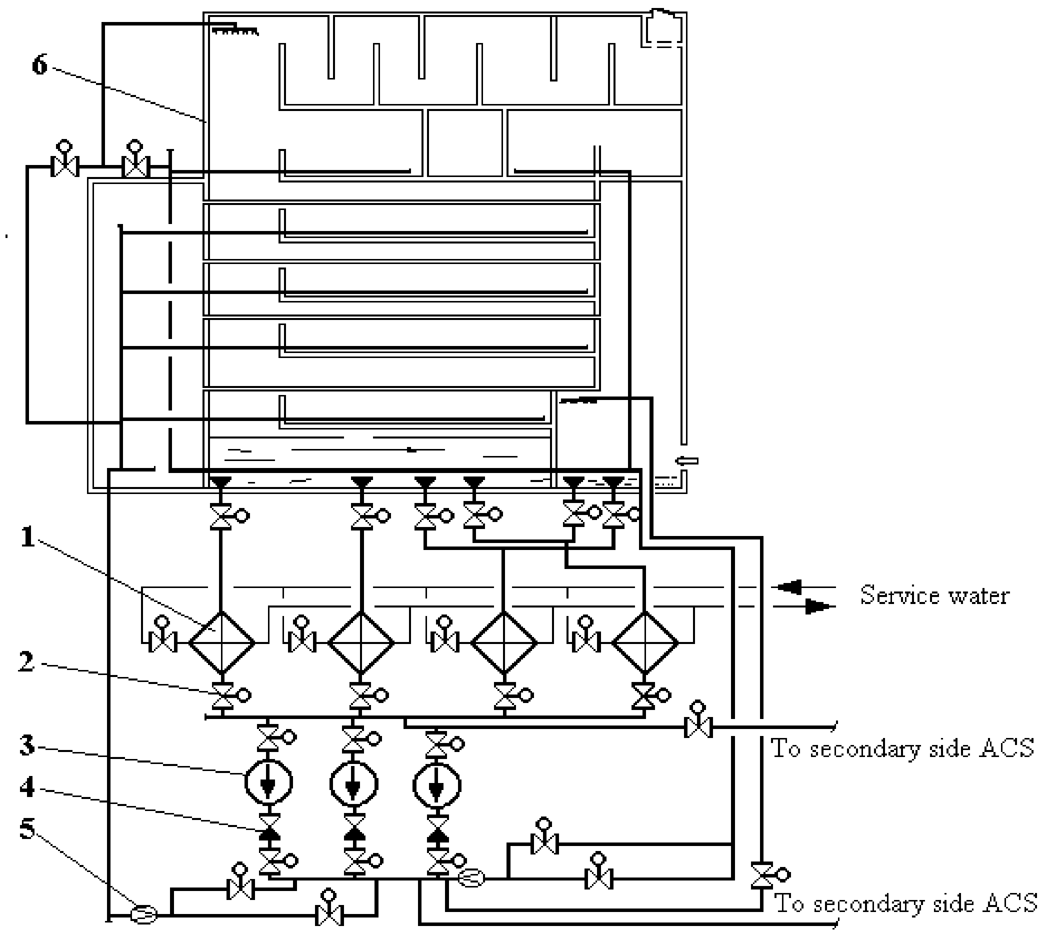


Fig. 6.7 Condenser tray cooling system
1 - heat exchanger, 2 - valve, 3 - water pump, 4 - check valve, 5 - mass flow rate gauge, 6 - ACS tower

As shown in Fig. 6.7, the main water supply is the ACS Hot Condensate Chamber (HCC), water is drawn from the HCC pool through five 400 mm diameter pipes. The valves provided on the suction side of the system are normally maintained open. To ensure reliable operation in the case any two pumps are not available, a 600 mm interconnection is provided at both the pump suction and pressure sides. An alternate water supply is the sump of the Bottom Steam Reception Chamber (BSRC) which receives the overflow condensate from the HCC and the runoff from the BSRC sprays. At both the inlet and the outlet of the heat exchangers valves are normally maintained open.

The CTCS pumps deliver water to the following systems of the ACS Tower:

- to condensing pools 1-4 by means of 600 mm diameter pipes which in the distribution net narrow down to 300 mm and finally to 150 mm,
- to the level 5 condensing pool via a 600 mm diameter pipe which at the distribution end narrows to 400 and 200 mm diameter,
- to the BSRC sprays by means of 200 and 100mm diameter pipes,
- to the gas delay chamber sprays through 200mm and 100 mm diameter pipes.

In Fig. 6.7 the lines supplying the sprays are indicated in the upper left hand and lower right hand corners of the schematic representing the ACS tower.

The secondary side of the ACS heat exchangers is supplied with service water via a 1200 mm header at a flow rate of 8400 m^3/h. The minimum flow through any heat exchanger is rated at 1500 m^3/h. Operator control of the pump units is maintained from the MCR. In case auxiliary power is lost, the pump units and valves are fed off the emergency power sources.

The operation of the ACS condenser tray pool cooling system is initiated when the temperature of water at the four bottom pools reaches 35°C or at the fifth pool - 50°C [88]. This system can also be activated by the ECCS start up signal two minutes after initiation of a LOCA.

For normal power operation, in addition to the outlined CTCS, a separate pump system is available. It is designed to maintain the prescribed water inventory in the level #5 condensing pools and has the capability to exchange water between the two ACS towers during scheduled maintenance. This system consists of two X90/85 K-1-2G type pumps with a capacity of 90 m^3/hour generating a hydraulic head of 8.5 kgf/cm^2. If the situation warrants it, the auxiliary deaerator makeup system pumps can also be used to supply water to the HCC pool. This system consists of 4 pumps and a 1500 m^3 water tank. In addition, the deminiralized water distribution system which includes 3 pumps and 2 tanks of a 5000 m^3 capacity each can be switched to supply water both to the HCC pool and to the auxiliary deaerator makeup system.

The availability of the CTCS is one of the requirements of normal plant operation. Thus operating procedures require the shut down of the reactor unless the following equipment failures or deviation from the design norms can be corrected:

- 5 minutes after a water level in one ACS condensing pool drops below 950 mm, or water temperature rises to 35 ^oC at condensing pools 1-4 or to 50 ^oC at pool 5,
- 30 minutes after water temperature at a level 5 condensing pool reaches a value 50 ^oC while an main safety valves opens and subsequently fails to close, or 2 minutes after any two main safety valves open and subsequently fail to close despite of corrective actions,
- 8 hours after 2 pumps of the condenser tray pool cooling system fail to operate,
- 2 hours after 2 diesel generators fail to automatically take a load, or immediately unless the remaining diesel generators are able to start up 4 pumps of the condenser tray pool cooling system.

6.3.5 Reactor Cavity Overpressure Protection System

Protection of the reactor cavity against overpressurization is an important part of the safety system of RBMK-1500. This system is intended to ensure reactor vessel leaktigthness and integrity of reactor metal structures under accidents caused by a rupture of a fuel or special channel, with resulting pressure growth in reactor cavity. The protection system provides:

- fast scram by the signal for the increase of the reactor cavity pressure over the setpoint 7.36 kPa (750 kg/m^3),
- immediate manual rector shutdown by the operator in case integral water leakage in reactor cavity exceeded 10 kg/h,
- removal of steam-gas mixture from reactor cavity to the condensation system by the steam-gas release circuit in case of a rupture of fuel or special channel.

A simplified schematic of reactor pressure relief system is shown in Fig. 6.8. In the reactor pressure relief system, a steam-gas mixture is discharged from the top of the reactor cavity through four 300 mm diameter and one 600 mm diameter pipes located along the periphery of upper metal structure. The pairs of 300 mm diameter pipes and 600 mm diameter pipe are connected to four 400 mm diameter pipelines that penetrate the boundary of ACS. Within the ACS, each pair of the 400 diameter mm pipes is connected to one 600 mm diameter pipeline which terminates at the header of the steam distribution devices within the fifth suppression pool, and they also connected to one group of the three membrane safety devices (350 mm diameter) within the ACS. The bottom pipes terminate in the second group of the three membrane safety devices within the ACS. When the membrane safety devices perforate (i.e., if setpoint of 59 kPa is exceeded), the steam- gas mixture is discharged into the ACS suppression pools 1 to 4.

The requirement for overpressure protection system is to have the capacity for coping with design basis accident involving one pressure tube rupture. Cavity pressure exceeding 0.314 MPa has been described as having the possibility to lift the upper-head shield assembly breaking the reactor seal, the pressure tubes, and affecting the operation of other safety functions.

The efficiency analysis of the reactor pressure relief system installed at Ignalina plant in 1996 showed that the capacity of the existing system makes it possibly to withstand multiple rupture of up to 9 pressure tubes under a conservative assumption of simultaneous guillotine ruptures.

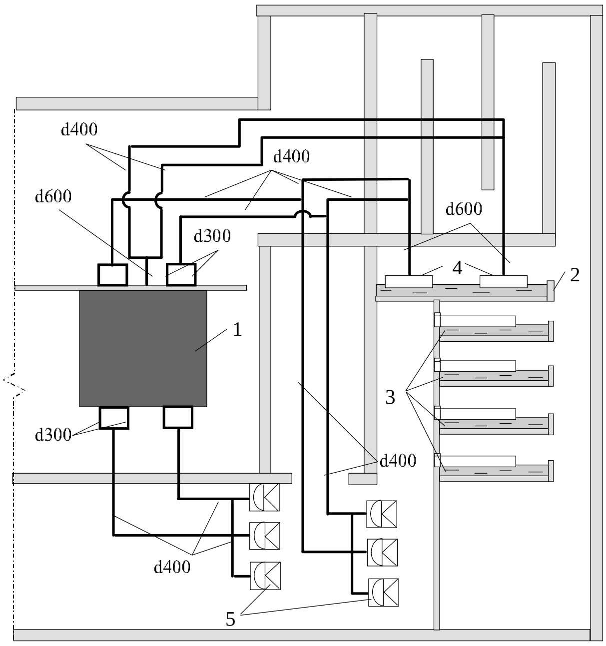


Fig. 6.8 Simplified schematic of the reactor pressure relief system

1 - reactor, 2- the fifth ACS suppression pool, 3 - suppression pools 1-4, 4 - steam distribution devices, 5 - membrane safety devices (350 mm diameter)

6.3.6 Description of ACS Response

The reactor cooling circuit of an RBMK-type reactor is rather complicated. The core is cooled by 1661 separate channels. These provide steam to four steam separators. The water and condensate mixture from each separator is returned to the core by three (of four) pumps. Thus the system consists of pipes of various diameters. The impact on the accident confinement system, i.e. the quantity of steam and hot water released, will depend on the size of the broken pipe. To be conservative, it is assumed that it is the biggest pipe in the in a given piping section. The DB-LOCA event, that is the largest break for the entire system is a complete guillotine rupture of the 900 mm diameter pressure header.

6.3.6.1 Accident in the Reinforced Leaktight Compartment

The ACS response to a loss-of-coolant-accident depends to a marked degree upon the location and the nature of the break which leads to primary system coolant loss. The DB-LOCA analysis assumes the rupture of a Pressure Header (PH) (4) (see Fig. 6.2) The initial blowdown rate of the steam produced by such an event is about 14.7 t/s [38]. From the break compartment the flashed vapor and compartment air first flows through the steam-release tunnels to the BSRC of both ACS towers. From there, it passes along the corridors leading to the steam distribution headers of the bottom four condensation pools and is forced to bubble through the pool water. During barbotage the steam condenses, while the air and other non-condensable gases pass through and continue to the air- removal channel and the gas-holding chamber. From there part of the clean air is vented to the atmosphere.

The venting systems functions as follows. The pressure increase in the steam-removal tunnels, is transmitted to the 100m^3 volume tank (3) (Fig. 6.5) through the impulse-transmitting pipe. The additional pressure initiates flow in the siphons (6) and water starts to flow into the air venting compartment sump. When the water level rises, the hollow metal spheres (7) float and close the inlets of the pipes (4). The pipes are closed within ~5 minutes.

After ECCS signal initiation (1-2 seconds after the break) the CTCS pumps starts. During next 5 minutes the nominal flow rate is reached. Cooled water is provided to the four condensation pools and to the sprays in the steam-reception and the gas holding chambers (Fig. 6.2). The condensed steam in the pools leads to an increase in the water level from 1.1 to 1.2 m, there it overflows weirs and spills out into the hot-condensate chamber. Since the height of the overflow barrier is 100 mm above the initial level of water in the pool, overflow from the pool begins ~20 seconds after the start of steam condensation. This prevents entrainment of water in the air current into the air removal channel. The gas velocity at this time is lower than 3 m/s, and the amount of spray carried away is small. The hot-condensate chamber can hold additional 152 t of condensate, the excess, (~360 t) flows through the overflow openings into the steam-removal corridor and reception chamber.

Due to the condensation of the steam in the pools and the reception chamber (by the spray system), the pressure at the accident location begins to decrease. When the pressure at the accident location is lower by about 5 kPa compared to the pressure in the compartment of heat exchangers at condenser tray cooling system, the over-pressure valves open and air from the compartment of heat exchangers flows to the steam-reception chamber.

6.3.6.2 Rupture of the Group Distribution Header

The Group Distribution Headers (GDH) are important components of the primary system which accept coolant water from the PH and distribute it to the core channels. A schematic of the piping between the PH and the GDH is shown in Fig. 6.9. As shown in the diagram, in order to reach the GDH (6), coolant water must pass through a flow limiter (2), a valve (3), a check valve (4) and enter the GDH through a ‘water mixer’(5) which in the event of a LOCA combines the primary system flow with ECCS water. The schematic also shows a bypass line (8) and three entry ports for ECCS water (10).

The rupture of a GDH has disrupts the coolant supply to ~40 fuel channels and generates a pressure transient in the ACS compartments. A complete rupture of the GDH leads to a reversal of the coolant flow direction in fuel channels (7) and an adequate cooling of the fuel. Of more concern is a partial rupture which could lead to flow stagnation and heat up of the fuel. This problem has been addressed in [87].

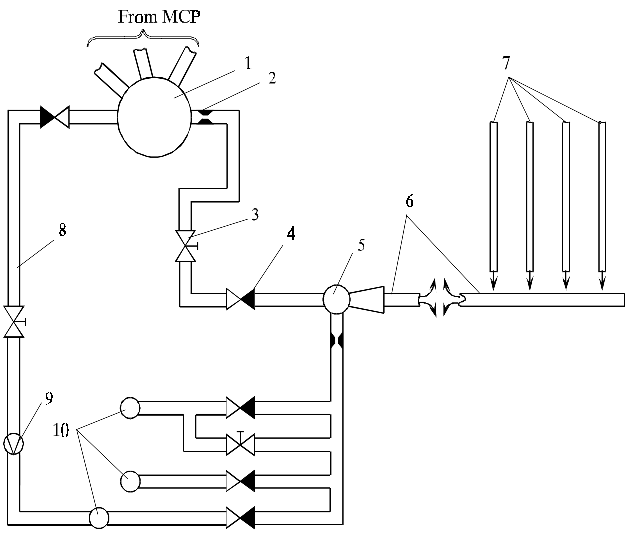


Fig. 6.9 GDH rupture after the check valve
1 - pressure header, 2 - flow limiter, 3 - valve, 4 - check valve, 5 - mixer, 6 - GDH, 7 - fuel channels, 8 - bypass line, 9 - flow meter, 10 - ECCS headers

The presence of a check valve and their influence on the direction of break and ECCS flow divides potential GDH breaks into two general types:

a) break before the check valve,
b) break after the check valve and the ECCS mixer, but before the piping to the fuel channels (Fig. 6.9).

Case b generates a more intense break flow and is of more interest as a possible challenge to the ACS. Reference [86] analyzes a broader range of scenarios, here only the case which has the maximum impact on the ACS will be considered.

Through the rupture the steam-water mixture discharges into the bottom water pipes compartments. After the pressure of steam at these compartments exceed 59 kPa, the special valves are opening and steam-water mixture flows through the steam release tunnel to both ACS towers. Further, the system works analogously to an accident in the Reinforced Leaktight Compartment. A GDH break represents a 'mid-size' LOCA. Thus the short-term loads imposed by the accident are substantially smaller then in the case for the DB-LOCA case considered in Subsection 6.3.6.1. They approach a peak value of ~130 kPa and thus do not threaten ACS integrity. The real challenge to the ACS is to ensure that the long term confinement capability is not impaired. This has a special importance for this event, because there is a small probability that a GDH break event could be accompanied by partial fuel damage and thus could have radiological consequences.

Long term analysis have been carried out which indicate that the 'long term' peak pressure (which occurs approaches, but does not exceed the peak pressures achieved short after the break [86].

6.3.6.3 Rupture of a Fuel Channel

Rupture of a fuel channel releases steam and liquid water into the inner spaces of the reactor graphite block. The graphite has a large heat capacity and elevated temperature, thus most of the liquid is vaporized. The steam-gas mixture from the reactor load goes through eight 300 mm diameter and one 600 mm diameter pipes, then through four 400 mm diameter headers to the steam distribution devices (4) and to the membrane safety devices (5) (see Subsection 6.3.5, Fig. 6.8).

After the temperature of water at the fifth pool reaches 50^{o}C, the CTCS is switched on, cooled water is provided to the fifth condensation pool and to the sprays in the gas-holding chamber. Meanwhile, the water pipes from the cooler to CTCS pools 1 - 4 are disconnected. The hot water pours through the overflow holes into the hot condensate chamber, from where it again goes to the CTCS coolers.

The volume of air released to the atmosphere will be equal to the amount of gas coming with the steam until the closing of the release pipes (maximum 1000 m^3). Water is provided to the section in order to close the release pipes (see Subsection 6.3.3, Fig. 6.6) by the MCR operator's signal, or automatically, on a signal from radioactivity meters in the outlet of the air release pipes.

6.3.6.4 Inadvertent Opening of the MSV's

The RBMK-1500 plant is provided with a total of 12 MSV's. The steam released from the MSV's during over-pressure events is directed to the top-most condenser trays of the ACS towers. When a protection valve has actuated, one or both ACS towers operate, depending on which high pressure steam loop experienced the abnormal pressure increase and the actuation of the SDV-A or MSVs. In the rest of this description, the operation of a single ACS tower is described. The operation of the other two is analogous. The steam goes through 600 mm diameter pipes to the rectangular chambers in the fifth condensation pool (see Subsection 6.3.3, Fig. 6.5). Afterwards, it goes to the steam distribution devices and then, bubbling through the water, is condensed. Air, displaced from the supply pipes and chambers by the steam, gets to the holding chamber, while an equal volume of clean air is released through the release pipes to the atmosphere. The two valves (vacuum breakers) are build into the wall of these rectangular chambers. Their purpose is to release air from the gas-holding chamber to rectangular chamber, when the pressure inside the chamber is lower by about 1.5 kPa compared to the pressure in the gas-holding chamber.

After the temperature of water at the fifth pool reaches 50^{o}C (no later that five minutes after the opening of the valve) the CTCS switches on, and cooled water is provided to the condensation pool, where it is distributed by means of the water header. Water heated to the maximum temperature, 98 ^{o}C, is poured over the barrier to the hot-condensate chamber. There, it mixes with the cold ECCS water. From the hot-condensate chamber, the water again goes to the cooler pumps.

At the same time, cold water from the CTCS coolers is provided to the sprays in the gas-holding chambers. The water pipes to the other pools are closed. The closing of the valves in the lines to pools 1 - 4 and opening to pool 5 and to the spray system occurs automatically. The provision of the water necessary for the closing of the air-release valve is switched on by the operator or by the radioactivity detectors at the exit of the gas release pipes. This reduces the probability that contaminated air would be released to the atmosphere. Due to the long gas path through the channels, the contaminated air cannot reach the release pipes by diffusion during the valve closure period (5-10 minutes). Eliminating the consequences of the opening of the protection valves, water from the hot-condensate chamber and the top condensation pool is cleaned in the power plant's water purification system.

6.3.6.5 Small Pipe Break Accident

The accident events analyzed up to this point assumed that the biggest pipe in the room is the one that breaks and that the pipe ruptures completely. This maximizes the amount of steam and hot water that is released. However, in reality, smaller breaks have a higher probability of occurring. A characteristic of accidents involving small pipe breaks in the MCP room or the bottom water pipes compartments, is that air from the pipes and the accident rooms is pushed out at a slower rate. Therefore, there it is probable that a relatively large amount of air remains in the inner ACS enclosures after the venting pipes in the gas-release room are shut by the action of the siphon. The larger air content increases the gas-holding chamber pressure. A mitigating factor is that the air enters sufficiently slowly that it can be cooled. The spray system is turned on in the gas holding chamber at the time when the gas release pipes are closed, consequently the pressure does not exceed 160 kPa, even in the case when no air is vented at all. The closure time of the gas release pipes is designed to be ~5 minutes so that they would not close before the initiation of the spray system, ensuring isothermal compression of the air. In other respects, the operation of the system does not differ from its operation in the event of a large break accident.

6.3.6.6 Hydrogen Release

During LOCA type accidents hydrogen is released together with steam-water mixture in the ACS compartments. The hydrogen arises in the reactor core from process of radiolysis of water and steam-zirconium interaction of high temperature.

In all of the accident cases considered, if pressure header of MCP breaks, approximately 85 m^3 (7.7 kg) of hydrogen is released in the ACS compartments [85]. It is difficult to calculate the hydrogen concentration in the gas- holding chamber accurately, because it is not known what proportion of it is released into the atmosphere with the air. But, assuming the worst case that all of the hydrogen is still in the gas-holding chamber, then its concentration would be 0.29 % of the entire volume. That is much less than the allowed limit, 0.4 % [50], and it is not necessary to take any special precautions to prevent formation of an explosive mixture.

In other pipes and compartments where there is a danger of hydrogen accumulation (steam reception chamber, steam distribution channels, top condensation pool isolation compartments, the top part of the gas-holding chamber), there must be hydrogen concentration meters and forced circulation. If necessary, the hydrogen concentration can be reduced to allowed limits by means of forced circulation.

There are an alarm instrumentation for monitoring hydrogen content in the ACS compartments (32 alarm sensors for each unit). This monitoring capability reduces the probability that explosive hydrogen concentrations could accumulate in the ACS compartments.

6.3.7 ACS Leakage Testing

Leak testing of the ACS is conducted at least once a year during preventive maintenance by visual examination of valves, seals of hatches, leaktight doors, tunnels of pipelines, devices and cables. Results of the testing are recorded.

Pressure testing is performed using plant compressed air or by employing a venting device of the re-circulating system. The BWPC are tested by air pressure up to about 1 kPa and the Reinforced Leaktight Compartments (RLC) and the ACS tower by an excess pressure of about 2 kPa. Test results are compared with the results measured at the start of the unit's operation. During leakage tests visual and audio inspections are conducted in the pressurized compartment. After a stable pressure is reached the pressure source is turned off and the rate of pressure decrease determined using manometers. The experimental results are employed to evaluate the leakage rate and the equivalent leakage area.

Criteria for leakage of the ACS compartment of the Ignalina NPP unit 1 state that total equivalent leakage area should not exceed 0.486 m^2 . During testing of the ACS of the Ignalina NPP unit 2 at the start of operation, the equivalent area for the reinforced leaktight compartments was found to be 0.0069 m^2. For the ACS towers these corresponding values were 0.0049 (left) + 0.0059 (right) = 0.0108 m^2. For the bottom water communication compartments it was 0.0217 m^2. The total equivalent leakage area of the ACS for unit 2 was 0.0394 m^2. Typical results for the ACS of unit 1 and unit 2 leakage test are shown in Tables 6.7 and 6.8, respectively.

Table 6.7 ACS leakage test results of the Ignalina NPP unit 1 [52]

Compartment	Excess testing pressure, kPa	Range of pressure changes, kPa	Time of pressure drop, s	Equivalent area of leakage, m^2
RLC	1.45	1.45 to 0.5	13	0.294
BWPC+RLC	1.20	1.20 to 0.5	11	0.0455
ACS towers	0.80	0.80 to 0.5	14	0.219

Table 6.8 ACS leakage test results of the Ignalina NPP unit 2 [53]

Compartment	Excess testing pressure, kPa	Range of pressure changes, kPa	Time of pressure drop, s	Equivalent area of leakage, m^2
RLC+BWPC+ACS towers*	3.96	2.0 to 1.0	130	0.0406
RLC+BWPC+ACS towers**	4.40	3.0 to 2.0	206	0.0850
RLC**	4.60	2.0 to 1.0	125	0.0308
BWPC**	0.50	0.5 to 0.2	13	0.0356
Left ACS tower**	0.80	0.80 to 0.5	117	0.0142
Right ACS tower**	0.80	0.80 to 0.5	98	0.0170

* By special preparation of ACS for testing
** Without any additions to the structure

6.4 REACTOR POWER CONTROL

Reactor neutron power control is the main key for reactor operation. Without reliable power control safe reactor operation is impossible. In this Subsection the RBMK-1500 neutron power control system, the so-called Control and Protection System (CPS) is described. The power of the nuclear reactor is regulated by inserting special rods of a neutron-absorbing material into the core. After the Chernobyl accident this system was significantly improved. For description of the effectiveness of the CPS operation, for reactor control and safety, special parameters - the reactivity coefficients are used.

6.4.1 Reactivity Coefficients

One of the fundamental parameter groups which influence the nuclear power plant's safety and controllability are the so-called "reactivity coefficients". They quantifies the effect which various other parameters (e.g. fuel of graphite temperature, amount of steam) have on the core neutron activity. For reactor stability, it is essential that the overall reactivity coefficient be negative.

In an RBMK reactor, one of the components of the overall reactivity coefficient, namely, the steam reactivity coefficient is positive. This means that, when the volume of steam in the core increases, the neutron activity also increases and, consequently, so does the amount of energy released. This is not the only type of reactor with this property - both sodium-cooled fast neutron reactors and heavy-water cooled CANDU reactors have positive coolant vaporization coefficients. In this case, it is important to ensure that the other components of the reactivity coefficient are sufficiently large so that the positive reactivity introduced into the core in any possible mode can never exceed $1\beta_{eff}$ (i.e. the "effective" fraction of delayed neutrons, obtained by adding the ^{235}U and ^{239}Pu fission). For the sake of clarity, the parameter β_{eff} is discussed first before discussing the current state of reactivity coefficients at the Ignalina NPP.

It is possible to categorize the neutrons released in the fission process into two groups:

- prompt, i.e. those which appear during fission,
- delayed, which appear from several seconds to several tens of seconds later.

The number of neutrons in the second group is much smaller, e.g. in the case of ^{235}U fission they make up only 0.65 % of the total neutrons, in the case of ^{239}Pu - only 0.21 %. With the reactor operating at a constant power, delayed neutrons in the fuel are such that β_{eff} = 0.45 % [3]. However, this small fraction of delayed neutrons is extremely important. Briefly stated, while the core reactivity is less than β_{eff}, reactor power changes occur slowly and they are easily controllable. When the reactivity exceeds this value, the situation changes and the reactor becomes more difficult to control. Therefore, all reactors used in power plants must be constructed so that their positive reactivity is compensated and never allowed to exceed $1\beta_{eff}$.

Before the Chernobyl accident, the RBMK reactors did not have this property. The destructiveness of the Chernobyl accident is due in large part to the fact that the unit 4 steam reactivity coefficient reached between 4β and 5β (where a positive void coefficient is determined as a relation of measured reactivity with β_{eff} value). Changes introduced later (see Subsection 6.3.2.4) reduced the steam reactivity coefficient in the RBMK reactors. The values of this and other reactivity-related parameters are shown in Table 6.9.

As shown in the Table, the reactor power coefficient in both units at that time reached α_N = -0.00019 β/MW. That means that if the core power increased for any reason (at a constant cooling water flow and temperature), its reactivity would decrease. Therefore, further increases in power would be prevented or at least reduced. This relation between power and reactivity is a necessary condition for stable reactor operation. Table 6.9 also shows that, according to the measurements taken at that time, the steam reactivity coefficient, i.e. the change in reactivity due to a change in the density of the reactor coolant (steam-water mixture), $\alpha_f = 0.7\beta_{eff}$. Before the above-mentioned changes to the core, this parameter was as high as 3.3 at the Ignalina NPP.

It is worth emphasizing that the steam reactivity coefficient is not proportional to the change in reactivity that would occur if all of the cooling water suddenly vaporized and only steam remained in the core. That would be a very large change in coolant density. The steam reactivity coefficient is measured by introducing a relatively small change in the amount of steam (by changing the feedwater temperature). The reactor power is held constant by the CPS, and the change in reactivity is determined by noting the new CPS rod positions. If the feed water supply were to be cut off for some reason and all of the water in the core were to vaporize, the coolant density would change from 0.6 g/cm^3 to almost zero. It is impossible to measure this change directly. According to calculations, also presented in Table 6.9, the reactivity would increase in that case by a factor α_w = 0.4β. The reliability of the calculation can be evaluated by comparing the steam reactivity coefficient calculated by the same method. The calculated value of this coefficient is approximately α_f = 1β, i.e. greater than the measured value. Therefore, it may be assumed that the actual water vaporization coefficient might be somewhat lower than the calculated value. Reactivity coefficients depend on the state of the core. They are affected by control rod positions, number of supplementary absorbers and,

Table 6.9 The state of the reactor at the Ignalina NPP [55]

Parameter	Unit 1	Unit 2
Measurement date	08.14.92	04.09.92
Reactor power, MW(th)	4050	4050
Fuel burnup, MW days/kg	7.38	7.47
Number of fuel assemblies	1606	1606
Average power of one fuel assembly, MW days	819	829
Operational reactivity reserves, manual control rods	53.9	53.9
Number of Supplementary Absorbers (SA's)	52	54
Number of group SA's included in above number	44	40
Flux tilt coefficients:		
K_Z	1.25	1.25
K_R	1.31	1.39
Reactivity coefficients:		
power α_N, (10^{-4} β/MW)	-1.9	-1.9
steam (measured) α_f, β	0.7	0.7
steam (calculated) α_f, β	1.02	1.07
core water vaporization α_w, β	0.4	0.45

especially, by the degree of fuel burnup. As the fuel burns up, its proportion of ^{235}U decreases and its proportion of plutonium isotopes, especially ^{239}Pu, increases. This has a double effect on the steam reactivity coefficient. First, since the proportion of delayed neutrons in ^{239}Pu fission is smaller, as the proportion of this isotope in the fuel increases, the value of β_{eff} decreases, and the reactivity expressed in units of β increases. Secondly, ^{239}Pu has a low energy resonance in the absorption cross-section. This cross-section is sufficiently low that, as the neutron spectrum hardens (which occurs when the amount of moderating water decreases), the average ^{239}Pu cross-section increases. This increases the absorption of neutrons by ^{239}Pu and, therefore, the core reactivity.

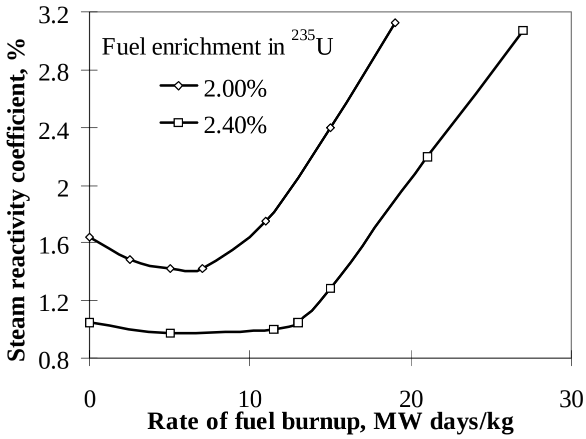


Fig. 6.10 Void reactivity coefficient versus fuel burnup

Fig. 6.10 show the calculated steam coefficient as a function of fuel burnup [56]. The conditions assumed in the calculations do not fully match the current fuel conditions at the Ignalina NPP, so the calculated value of the steam coefficient cannot be used directly.

The results of the calculations show that when the fuel burnup reaches 10 MW days/kg and more, the steam reactivity coefficient begins to increase sharply. This has economic implications. It shows that, in order to limit the size of the positive steam reactivity coefficient, it is necessary to reduce the average fuel burnup. This is clearly shown in the data presented in Tables 6.10 and 6.11.

These Tables summarize the history of steam reactivity coefficient measurements at both units of the Ignalina NPP [57]. To show the state of the reactor, the Tables show its power level (column 2) and average burnup of one fuel assembly (column 3).

As shown in Table 6.10, during the initial commissioning of the unit, when the fuel was completely fresh and there were a large number of supplementary absorber rods in the not-fully-loaded core, the steam coefficient was even negative. As the core was fully loaded with fuel and the degree of fuel burnup increased, the steam coefficient became increasingly positive. After 1986, steps were taken to reduce it. About 50 supplementary absorbers were loaded into the core (meanwhile removing fuel assemblies), and the average fuel burnup was reduced. These steps allowed the positive steam coefficient to be kept at the 1β level.

The presented data shows that the state of the reactivity coefficient (during the measurement period) meets the safety criteria. If it is assumed that the calculations and measurements correspond to reality, then it is very unlikely that a positive reactivity greater than 1β could be introduced into such a reactor. A accident similar to the one that occurred at Chernobyl unit 4 is then excluded.

Table 6.10 Measurements of steam reactivity coefficient and means of reducing it at the Ignalina NPP unit 1 [57]

Date of measurement	Reactor power, MW	Average fuel burnup, MWdays/ (fuel assembly)	Reactor loading, numbers of:			Reactivity reserves, numbers of MCR	Steam reactivity coefficient
			Fuel assemblies	Supplementary absorbers	Water columns		
84.03.20	1650	43	1436	214	11	42	-0.9
84.11.05	3420	-	1478	138	45	50	-0.26
85.03.22	4330	50	1545	101	15	46	0.4
85.04.18	2470	-	1551	80	29	75	-0.7
85.07.06	4600	720	1572	74	14	35	1.9
85.08.13	2460	-	1585	46	28	37.3	1.76
85.08.20	2450	-	1585	46	28	65.2	1.65
85.09.06	4500	-	1606	46	8	59.2	1.73
86.02.06	4500	1143	1659	1	1	37	3.3
86.08.01	4280	1162	1657	1	3	42	2.6
86.10.05	3120	1160	1658	1	2	51	2.1
87.04.09	4270	1124	1652	1	8	56	1.75
87.07.15	3350	1120	1652	1	8	53	1.7
87.12.24	3360	942	1595	53	13	47	0.5
88.02.13	3340	968	1605	53	3	45.2	0.9
88.08.09	3350	943	1606	53	2	50.6	0.8

89.01.06	3900	940	1606	53	2	54.5	1.0
90.09.05	4050	960	1605	53	3	54.2	1.0
90.11.24	4000	885	1607	52	2	56.2	0.8
91.07.19	3900	845	1604	52	5	53.7	0.7
92.08.14	4050	819	1606	52	3	53.9	0.7
96.03.20	4050	843	1607	53	1	55	0.92
97.02.07	3550	812	1471+140*	49	1	56.7	0.7

* Fuel assemblies with 2.4% enriched ^{235}U with erbium

Table 6.11 Measurements of steam reactivity coefficient and means of reducing it at the Ignalina NPP unit 2 [57]

Date of measurement	Reactor power, MW	Average fuel burnup, MWdays/ (fuel assembly)	Reactor loading, numbers of:			Reactivity reserves, numbers of MCR	Steam reactivity coefficient
			Fuel assemblies	Supplementary absorbers	Water columns		
87.10.03	2260	59,4	1434	216	11	44.6	-0.9
88.04.21	4300	522	1505	123	33	46.8	0.3
88.11.11	3950	760	1574	81	6	51.8	0.7
88.12.18	3960	828	1592	65	4	51.6	0.8
89.02.11	3900	850	1597	60	4	51.4	0.8
89.05.06	3900	875	1599	55	7	48.9	0.87
89.05.11	2300	879	1599	55	7	49.9	0.88
89.09.27	3900	877	1604	54	3	57.1	1.0
90.04.06	3900	900	1605	54	2	54.1	0.8
90.04.06	2200	900	1605	54	2	41.4	1.4*
91.02.09	3900	865	1605	54	2	54.1	1.0
91.11.29	4100	833	1605	54	2	53.7	0.8
92.04.09	4050	829	1606	54	1	53.9	0.7
96.06.25	3400	848	1411+195*	53	2	55	0.6
97.02.27	3800	966	1226+397*	33	5	53.9	0.75

* Fuel assemblies with 2.4% enriched ^{235}U with erbium

Beginning in 1995, the replacement fuel assemblies being loaded into the Ignalina NPP had fuel pallets with a different composition. The new fuel is 2.4% enriched ^{235}U (instead of 2.0% enr. fuel) and, includes 0.4% by mass of the burnable absorber Erbium (^{68}Er). The use of the new fuel makes it possible to remove supplementary absorbers out of the core. This reduces parasitic neutron loss in the core and makes it possible to increase the fuel burnup, improving the economical parameters of the reactor. Replacement of the 2.0% enriched fuel with 2.4% enriched fuel with Erbium, decreases the steam reactivity coefficient, makes the power coefficient of reactivity more negative, increases the operational reactivity margin, and reduces the flux tilt coefficient K_R. Tables 6.10 and 6.11 include information the number of fuel assemblies of the new type that have been loaded into both units up to 1997. Replacement fuel will increasingly use the new type fuel. Eventually the core loading will consist of 2.4% enriched, Er spiked assemblies.

6.4.2 Measurement and Control of Reactor Power

6.4.2.1 Reactor Neutron Flux Measurement

Reliable measurements of reactor neutron flux (power density distribution) are essential to the effective and safe control of a large nuclear reactor. If the coolant flow through the reactor channels were single-phase, power production could be determined by conventional means from the coolant flow and temperature differences. However, in a RBMK reactor, the fuel is cooled by boiling water, so another method is necessary. This method is based on neutron flux or gamma radiation intensity measurements, which are used to calculate the power released in the fuel assemblies. The neutron flux or gamma radiation are measured at discrete locations in separate, pre-chosen reactor channels, and then are interpolated and extrapolated to the other channels.

The neutron flux and at the same time the reactor axial and radial power density field in the core are measured by two methods (Fig. 6.11). In the first method, the

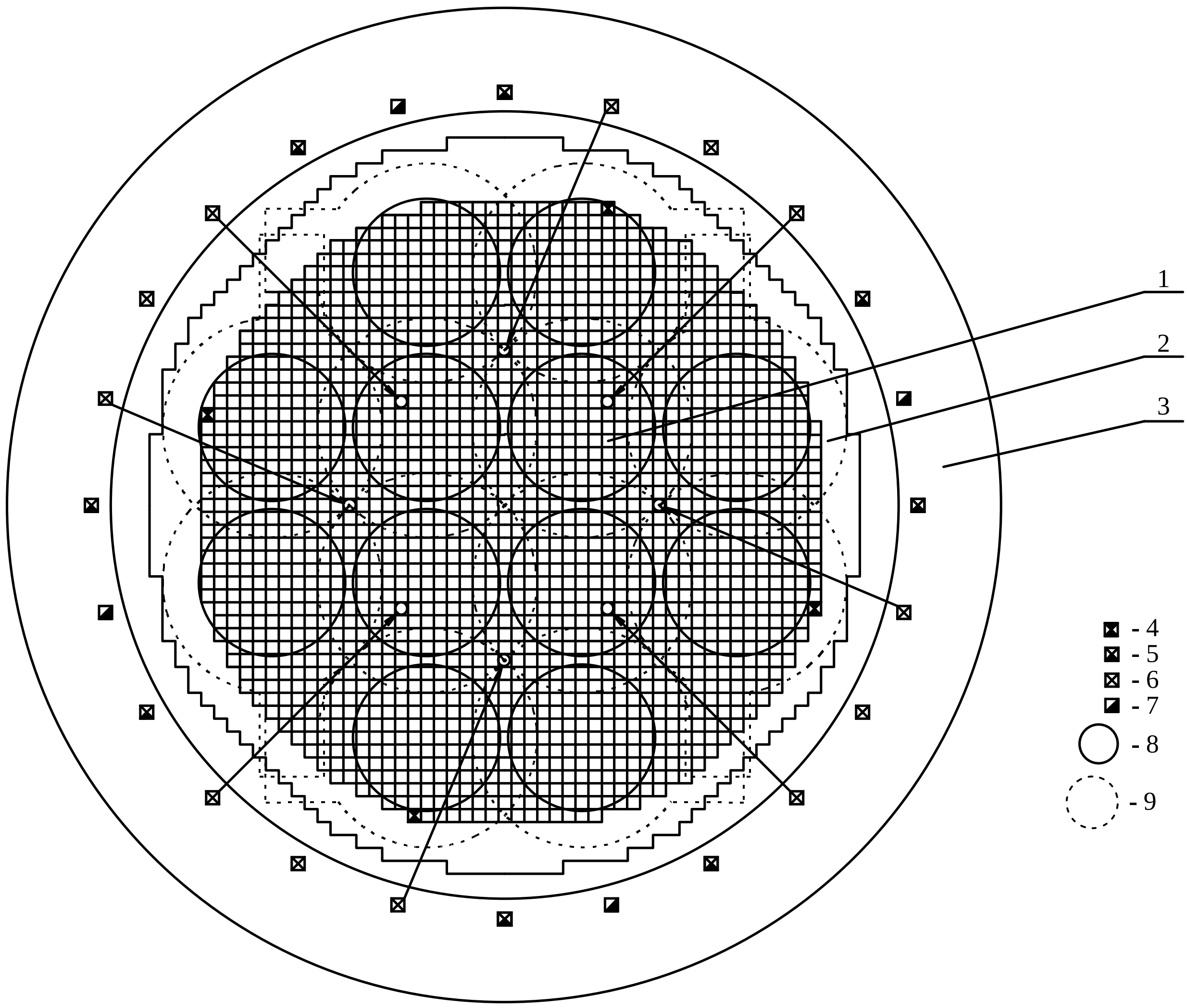


Fig. 6.11 Neutron flux measurement location
1 - reactor core region, 2 - reflector region, 3 - radial biological shield, 4 - fission chamber, 5 - ionization chamber for reactor startup, 6 - ionization chamber for normal reactor operation, 7 - chart recorder channel, 8 - LAC- zone, 9 - LEP-zone

neutron flux is measured by high precision detectors, located outside the core, where the neutron flux is lower. In the second method, the neutron flux is measured by non-inertial sensors, located right in the reactor core [35,38].

As mentioned, in the first method, the neutron flux is measured by high precision detectors, located outside the core, which uses:

- four fission chambers,
- eight ionization chambers during reactor startup,
- sixteen ionization chambers during normal reactor operation.

Ionization chambers are located in special peripheral radial ionization chamber channels, and the fission chambers are located in reactor reflector channels. The ionization chambers and the fission chambers are placed in the same channels as the control rods. These channels are described in Subsection 4.3.1. All these channels are cooled using a system separate from the fuel channel cooling system - the CRCC. This system is described in more detail in Section 5.7.

The fission chamber (type KNT-31-1) is inserted into the radial reflector channel by a special suspension bracket, shown in Fig. 6.12. The internals of the suspension bracket consists of a support plate (2), tube and coupling (3), external (5) and internal (6) bushings.

Shell (4) is fixed to the upper part of a CPS channel by two screws (not shown). Power regulator (1) is fixed to the support plate (2). To reduce the effect of vibration, a rubber seal (12) is inserted between (4) and (2). The conductor (tube and coupling) (3) is connected by couplings (7) and (13) to the fission chamber (8) and to the power regulator (1), and hermetic seals (10) and (11). External bushing (5), fastened the seal (11) make up the magnetic screen of the fission chamber, and the internal bushing (6) is the electrostatic screen for the fission chamber. The space between the fission chamber (8) and the bushing (6), as well as between the two bushings (5) and (6) is filled with insulation layers of glass fiber. The two bushings are filled with highly purified nitrogen (type VTU35ChP-662-63) to prevent corrosion and to enhance insulation. The suspension bracket is suspended in such a way that the fission chamber is properly oriented within the reactor core. Total length of the suspension bracket is 12.94 m, its mass is 20 kg. Length of the shell (4) is 15.98 m, its mass is 76.32 kg. The shell is fabricated of stainless steel type 0Ch18N10T.

The ionization chamber (type KNK-53) for reactor startup is mounted on the suspension bracket and placed in a special channel within the radial biological shield. The suspension bracket, Fig. 6.13 consists of shell (3), a conductor contact support box (1) and a thick external protective plug (11). The box of contacts (1) is connected to the instrumentation system via three couplings (12). The threads of the conductors run across insulators (2) of the contact box in longitudinal slots of the internal

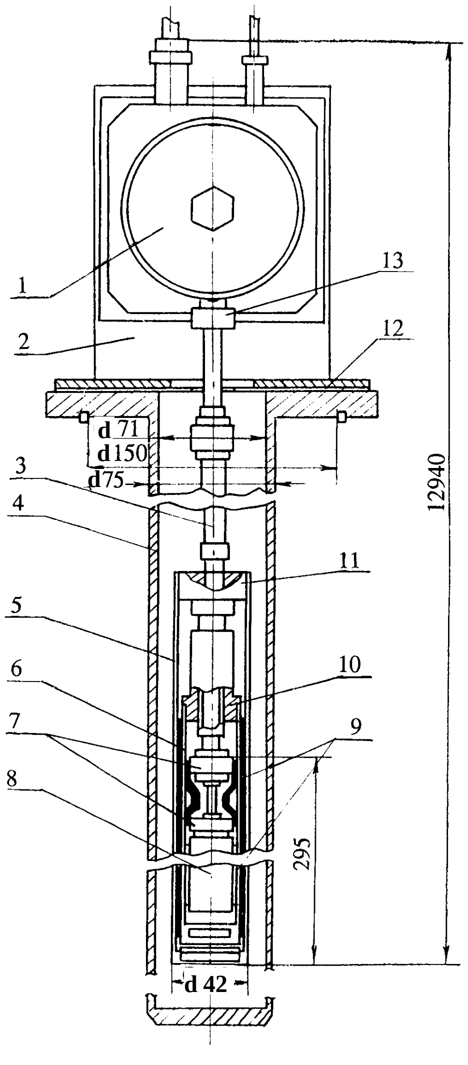


Fig. 6.12 Cross-section of a suspension bracket of the fission chamber

1 - power regulator, 2 - support plate, 3 - fission chamber conductor (tube and coupling), 4 - shell, 5 - external bushing, 6 - internal bushing, 7 - conductor connection (coupling),8 - fission chamber (type KNT-31-1), 9 - insulation, 10,11 - hermetic seals, 12 - seal, 13 - coupling

protective plug (10) to the ionization chamber (7) for reactor startup. The ionization chamber is fastened to the box of contacts (1) via a beam of variable length and three pins (8). The top (4) and the bottom (6) parts of the rod are tied by the sleeve (5) on three screws. Its length can be adapted to the central position of the ionization chamber (7) in the reactor core. Insulation layers (2) of the conductor fibers are fastened to the flanges of the beam. The shell is filled with highly purified nitrogen up to 0.02 MPa at 20 to 30 ^{o}C through a special tube (13), which is sealed after the filling operation is over. The suspension bracket casing is fastened in the CPS channel by two screws insulated by sleeves from the flange of the shell (3). The shell (3) is spaced from the internal wall of the CPS channel by seven sets of guides (9) and the insulation ring. Total length of the suspension bracket is 13.066 m, its mass 75.5 kg.

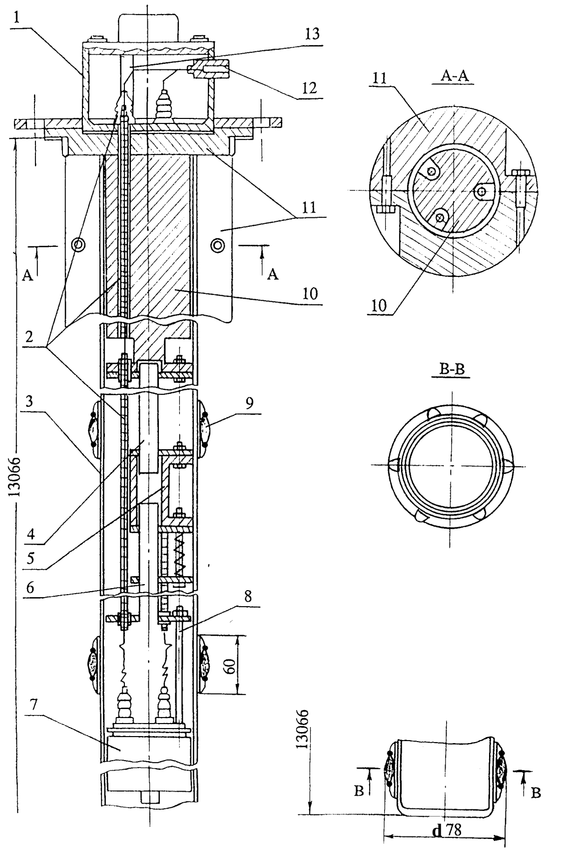


Fig. 6.13 Suspension bracket section of the ionization chamber for reactor startup
1 - contact support box, 2 - insulation layers, 3 - shell, 4 - top rod, 5 - sleeve, 6 - bottom rod, 7 - ionization chamber, 8 -pin, 9 - guide, 10 - internal protective plug, 11 - external protective plug, 12 - couplings, 13 - nitrogen tube

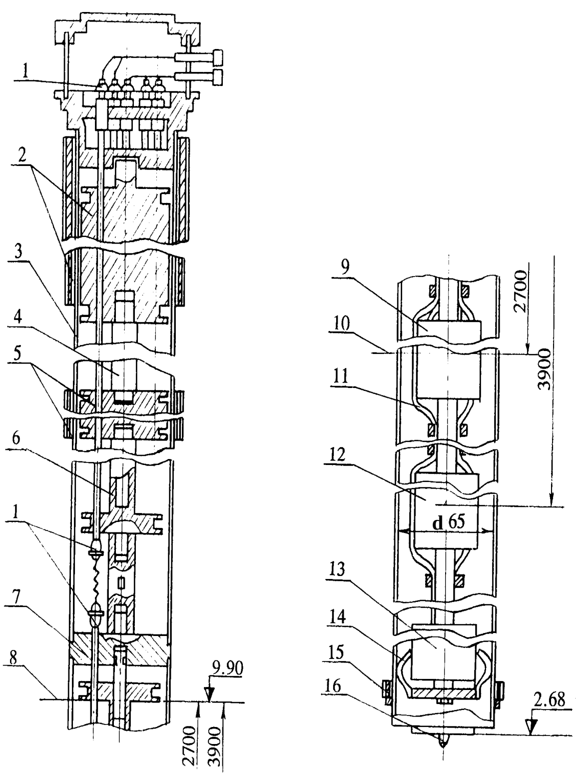


Fig. 6.14 Cross-section of the suspension bracket ionization chamber for normal reactor operation
1 - insulators, 2 - top protective plug, 3 - shell, 4 - top rod, 5 - bottom protective plug, 6 - bottom rod, 7 - flange, 8 -top boundary of the reactor core, 9 - ionization chamber No 1, 10 - inside of ionization chamber, 11 - conductor, 12 - ionization chamber No 2, 13 - ionization chamber No 3, 14 - joint, 15 - guide, 16 - valve for nitrogen filling

The ionization chamber for normal reactor operation is placed in the radial biological shield of the reactor. Three ionization chambers for normal reactor operation (type KNK-53M) are held by suspension brackets, their construction being similar to those of the ionization chamber for reactor startup, Fig. 6.14. The suspension bracket of this ionization chamber is inserted into the CPS channels.

The ionization and fission chambers are of different sensitivities and measure neutron flux on either logarithmic or linear scales, Table 6.12 [35,37]. A logarithmic scale is used during reactor startup and shutdown, while a linear scale is used during normal reactor operation. Twenty-four ionization chambers, located in eight radial ionization chamber channels, are used not only for neutron flux measurement but also for direct reactor power regulation. Each radial ionization chamber channel controls one corresponding control and protection system rod (shown in Fig. 6.11 by arrows).

As mentioned above, in the second method, the neutron flux (reactor power) is measured by non-inertial sensors, located right in the reactor core, Table 6.12. Their purpose is to provide information about local core power. For this purpose, the reactor core cross-section is divided into sectors, called Local Automatic Control (LAC) and Local Emergency Protection (LEP) zones. The twelve LEP zones overlap and, unlike the LAC zones, fully cover the entire reactor cross-section. The signals from the sensors in these zones are processed by the Power Density Distribution Monitoring System (PDDMS). This system generates the signals which are used by the CPS to control the reactor power. Monitoring of neutron flux (reactor power density distribution) is explained in Table 6.12 [37,38].

6.4.2.2 Power Density Distribution Monitoring System

Power release in the reactor core is measured and controlled by the PDDMS. This system collects data from flux sensors, discretely distributed in the core, then performs initial processing on the data and passes it to the Information Computing System (ICS). The completely processed information, showing the radial and axial power distribution and deviations from power setpoints, is presented on an informational display (tableau) in the MCR. Besides this fundamental function, the PDDMS also:

- records and displays information about flux going above allowed limits, heat exchange crisis and fuel element overload,
- generates LAC signals for the CPS for local automatic control,
- generates LEP signals when flux goes above set limits, and provides these signals to the CPS for local power level protection of the reactor,
- provides signals to the ICS for periodic computation of the reactor power, reserve coefficients (margin to allowed power limit) for each fuel channel, and other general reactor parameters,
- generates a general reactor power signal, used in various systems (chart recorders, CPS, etc.).

The PDDMS consists of:

- radial and axial power density distribution sensors,
- electronic equipment,
- calibration sensors and electronic equipment for testing the operating sensors,
- special mathematical software in the ICS, used to process PDDMS data.

Table 6.12 Neutron flux measurement [37,38]

Instrument	Purpose	Measurement range	Number of channels
Fission chamber	Provides data to CPS	Logarithmic scale (10^{-12} to 10^{-7}) N_{nom}	4
Ionization chamber for	Provides data to CPS	Logarithmic scale (10^{-10} to 10^{-4}) N_{nom}	3
reactor startup	Provides data to CPS	Logarithmic scale (10^{-8} to 10^{-1}) N_{nom}	4
	Provides data to CPS	Reactor startup period from 1 to 10 s (10^{-3} to 1.2) N_{nom}	1
		Total:	8
Ionization chamber for normal operation	Provides data to CPS	Reactor startup period from 1 to 10 s (10^{-3} to 1.2) N_{nom}	3
	Provides data to CPS	Linear scale (10^{-8} to 1.2) N_{nom}	1
	Control of corresponding CPS	Linear scale (10^{-3} to 1.0) N_{nom}	8
		Total:	12
	Control of corresponding CPS	Linear scale (10^{-3} to 1.0) N_{nom}	8
		Total:	12
Chart recorders	Provides data to MCR	Linear scale (10^{-8} to 1.2) N_{nom}	4
		Total radial ionization chamber channels:	24
Non-inertial sensors connected to the PDDMS	Provides data to PDDMS for control of corresponding LAC rods	Linear scale (10^{-3} to 1.0) N_{nom}	12 LAC zones
	Provides data to PDDMS for control of corresponding LEP rods	Linear scale (10^{-3} to 1.0) N_{nom}	12 LEP zones

Table 6.13 Reactor power density distribution monitoring system sensors

Purpose of detectors	Location	Type	Specific features	Number of channels	Sensor characteristics
Mapping of axial flux distribution (by reactor height)	CPS channels	PDMS-A	Evenly distributed in the core	12	Non-inertial
			At the periphery of the core	8	Non-inertial
			Total PDMS-A:	20	
Mapping of radial flux distribution	Fuel channels	PDMS-R1	Connected to LEP system	55	Non-inertial (hafnium oxide emitter)
		PDMS-R1	Connected to LAC and LEP systems	72	Non-inertial (hafnium oxide emitter)
		PDMS-R2	Monitoring of power	125	Inertial (silver

density distribution	emitter)
Total PDDMS: 252	

Radial power density field sensors are located right in the fuel assemblies, Table 6.13. The reactor structure allows for about 500 fuel assemblies with hollow central carrier rods. Of these, 252 are used for radial power distribution sensors, and not less than 250 are free to be used for periodic calibration of the sensor sensitivities.

The radial power density distribution instrumentation consists of 127 non-inertial sensors with hafnium oxide emitters PDMS-R1 and 125 inertial sensors with silver emitters PDMS-R2.

Radial flux distribution sensors belong either to the LAC or to the LEP systems and are divided:

- connected to the LEP system non-inertial sensors (with hafnium oxide emitter) are the in-core Power Density Monitoring Sensors Radial (PDMS-R1),
- connected to the LEP and LAC systems and directly controlling corresponding LAC zone rods, non-inertial sensors (with hafnium oxide emitter) - PDMS-R1.

The inertial Power Density Monitoring Sensors Radial with silver emitter (PDMS-R2) are uniformly distributed in the reactor core and used for monitoring of reactor power density distribution. Table 6.13 gives the characteristic information about the radial flux distribution sensors.

A radial power density monitoring sensor, Fig 6.15, consists of a sensitive element (1) in a sealed stainless steel body, sealed cable connectors (3), and biological shielding elements (2). The sensor is connected to the

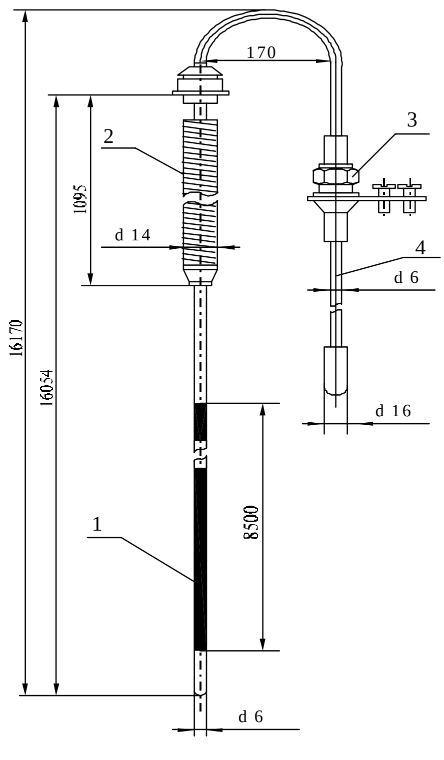


Fig. 6.15 Sensor for the radial power density monitoring (PDMS-R)
1 - sensitive element, 2 - biological shielding, 3 - sealed cable connector, 4 - cable

PDDMS equipment by a cable (4). To protect the cladding of the sensitive element from corrosion, the sensor body is filled with inert gas (argon). The sensitive element is an emission sensor manufactured from 3 mm diameter temperature resistant cable encased in a stainless steel cladding further covered with magnesium oxide. The central conductor of the cable is made either from hafnium oxide PDMS-R1 or silver PDMS-R2.

The flux distribution along the height of the core is measured by 20 in-core power density sensors of axial monitoring (PDMS-A). These are located in separate CPS channels. Twelve PDMS-A sensors are evenly distributed within the reactor core, while eight are located at the periphery of the core, in the first cells of the radial reflector.

The PDMS-A consists of a special suspension bracket, in the shell of which are inserted four two-stage chambers of type KT-21. A suspension bracket, Fig. 6.16, consists of the following parts: flange (1) sealed by screws (2) on the top of the CPS channel, cassette (4) with a protection sleeve (3), displacer (5) and guide (6). Fig. 6.16 illustrates the structure of the suspension bracket, without showing the chambers. Cover (12) is fastened to flange (1) and has four couplings (13) for the contacts of the conductors from the chambers. Cassette (4) is a tube with sleeve (3) and a flange (9) at the end. In the cassette eight tubes (10) are welded, each cassette is 11mm in diameter of 1 mm thick walls with 120 holes of 20 mm diameter at 100 mm spacings for cooling water to enter from the control rod cooling system. Six more holes of 15 mm diameters are made at the bottom near the flange (9). They also let the cooling water in and reduce side absorption of the construction materials of the cassette. Each of displacers (5) contain eight 9 mm diameter tubes (7) of 0.3 mm thick walls. Cooling water comes in through six radial 15 mm diameter holes near the flange (8) and is released through eight 6 mm diameter holes in the bottom of the displacer. Four tubes of the cassette contain type KT-21 sensors (gamma-chambers), the other four are used in experiments. A special indicator is placed in the central tube (11) during check-over operations. It indicates the level of radioactivity of the cassette walls and its readings are introduced as corrections in the error rate estimations. Total length of the suspension bracket is 18.192 m, its volume is 60 kg, it is made of stainless steel type 12Ch18N10T.

The structure of a three-axial bi-sectional gamma chamber (type KT-21) is shown in Fig. 6.17. The body of the chamber is made from 0.3 mm thick corrosion resistant steel, and filled with argon. The body (2) is welded to the cable conduits (3). Inside, there are two 100 mm long electrodes (1), 875 mm apart from each other. Each electrode is connected to one conductor of four bi-sectional, practically non-inertial, gamma chambers at separate heights, allows measurement of the tri-axial cable, and has an electrically independent cable conduit (4) to the measurement equipment. The use flux distribution at eight points along the height of the reactor.

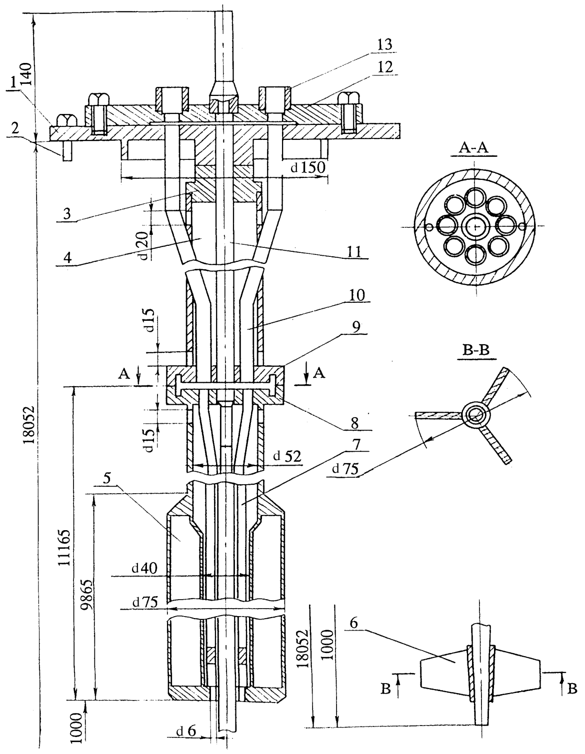


Fig. 6.16 The suspension bracket for in-core power density sensor of axial monitoring (PDMS-A)
1 - fastening flange, 2 - fastening screw, 3 - protection sleeve, 4 - cassette, 5 - displacer, 6 - guide, 7 - displacer tube, 8 - displacer flange, 9 - cassette flange ,10 - cassette tube, 11 -central tube, 12 - cover, 13 - sleeve for conductor contacts

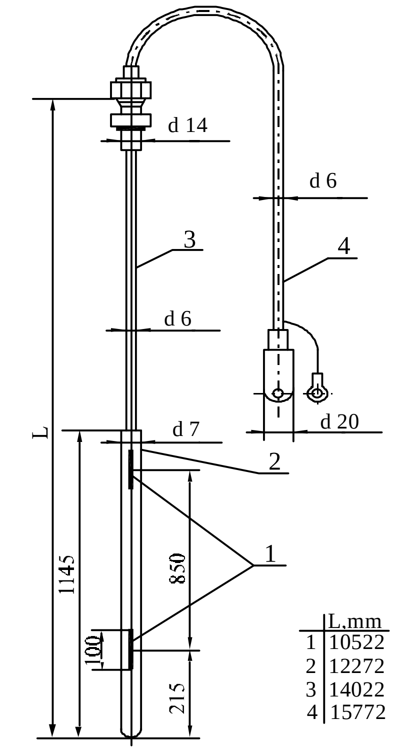


Fig. 6.17 Tri-axial bi-sectional chamber used in the PDMS-A detector
1 - electrodes, 2 - body, 3 - cable conduit, 4 - conduit

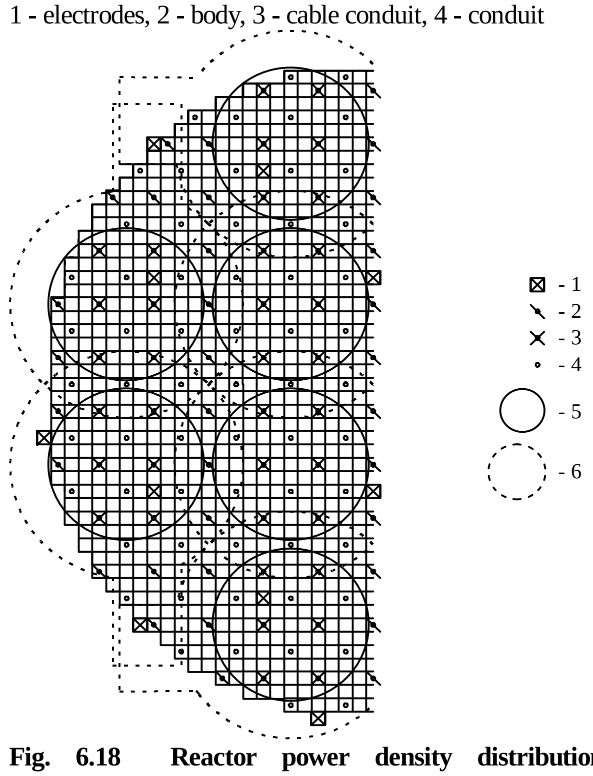


Fig. 6.18 Reactor power density distribution monitoring system sensor distribution in the reactor cross-section
1 - PDMS-A, 2 - PDMS-R1 connected to LEP system, 3 - PDMS-R1 connected to LAC and LEP system, 4 - PDMS-R2, 5 - LAC zone, 6 - LEP zone

Fig. 6.18 shows the distribution of PDMS-R1, PDMS-R2 and PDMS-A sensors and of LAC and LEP zones in a cross-section of the reactor core. The PDDMS, analyzing the sensor signals, sends information to the IC, as well as to the MCR. This system also analyzes the sensor data by LAC and LEP zones, and sends signals to the CPS. This data is used for neutron flux correction in the respective zone. If two or more PDMS-R1 sensors in a single LEP zone indicate a reading above a prescribed limit, this system then sends a signal to the CPS for shutting down the chain reaction in the given LEP zone. If such shutdown signals are present simultaneously in three or more LEP zones, then the PDDMS and CPS work together to execute an immediate reactor shutdown.

The PDDMS is very important to the control of the reactor. If both sensor systems for either of the combinations, PDMS-R1 and PDMS-R2, or PDMS-R1 and PDMS-A, fail completely, the reactor shut down manually. If any one of the PDMS-R1, PDMS-R2 or PDMS-A systems fail, and the failure cannot be corrected within two hours, the reactor must also be shut down.

6.4.3 Control and Protection System

The power of the nuclear reactor is regulated by inserting special rods of a neutron-absorbing material into the core.

The RBMK-1500 reactor power and the relative power density at the periphery of the core are controlled by the CPS. An especially important function of this system is the shutdown of the reactor in the case of an accident. This system also allows manual adjustment of the power distribution in the core, to compensate for fuel burnup, poisoning and temperature effects. The CPS automatically maintains reactor power to within 1 % of the setpoint. The CPS comprises:

- neutron flux measurement instrumentation,
- special equipment for information processing and system control,
- implementation mechanisms and absorber rods,
- control panels.

The CPS includes two subsystems:

- Local Automatic Control (LAC),
- Local Emergency Protection (LEP).

Both of these subsystems use signals from the Power Density Distribution Monitoring System (PDDMS). The LAC system automatically stabilizes the neutron flux distribution in the core. The LEP system ensures reactor emergency protection from exceeding allowed fuel assembly power in separate sectors of the core, and backs up the LAC system in case of failure.

6.4.3.1 Neutron Flux Control

The neutron flux is controlled by two separate methods (Fig 6.19). In the first method, discussed in earlier sections, the neutron flux is measured by fission chambers and ionization chamber for reactor startup and normal reactor operation. These chambers are used to register neutron flux and to shut down the reactor in an emergency. Ionization chambers in 8 of the 24 radial ionization chamber channels are connected for direct control of the neutron flux. Each radial ionization chamber channel operates one corresponding automatic control rod (shown by arrows in Fig. 6.19 [2,37]).

In the second method, the neutron flux is controlled according to the signals from the PDDMS. The reactor core is divided into twelve LAC and LEP zones. The LEP zones completely cover the entire core. The LAC and LEP absorber rods in these zones are controlled based on the corresponding PDDMS detector signals. Also, the automatic control of flux in the above-mentioned 12 zones can be supplemented by manual control. Manual control is performed based on parameters computed from the detector readings, which are shown on the MCR displays (e.g. core power distribution).

6.4.3.2 Control Rods

Control rods - neutron moderators are inserted in special channels of the CPS, which is independent of the fuel channel system. This is done to maintain the CPS in operation in case of emergency, when the fuel channel fails.

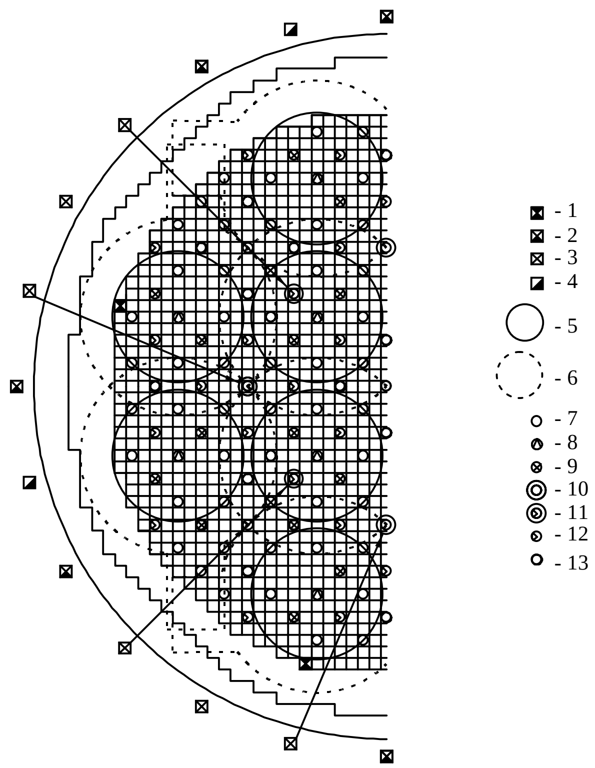


Fig. 6.19 Arrangement of control and protection system absorber rods in the core
1- fission chamber (4 per reactor), 2 - ionization chamber for reactor startup (8 per reactor), 3 - ionization chamber for normal reactor operation (12 per reactor), 4 - chart recorder (4 per reactor), 5 - LAC zone (12 per reactor), 6 - LEP zone (12 per reactor), 7, 8, 9, 10 - Manual Control Rods, 11, 12 - Shortened Absorber Rods, 13 - Fast-

Acting Scram Rods

Control rods are placed in an autonomous cooling loop with its own pumps and heat carrier (more detail about this circuit were described in Section 5). Both external and internal shell of the control rods are wetted by downward water flow, which is heated to 40 or 90 ^{o}C. As the reactor is in operation, the interior space of the channel is filled with water irrespection of the positions of the rods. An extraction of a moderator rod from the active zone would mean its replacement by water, which is an active neutron absorber. To avoid this, most of the control rods are equipped with both an absorber of boron carbide portion and graphite portion which substitutes that water. A considerable improvement of graphite balance is achieved.

The RBMK-1500 reactor contains control rods of three types. The parameters of the three types are presented in Table 6.14 for their design shown in Fig. 6.20. The constituent parts of Type 1 (MCR) shown in the inserted position one short and six normal: fastening unit (5), absorber (6), telescope joint (7) and graphite (8).

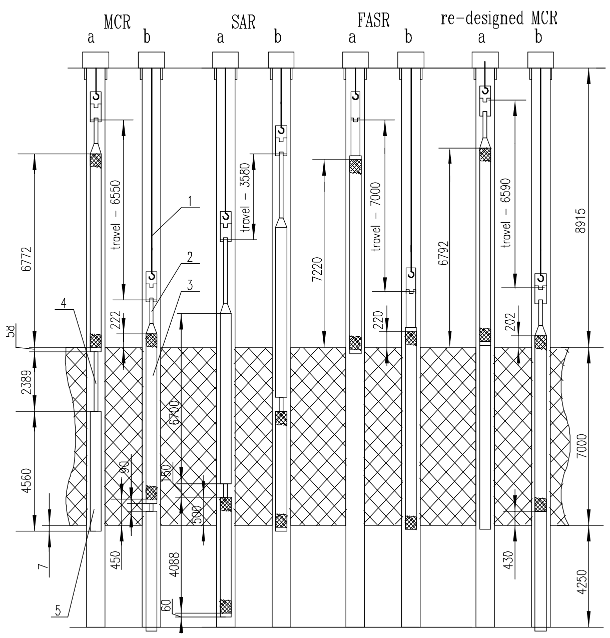


Fig. 6.20 Control rod design
a - withdrawn position, b - inserted position,
1 - suspension steel tape, 2 - fastening unit, 3 - absorber,
4 - telescopic joint (pull), 5 - graphite

Fastening unit (5) secures the rod to the damper, absorber chains (6) contain neutron absorbing boron carbide B_4C, 1650 kg/m^3 density, inserted in annular channels of a special alloy type SAVTI. At the top end the tube is sealed and its bottom end is equipped with a clamp for the subsequent chain. The bottom chain ends with a joining device where the top of the telescopic joint (7) moves freely, so that the rod-channel lengths can be adapted to the CPS channel's length. The top of the telescopic joint (7) and the bottom end of the lower most absorbing chain constitute a joint of an original construction. The bottom telescopic joint (7) ends with a clamp for the topmost water-displacer chain (8), these are graphite-loaded chains, connected by clamps, similar to the graphite-absorber chains. The bottom of the water displacer chain is equipped with an end-piece, which rests on bottom of the channel in the active zone.

In 1997 control rods of the 1st type were re-designed. The new control rod type has a special absorber "skirt" at the bottom of absorbing section of the rod, which covers the telescopic joint between absorber and graphite displacer. When such a control rod is inserted into the core, the telescopic joint slides into the inner part of the absorber, and the "skirt", moving downwards, partly covers graphite displacer. This makes it possible to eliminate the water column in the bottom part of CPS channel. The "skirt" consist of 14 segments, that are joined so that they form a cylindrical surface around the telescopic joint of the control rod. The absorber material used in the segments of the "skirt" is Dy_2TiO_5, and is located inside the stainless steel tubes. Material for the telescopic joint itself is aluminum alloy.

Table 6.14 Structural types of control rods [62]

Type of control rod	Geometrical parameters: Neutron absorber portion length, mm	Graphite portion length, mm	Distance of travel, mm	Number of rods
Type 1, Manual Control Rods (MCR)	6772	4560	6550	147
Type 2, Shortened Absorber Rods (SAR)	4088	6700	3680	40
Type 3, Fast-Acting Scram Rods (FASR)	7200	-	7000	24
Re-designed MCR	6772	7034	6590	-
			Total:	211

Rods of Type 1 (MCR) control the radial field of energy emission. Rods of Type 2 (SAR) are similar in construction to rods of Type 1, but they are inserted upwards into the active zone, and control the depth wise variations of energy emission. As observed from upside down, each rod consist of one fastening unit, six water displacer chains and four neutron absorber chains. The design of their constituent elements and joints is the same as in Type 1, but the telescopic joints are replaced by rigid joints.

Rods of Type 3 (FASR) contain no water displacer chains. They are intended to rapidly cease the nuclear reaction. Each of the rods is inserted in a channel which is cooled internally by a downward flow of nitrogen gas, and externally by a gravity film flow of water. The housing tubes of their absorber lengths are 74 mm in diameter, compared to 70 mm in rods of Type 1 and Type 2. Separate chains in Type 3 rods are fixed in an original manner, to allow internal cooling by water. Next to the absence of water-displacing chains, the most important feature of Type 3 rods is the internal tube for the gas cooler flow, which is closed by a biological cover at the top.

Each rod of type 3 consists of seven hinged chains of boron carbonate, their tops clamped to the lifting device. A sprayer at the lower ends creates a film flow of water over the CPS channel walls. The top clamp is used simultaneously with the biological cover. It is a cylinder of a specific alloy type SAV and has 55 mm diameter and is 700 mm long. The gas tube which is introduced into the sprayer at the bottom end. Since there is no hydraulic the rod is fastened to the clamp at the top end and to drag of the gas, the rod can be rapidly inserted into the active zone. The gas pipe facilitates equal pressure of the gas at the top and the bottom ends of the rod.

A radial bearing in the clamp permits the rod to rotate freely in its channel and prevents any twisting of the suspending steel tape. A damping support on the bottom of the channel saves the rod from failure in case it drops.

Rapid insertion of a FAS rod is facilitated by the absence of water in the channel. In normal reactor operation the rods are in the withdrawn top position and the channels are cooled by water-film flows from the sprayers.

Each of the absorbing and displacing chains in arbitrary rods are equipped with four protrusions to maintain central positioning of the rod in the channel. An annulus of about 2 mm is formed between the rod and the channel wall to provide free motion of the rod.

Control rods of Type 1 (MCR) and Type 2 (SAR) are suspended on steel tapes via dampers, which absorb their lifting and bottoming shocks and protect from twisting. Servo drives of the rods were discussed in Section 4.

6.5 EMERGENCY PROCESS PROTECTION SYSTEM

The process parameters are not processed by the CPS, but by the Emergency Process Protection System (EPPS). The EPPS is a system of safety functions using process parameters to trip the reactor or reduce reactor power if abnormal process conditions in the reactor core protection, the main coolant system, the separator drum, the turbine generator, reinforced leaktight compartments and electric power supply. This system generates signals that are issued to the CPS in order to actuate the reactor protection features based on process parameters. Besides generating integrated signals for CPS, EPPS generates signals for actuating safety systems (ECCS, ACS, diesel-generators, etc.) and process safety-related systems of normal operation (components of the CPS cooling circuit, gas circuit, etc.). The same system is also involved in the control of process in the nuclear power plant. The EPPS:

- monitors the process parameters and the current state of the process equipment,
- retrieves required data from the CPS and from the power density monitoring system,
- detects departures of the monitored parameters beyond their setpoint,
- receives signals from the manual control switches and buttons,
- generates signals according to the prescribed logic and issues them to the lower-level control systems,
- continuously monitors its own performance for internal failures as well as allows periodic manual tests under operating or shut-down conditions to detect latent failures,

- transfer analogue and digital data on the process parameters and its own performance to the indicating and recording devices.

The EPPS consists of the following 13 separate subsystems:

1) Emergency protection against fuel channel breaks: A FASS signal is generated in response to an increase in excess pressure in the reactor cavity to ~7.5 kPa (750 kgf/m^2).
2) Emergency protection against pressure rise in ACS compartments: A FASS signal is generated due to a excess pressure rise in the ACS compartments to ~2 kPa (200 kgf/m^2).
3) Emergency protection against out-of-range pressure in drum separators: An AZ-1 signal is generated in response to a excess pressure rise in separator drum of either loop to 7.26 MPa (74 kgf/cm^2).
4) Emergency protection against out-of range levels in separators drum: An AZ-1 signal is generated in response to level gauge readings at the scale from -1200 mm to +400 mm given below:
    - ∗ level -500 mm when reactor operates at powers below 60 percents of nominal power,
    - ∗ level -1000 mm at any reactor power,
    - ∗ level +250 mm at any reactor power.
5) Emergency protection against low flow in the MCC and MCP trip: An AZ-1 signal is generated due a multiple MCP trip in either loop, or due to a flow decrease in either loop to 1.39 m^3/s (5000 m^3/h) in two out of three operating MCPs, or one of two operating MCPs.
6) Emergency protection against low feedwater flow: An AZ-1 signal is generated if feedwater flow to either loop reduces to 50 percent of the current value, which corresponds to the current reactor power in range of 60 to 100 percent of the nominal power.
7) Emergency protection on load rejection and trip of turbine generators: An AZ-1 signal is generated in response to trip of both turbine generators, or the single operating turbine generator (i.e., closing of the stop valves or the main steam gate valve and of their bypasses) and load shedding by both turbine generators or by single operating turbine generator (i.e., drop of excess pressure downstream of the turbine stop-control valve below 1.27 MPa (13.0 kgf/cm^2)).
8) Emergency protection against the loss of AC power supply: An AZ-1 signal is generated if power supply is lost.
9) Emergency protection against voiding of CPS channels: An AZ-1 signal is generated in response to:
    - ∗ water level in the CPS circuit head tanks dropping down to 5.15 m, as indicated in the main control room, (i.e., 5.8 m above the bottom of the tanks),
    - ∗ decrease of water flow to the CPS distribution header to 0.25 m^3/s (920 m^3/h),
    - ∗ excess pressure decrease in the CPS distribution header to 0.15 MPa (1.5 kgf/cm^2).
10) Emergency protection against low levels in ECCS accumulators: An AZ-1 signal is generated in response to a level drop in the accumulators down to 4550 mm as indicated by the standard level gauges.
11) Emergency closing of throttling regulating valves of MCP: An AZ-4 signal is generated when valves close.
12) Protection on temperature rise at condensation trays of the ACS: Pumps and heat exchangers are activated if temperature in pools 1 to 4 rises to 35 ^{o}C, or if temperature of pool 5 rises to 50 ^{o}C.
13) FASS on pressure rise in the MCC compartments: A FASS signal is generated in response to pressure in the ACS compartments to ~2 kPa (200 kgf/m^2 . This latter subsystem has been added to provide a redundant scram signal for loss of coolant accidents.

The EPPS includes appropriate process parameters sensors, three sets of data processing equipment for subsystem 1 through 12, three sets of data processing equipment for subsystem 13, and the appropriate connecting cables. Initiating of reactor protection and safety related systems is carried out through voting logic upon redundant parameter measurements. Every time a protection is actuated or disabled a light-sound signal appears on a special panel in the main control room. The signals are recorded in the TITAN monitoring system. Special engineered means prevent unauthorized access to the actuation/disabling means for protection and interlocks.

6.6 EMERGENCY CORE COOLING SYSTEM

6.6.1 Purpose of the ECCS

Even after the nuclear chain reaction has been stopped by emergency protection AZ-1, heat continues to be released within the reactor. This is due to three causes:

- the nuclear fission rate is decaying exponentially, but for a brief time it is still contributing energy,
- radioactive decay of fission products,
- release of heat which has accumulated in the fuel assemblies, graphite stack and reactor metal structures.

Calculations show that the heat accumulated in the fuel assemblies is removed by the coolant in the first 100-120 seconds, but the graphite stack cools down to the coolant temperature only in 5-7 hours. In the first few hours, twice as much heat is released from the graphite stack, as would be released due to radioactive decay. Table 6.15 shows the thermal power of an RBMK-1500 reactor as a function of time during emergency protection AZ-1 (from the moment the control rods enter the core) [38].

Table 6.15 Thermal power of an RBMK-1500 reactor as a function of time, during emergency protection AZ-1

Time	N_{therm}/N_{nom}, %
0	100
5 s	98
10 s	86
20 s	55.6
30 s	35.8
1 min	15.0
2 min	8.1
3 min	7.0
5 min	6.5
10 min	5.5
30 min	4.54
1 h	3.35
5 h	0.82
10 h	0.59
24 h	0.44

In case of an emergency (e.g. break in a primary circuit pipe or a failure of some control system elements) cooling of the reactor core is maintained by the ECCS. This system is designed to respond before the fuel elements overheat as a consequence of the accident. In case of a break in a large diameter pipe, there is the possibility that the coolant water in the channels above the break reverses direction and flows from the separator drum downward through the core and out through the break. For most break location the presence of check valves and the volume of the ECCS prevent this from happening. However, for breaks in several locations, e.g. for a guillotine break of a GDH or for breaks of the pipes leading from the GDH to the fuel channels, reverse flow will persist. Analysis has shown that the reverse flow of even a high quality steam-liquid mixture can keep the channels from overheating.

The initial ECCS flow rate depends on the up-stream accumulator pressure and the pressure decrease rate within the primary system. Table 6.16 shows an estimate of the time dependent ECCS flow rate in the first hour of the maximum design basis accident (a guillotine break in a pressure header of MCP) [38].

Table 6.16 Estimate of ECCS flow rate during a DB-LOCA

Time, s	ECCS water flow, kg/s
0	1700
5	1630
10	1520
20	1100
30	710
40	500
60	370
80	280
100	230
120	220
3600	208

6.6.2 System Description

The Emergency Core Cooling System can be divided into two subsystems: a system which provides emergency coolant water immediately after a break occurs (the short-term system) which is served primarily by accumulators having a limited capacity and a long-term systems served by pumps which can draw from large water reservoirs. A schematic of the entire system is presented in Fig. 6.21.

The short term ECCS cooling system consists of 3 independent trains each of which is capable of supplying ~50% of the required cooling capacity. Two of these ECCS trains obtain cooling water from accumulators pressurized with compressed gas (22) and the third draws water from the main feed water pump (4). The Main Feed Water Pump (MFWP) is supplied with water from the deaerator tanks (2). The short- term ECCS delivers water only to the damaged coolant loop. Once the signal for activation of ECCS is

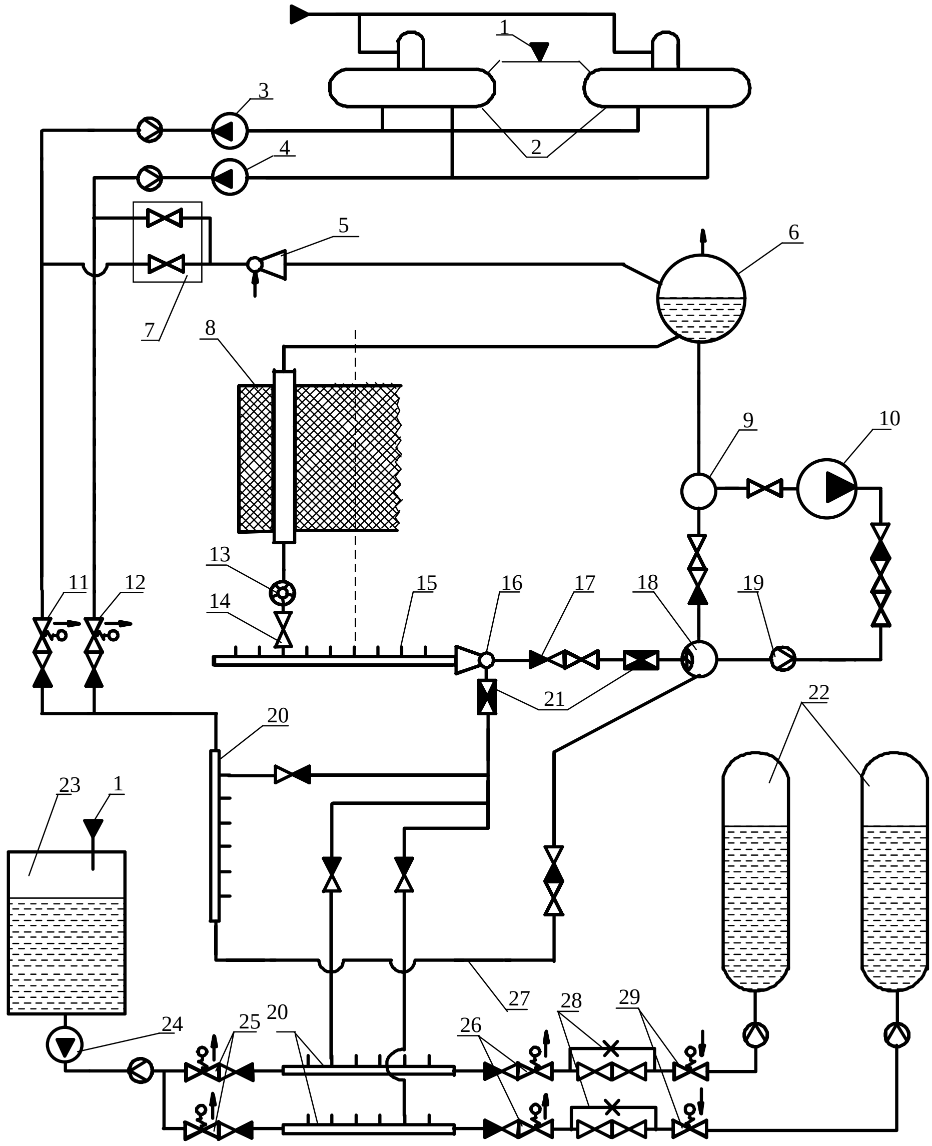


Fig. 6.21 Emergency core cooling system (only one loop of MCC is present)
1 - make- up from demineralized water reservoir, 2 - deaerators, 3 - auxiliary feed water pumps, 4 - main feed water pumps, 5 - mixer for the feed water and the return water from the PCS, 6 - separator drum, 7 - main feeder and auxiliary feeder, 8 - reactor, 9 - suction header, 10 - MCP, 11 - fast acting opening valves on pipelines from auxiliary feed water pumps to ECCS header, 12 - fast acting opening valves on pipelines from main feed water pumps to ECCS header, 13 - ball type flow-rate meter, 14 - isolation and control valve, 15 -GDH, 16 - mixer for the main coolant and the ECCS water, 17 - check valve, 18 - pressure header 19 - throttling type flow-rate meter, 20 - ECCS headers, 21 - flow limiters, 22 - ECCS accumulators, 23 - hot condensate chambers of ACS towers, 24 - ECCS pumps, 25 - fast acting opening valves on pipelines from ECCS pumps to ECCS header, 26 - fast acting opening valves in pipelines from ECCS accumulators to ECCS header, 27 - ECCS bypass line, 28 - throttle bypass, 29 - fast acting closing valves in pipelines from ECCS accumulators

received, and the loop where the break has occurred is identified, the fast acting valves (26) from accumulators and the fast acting valves (12) from main feed water pumps to the ECCS headers of this loop open. In order to maintain the required amount of the ECCS water in the feed line from the ECCS accumulators, there are the intermediate throttling stage (28). As the water level in the accumulator tanks drops below a specified set-point, fast acting valves (29) close so that the accumulator gases do not enter the primary system.

The long term ECCS also consists of three independent 50% capacity trains. Two of these trains obtain water from the hot condensate chamber (23) (they are served by six ECCS pumps (24), with a capacity of 70 kg/s each) while the third one uses the Auxiliary Feed Water Pumps (AFWPs) (3) (total of 6) to draw water from the deaerator (2). The ECCS cooling is directed into 3 independent headers (20) on each side of the reactor. Each header is connected to a pipe with a check valve. As shown in Fig. 6.21 beyond the check valves the individual ECCS pipes join together and are connected to a GDH mixer (16) which uses a flow limiter (21). This limiter reduces the amount of water which could be by-passed in the event that the header itself is ruptured. The long term ECCS supplies water to both circulation loops after the valves (11, 25) open. The water supply in the deaerators can be supplemented by the auxiliary make-up water system (1).

The main characteristics of the pumps serving the system are shown in Table 6.17. The column label 'Number per reactor' lists the pumps available for standard operation and pumps provided for back-up.

The deaerator tanks and the hot condensate chambers are supplied by the auxiliary deaerator makeup system. This system consists of a 1500 m^3 capacity demineralized water tank and the deaerator auxiliary makeup pumps which supply water to the deaerators or

Table 6.17 Characteristics of the ECCS and auxiliary deaerator makeup system pumps

Pumps	Number per reactor	Main characteristics		Remarks
Main feed water pumps (PEA1650-80)	6+1	Capacity of single pump Head	1650 m^3/h 8.9 MPa	Can not be operated by diesel generators
Auxiliary feed water pumps (PEA250-80)	5+1	Capacity of single pump Head	250 m^3/h 8.6 MPa	Can be operated by diesel gen.
ECCS pumps (PEA250-75)	5+1	Capacity of single pump Head	250 m^3/h 8.1 MPa	Can be operated by diesel gen.
Pumps of deaerator auxiliary makeup system (to supply water from demineralized water tank)	3+1	Capacity of single pump Head	500 m^3/h 2.2 MPa	Can be operated by diesel gen.
Pumps to supply water from special cleaned condensate and primary grade water tanks to demineralized water tank	2+1	Capacity of single pump	500 m^3/h	Can be operated by diesel gen.

Table 6.18 Water reservoir capacities for the ECCS

Amount	Number per reactor	Main parameters		Remarks
Water reservoir (accumulators)	16	Total volume of the water Total volume of gasses Gas pressure Water temperature	212 m^3 202 m^3 9.1 MPa 10-30 ^oC	Usable amount of water - 177 m^3
Deaerators	4	Total volume of the water Water pressure MPa Water temperature	480 m^3 0.95-1.27 10-30 ^oC	
Hot condensate chambers	2	Total volume of the water Pressure atmospheric Water temperature	1000 m^3 20-40 ^oC	
Demineralized water tank	1	Total volume of the tank	1500 m^3	Must be maintained at 1000 m^3
Treated condensate and primary grade water tanks	3*	Total volume of each tank	5000 m^3	All three tanks must hold at least 5000 m^3

* These tanks are common to the two units of the Ignalina plant

to the hot condensate chambers. The pumps are started automatically when low liquid level trips a set-point. If the water level in the demineralized water tanks decreases, according to operating procedure operators should manually start the stand-by pumps which can draw on three tanks storing cleaned condensate and primary grade water. This water supply allows the system to operate over the long term cool down period. The various reservoirs which are available for the ECCS are listed in Table 6.18.

The emergency power supply for the Ignalina NPP is provided by 6 diesel generators per unit. In the event of a loss of off-site power they are started up and can supply emergency loads in about 35 seconds. The diesel generators provide power supply for AFWPs, ECCPs, CPS pumps, but MFWPs and MCPs are tripped in the case of loss of alternating current power.

6.6.3 Operation of the ECCS

The ECCS is designed to provide cooling during the following types of accidents:

- rupture of downcomers or pipes connecting the separator drums in the liquid phase region,
- rupture of GDH, MCP piping or MCP header,
- rupture of piping in blowdown and cooling system,
- rupture of steam line upstream of main steam isolation valves,
- failure of main safety valves to re-close after opening,
- loss of feed water supply to separator drums.

The signal for the initiation of the ECCS requires the coincidence of 1) a pressure increase in compartments surrounding the main circulation piping and 2) either or both of the following:

- low level in separator drums,
- low pressure difference between MCP pressure header an separator drums.

Either of the latter two signals indicates which half of the circuit is damaged. The ECCS initiation signal triggers the opening of fast acting valves in pipelines from the accumulators and from the MFWPs (short-term system). Cooling water is thereby delivered to the damaged half of the reactor. Flow rate is controlled by valves in the throttling bypass lines.

During the first minutes, when the short-term ECCS is in operation, diesel generators are started, AFWPs and ECCS pumps are started, and valves are opened, which enables the long-term ECCS to supply water to both halves of the reactor GDHs. In case of feed water flow disturbance due to failure of MFWS pumps, indicated by low pressure in the MFWS pressure header or low feed water flow rate, the AFWPs and ECCS pumps start and valves are opened to direct water to the GDH's on both halves of MCC.

Fig. 6.22 presents a block diagram summarizing the conditions which lead to ECCS activation.

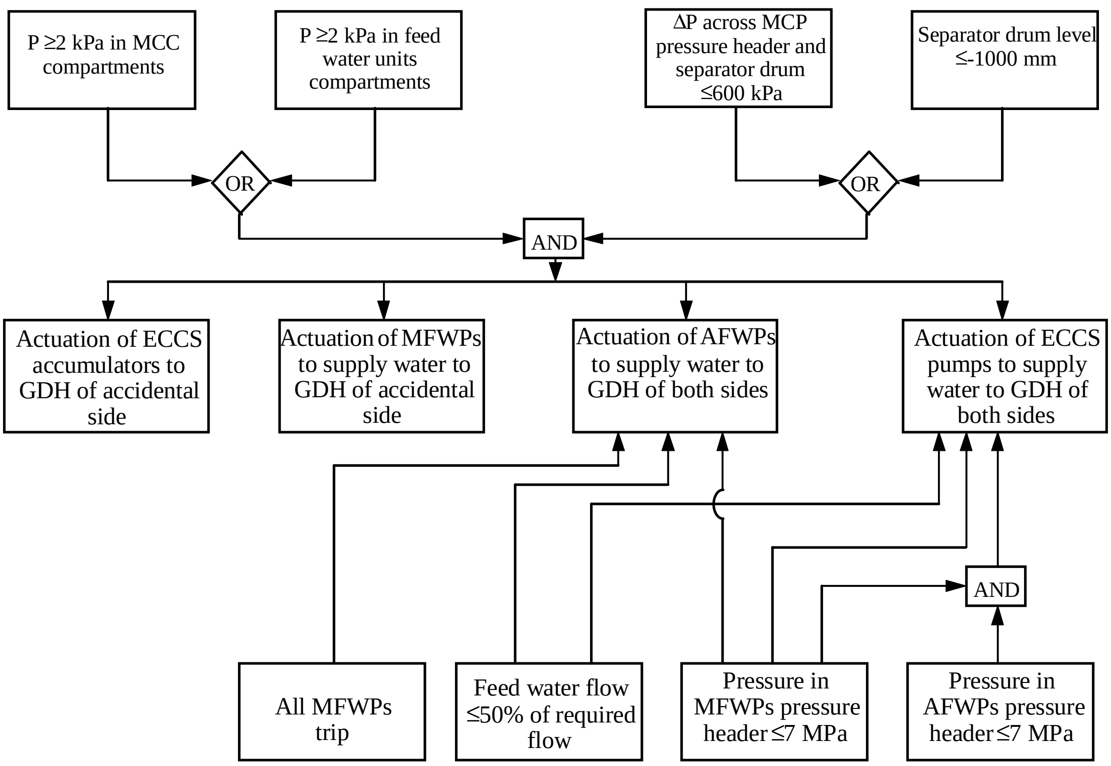


Fig. 6.22 ECCS actuation logic