



INTERNATIONAL ATOMIC ENERGY AGENCY

WWER-1000 Reactor Simulator

*Material for Training Courses
and Workshops*

Second Edition

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WWER-1000 Reactor Simulator

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Second Edition

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FOREWORD

The International Atomic Energy Agency (IAEA) has established an activity in nuclear reactor simulation computer programs to assist its Member States in education. The objective is to provide, for a variety of advanced reactor types, insight and practice in their operational characteristics and their response to perturbations and accident situations. To achieve this, the IAEA arranges for the development and distribution of simulation programs and educational material and sponsors courses and workshops. The workshops are in two parts: techniques and tools for reactor simulator development; and the use of reactor simulators in education. Workshop material for the first part is covered in the IAEA publication: Training Course Series No.12, Reactor Simulator Development (2001). Course material for workshops using a pressurized water reactor (PWR) simulator developed for the IAEA by Cassiopeia Technologies Inc. of Canada is presented in the IAEA publication, Training Course Series No. 22, 2nd edition, Pressurized Water Reactor Simulator (2005) and Training Course Series No.23, 2nd edition, Boiling Water Reactor Simulator (2005).

This report consists of course material for workshops using the WWER-1000 Reactor Department Simulator from the Moscow Engineering and Physics Institute, Russian Federation. N. V. Tikhonov and S. B. Vygovsky of the Moscow Engineering and Physics Institute prepared this report for the IAEA. The IAEA officer responsible for this publication was R.B. Lyon from the Division of Nuclear Power.

EDITORIAL NOTE

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CONTENTS

1.	OVERVIEW OF WWER-1000 SYSTEMS AND OPERATION	1
1.1.	Plant overview	1
1.2.	Nuclear steam supply system	2
1.2.1.	Reactor	3
1.2.2.	Reactor core.....	4
1.2.3.	Main circulation pumps.....	6
1.2.4.	Steam generators	6
1.2.5.	Pressure compensation system	8
1.2.6.	Primary circuit feed and bleed system	10
1.2.7.	Secondary circuit.....	12
1.3.	Reactor control and protection system	13
1.3.1.	Control rod system	14
1.3.2.	Boron regulation system.....	16
1.3.3.	Neutron flux monitoring system	17
1.3.4.	In-core instrumentation	17
1.3.5.	Control and protection system operation	19
1.4.	Safety systems	24
2.	SIMULATOR MODEL FUNDAMENTALS.....	26
2.1.	Mathematical model of core neutron kinetics	26
2.1.1.	List of variables	26
2.1.2.	Theoretical approach	26
2.1.3.	Neutron flux density calculation	28
2.1.4.	Power release calculation	28
2.1.5.	Main equations and methods of solving.....	30
2.1.6.	Neutron flux instrumentation model	33
2.2.	Mathematical model of thermal hydraulics.....	34
2.2.1.	List of variables	34
2.2.2.	Basic equations for coolant	35
2.2.3.	Spatial discretization	36
2.2.4.	Time discretization.....	36
2.2.5.	Closure correlation equations.....	37
2.2.6.	Fuel assembly model.....	42
2.2.7.	Final form of equations	43
3.	SIMULATOR USER GUIDE	45
3.1.	Introduction	45
3.2.	Installing, uninstalling and simulator startup	45
3.3.	Training tasks selection	47
3.4.	Simulator start, stop and exit.....	49
3.5.	Main simulator controls	49
3.5.1.	Control rod operation	49
3.5.2.	Pump operation	50
3.5.3.	Valve operation	50
3.5.4.	Automatic controller operation	50
3.6.	Protocol analysis	51
3.6.1.	Protocol selection	51

3.6.2. Protocol viewer window functions.....	52
3.7. Simulator display pages	54
3.7.1. Reactivity control page (CPS).....	54
3.7.2. Enunciators page (TAB).....	61
3.7.3. Primary circulation circuit page (1C).....	63
3.7.4. Feed and bleed system page (TK).....	65
3.7.5. Process support systems page (TQ)	66
3.7.6. Extraction water cooling system page (TF)	67
3.7.7. Secondary circuit page (2C).....	68
3.7.8. Trends page (GRP).....	69
3.7.9. Reactor core parameters page (PAR).....	71
3.7.10. Three dimensional diagram page (3D).....	72
4. SIMULATOR EXERCISES	74
4.1. Overview	74
4.2. Training exercises	74
4.2.1. Exercise 01	74
4.2.2. Exercise 02	75
4.2.3. Exercise 03	75
4.2.4. Exercise 04	76
4.2.5. Exercise 05	76
4.2.6. Exercise 06	77
4.2.7. Exercise 07	77
APPENDIX A: LIST OF MAIN VARIABLES FOR WWER-1000 REACTOR DEPARTMENT SIMULATOR.....	79
APPENDIX B. LIST OF ITEMS AT THE ALARM PANEL FOR WWER-1000 REACTOR DEPARTMENT SIMULATOR.....	81

1. OVERVIEW OF WWER-1000 SYSTEMS AND OPERATION

1.1. Plant overview

WWER is an abbreviation for ‘water water energy reactor. It is a pressure vessel type nuclear reactor with water used both as moderator and coolant, resulting in a thermal neutron spectrum. The number following the reactor type usually indicates the rated power of the unit. Thus, WWER-1000 designates a unit with 1000 MW electrical power.

Figure 1 presents the major components of the NPP with a WWER reactor.

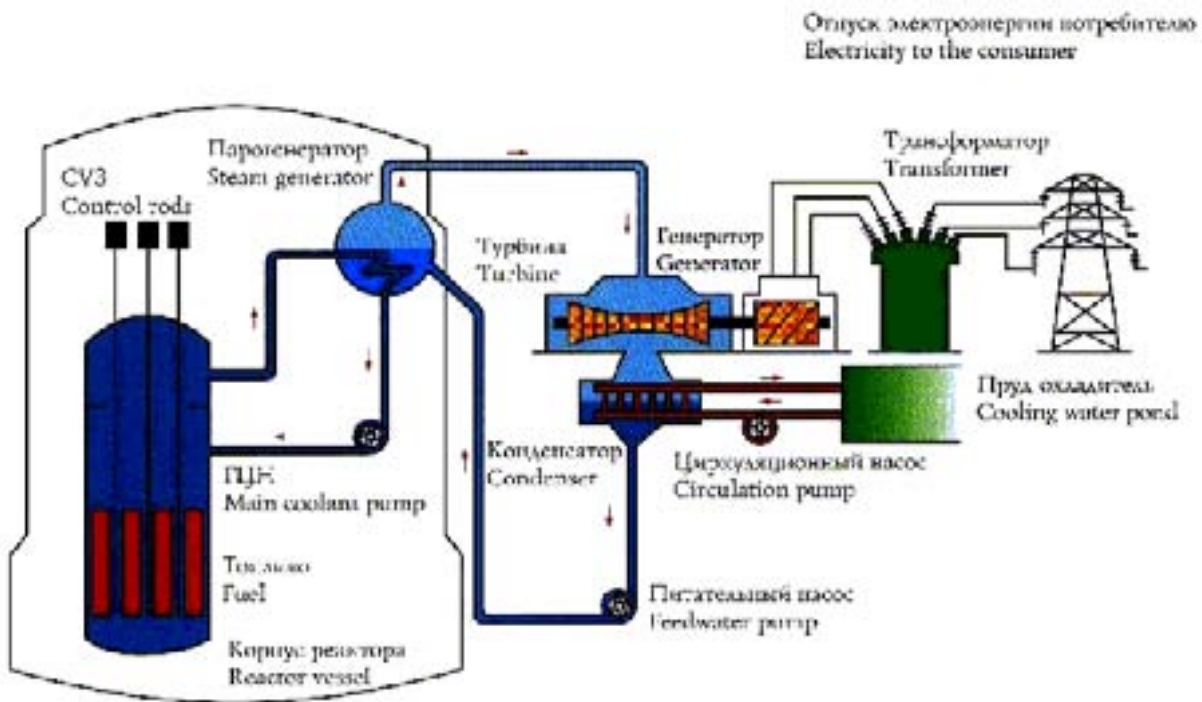


FIG. 1. Schematic diagram of NPP with WWER reactor.

Heat that is generated in the reactor core from the fission of nuclei in the fuel is removed by the coolant (for NPP with WWER, the coolant is water or water-steam mixture). After leaving the reactor core, the coolant is transported along the part of the primary circulation circuit called “the hot leg” to the steam generator.

The steam generator is a heat exchanger in which the heat from the primary circuit coolant transfers to the feed water of the secondary circuit to form steam.

After the steam generator, the coolant is transported along the part of primary circulation circuit called the “cold leg” back to the reactor vessel.

There are four circulation loops in the primary circuit of the NPP with WWER-1000 reactor. The coolant is pumped by four main circulation pumps, installed one in each loop.

In the secondary circuit, steam formed in the steam generators is transported to the “balance of plant systems”. Most of the steam formed in the steam generators is sent to the turbine, with a much smaller part to feed water heating.

After the turbine, steam is dumped to the condenser and condensed. From the condenser the water is transported through the low-pressure heaters to the deaerator for removal of non-condensable gases. From the deaerator, feed water is transported through high-pressure heaters to the steam generator.

The turbine systems are presented in the Simulator in a simplified way.

1.2. Nuclear steam supply system

The main systems of the NPP with WWER-1000 reactor are the following:

- Reactor;
- Primary circuit: main circulation pipelines, main circulation pumps (MCP), steam generators (SG);
- Pressurizer and primary circuit pressure compensating system;
- Primary circuit feed and bleed system, including boron regulation;
- Secondary circuit steam lines and feed water pipelines;
- Control and protection system;
- Safety systems.

These systems are considered in this section of the manual.

There are many auxiliary NPP systems, which are not presented in the simulator and are not considered in this manual.

Primary circuit equipment layout is presented in Figure 2.

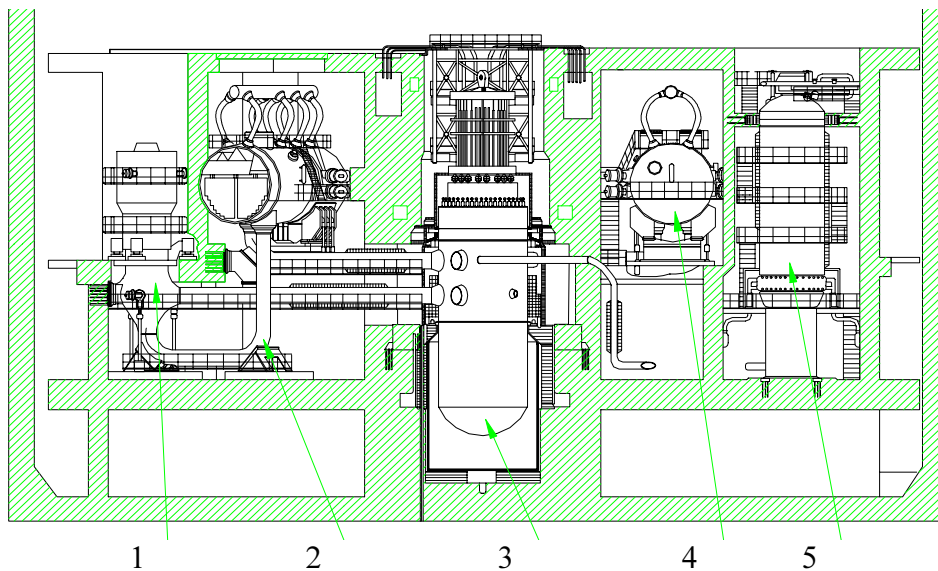


FIG. 2. Crosscut view of WWER-1000 containment.

1. Main circulation pump.
2. Primary circuit pipelines.
3. Reactor vessel.
4. Steam generator.
5. Pressurizer.

Figure 3 presents the major components of the WWER primary circuit included in the scope of modelling in the Simulator, as well as a part of the secondary circuit equipment, namely the steam generator pressure control valves and the steam generator inventory level control valves.

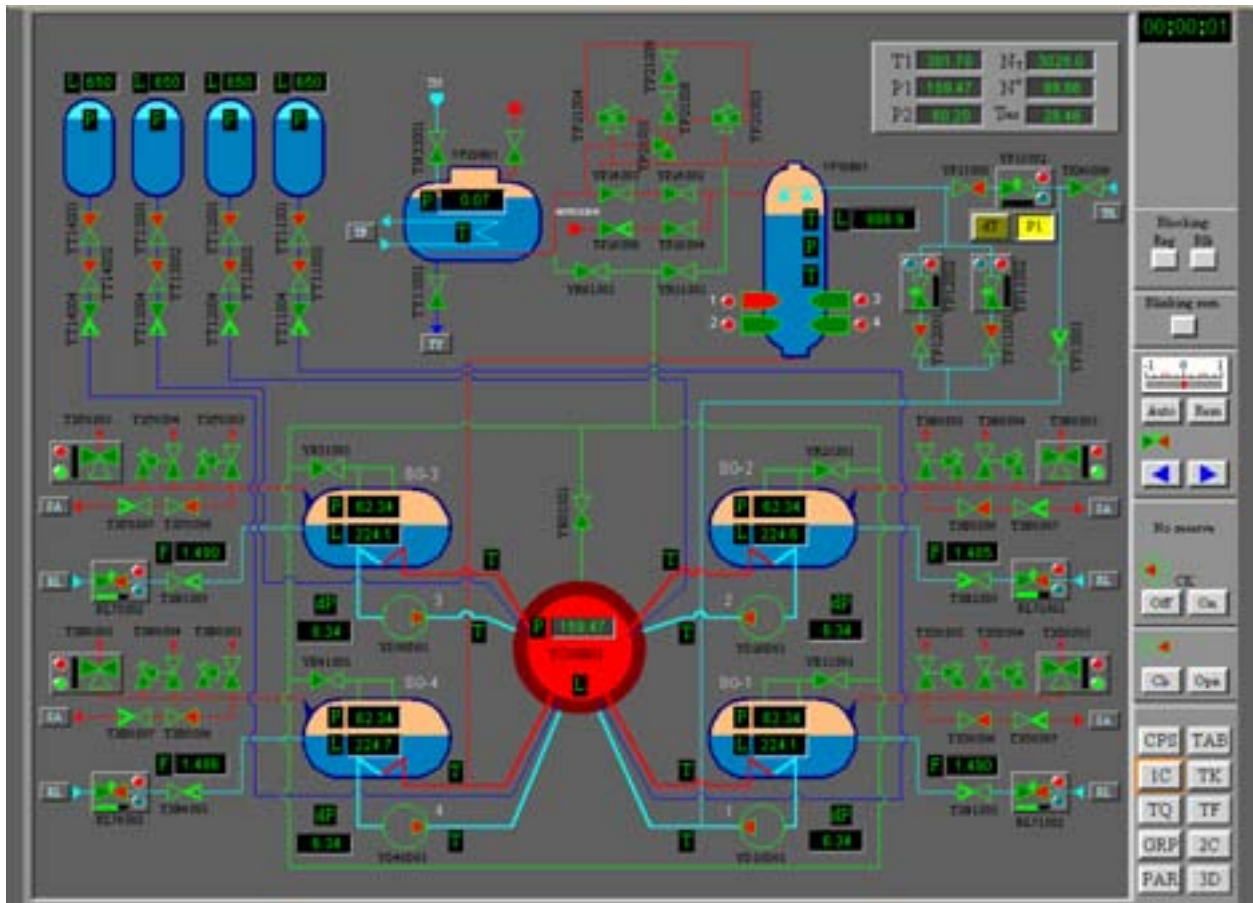


FIG. 3. WWER-1000 primary circuit in the simulator.

1.2.1. Reactor

The WWER-1000 reactor is a vessel-type Light Water Reactor where chemically purified water with boric acid serves as coolant and moderator. The reactor is intended for generation of heat within the NPP nuclear steam supply system. Regulation of reactor power and suppression of the fission chain reaction is carried out by two systems adjusting reactivity, that are based on different principles:

- Introducing solid absorbers - control rods system (control & protection system – CPS); and
- Injection of liquid absorber - boron regulating system.

Control rods are used for changing reactivity in maneuvering regimes and for reactor shutdown in normal and emergency operation conditions. Boron regulation is used for slow changes in reactivity. The boron concentration is changed during the life cycle.

The coolant is heated while it flows in fuel assemblies due to the energy of nuclear fission in the fuel. The coolant enters the reactor through input nozzles, passes a ring gap between the

reactor vessel and the core-well, and, through a perforated bottom plate, enters fuel assemblies installed in the reactor core. The coolant then passes through the perforated plate, enters the inter-tube space of the protection tubes block, then goes to the ring gap between the core well and the vessel and through outlet nozzles exits the reactor vessel to the “hot leg”.

Figure 4 presents the WWER-1000 reactor layout.

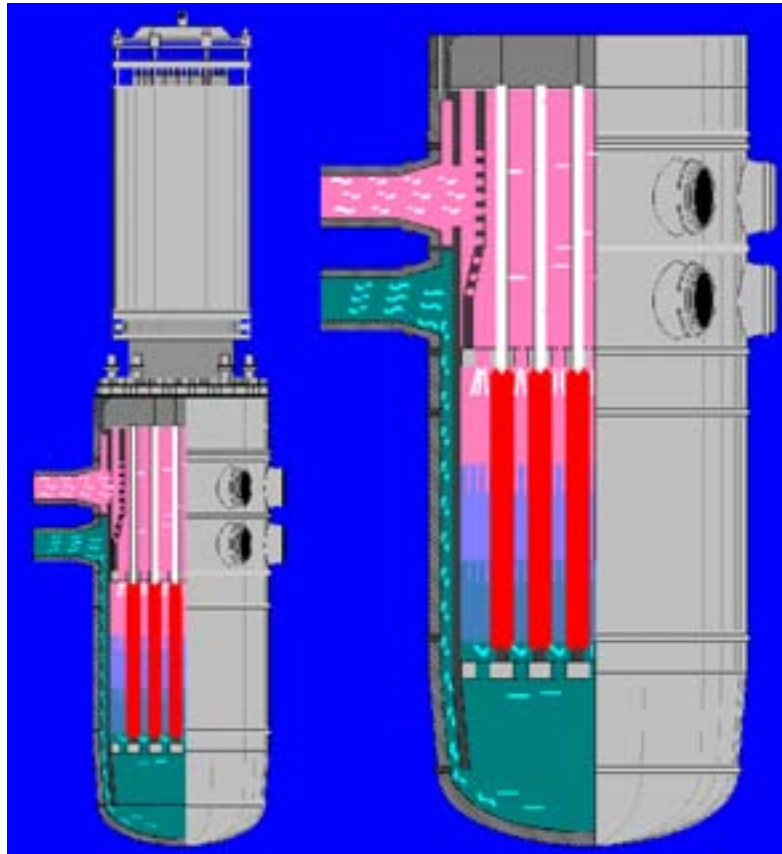


FIG. 4. WWER-1000 reactor general view.

The reactor includes the following components:

- Core;
- Reactor vessel;
- In-vessel installations and upper block;
- Step-type electro-magnetic gears of the control rods;
- Neutron flux measuring instrumentation.

1.2.2. Reactor core

The WWER-1000 core is composed of fuel assemblies (FA) having hexagonal form and located on a hexagonal grid with a constant step of about 236 mm. The number of fuel assemblies in the core depends on their size and the reactor rated power. The maximum size of a fuel assembly is limited by nuclear safety requirements in order to eliminate the

possibility of critical mass occurrence, and the minimum size of the fuel assembly is limited by the reasons of cost efficiency.

The WWER-1000 reactor core major characteristics are as follows:

- Total number of fuel assemblies in the core 163,
- Number of fuel assemblies with control rods 61,
- Height of heating part (in cold state) 3.53 m,
- Step between fuel assemblies 0.236 m,
- Pass section of the core in the heating part 4.17 m²,
- Coolant's flow rate through the core 17650 kg/s,
- Reactor thermal power 3000 mW.

Figure 5 presents the WWER-1000 core plan view and fuel assembly configuration.

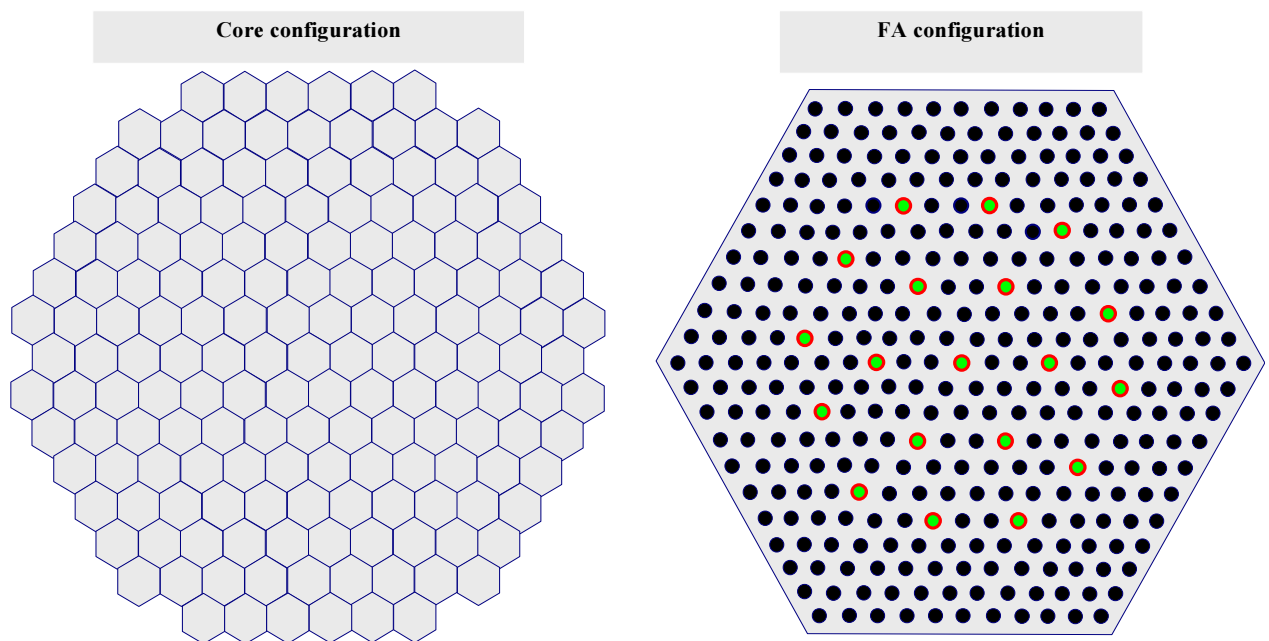


FIG. 5. WWER-1000 core plan view and fuel assembly configuration.

The fuel assembly for the WWER-1000 consists of a regular grid of fuel rods. In certain positions fuel rods are replaced with non-fuel elements, for example absorbing elements of control rods (total of 18) or rods with burnable absorbers.

The WWER-1000 fuel assembly major characteristics are as follows:

- Number of fuel rods 312,
- Step between fuel rods 12.75 mm,
- Number of tubes for absorber elements 18,
- Length of FA active part 3530 mm,
- Number of distant grids (support plates) 14.

Figure 6 presents the main WWER-1000 FA elements schematically.

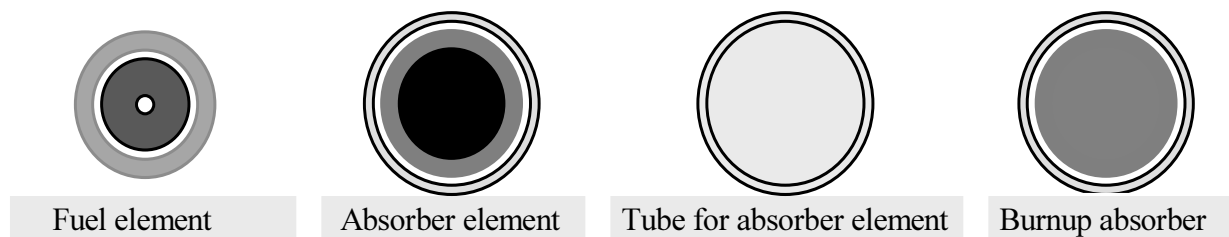


FIG. 6. WWER-1000 FA elements.

WWER-1000 fuel pin major characteristics are as follows:

- Fuel pin diameter 9.1 mm,
- Cladding thickness 0.69 mm,
- Cladding material alloy – Zr110,
- Fuel part diameter 7.53 mm,
- Fuel material UO₂,
- Diameter of central aperture in fuel pellet 2.3 mm,
- Fuel density 10.4 g/cm³,
- Enrichment of feeding fuel 3.3, 4.4, 3.0 and 4.0 %.

1.2.3. Main circulation pumps

The four main circulation pumps are vertical centrifugal pumps with mechanical shaft seals. Each pump is driven by a vertical air-water cooled electric motor.

The MCP rotating part has significant rotation inertia and in case of loss of motor power, the rate of coolant flow decrease matches the reactor power rundown caused by a reactor trip. After the MCP stops completely, natural circulation maintains core cooling.

1.2.4. Steam generators

The steam generator (SG) is intended for heat removal from the primary circuit coolant and forming saturated steam in the secondary circuit. Steam generators at NPP with WWER-1000 reactors are of the horizontal type.

Figure 7 presents a crosscut view of WWER-1000 steam generator, and Figure 8 presents the general view.

The steam generator vessel is marked in red. Primary side coolant goes via horizontal tube bundles, marked in black. Above the steam generator there is a steam collector, marked in green. Through this collector, steam formed in the steam generator goes to the main steam header and then to the turbine. The mechanical separator is marked in yellow. The perforated plate is marked in dark blue. Feed water pipes are marked in blue.

While operated under normal conditions at rated power the following parameters are maintained in the steam generators:

- Pressure in SG (6,27+0,19) MPa;
- Temperature of feed water (220+5) °C;
- Water level (320+50) mm;
- Steam humidity at SG outlet not more than 0.2%.

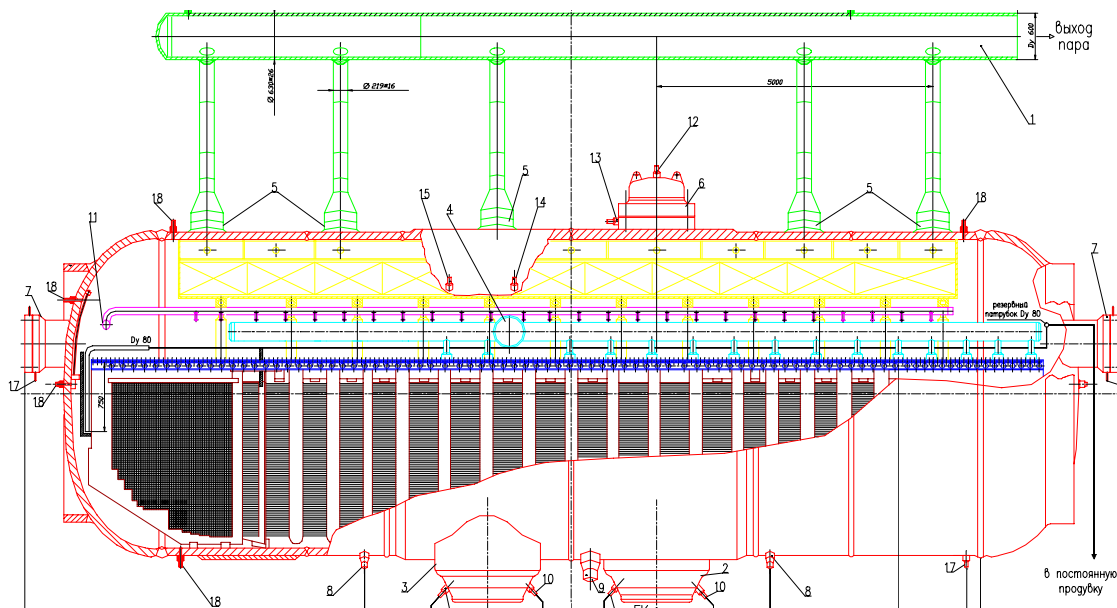


FIG. 7. Crosscut view of WWER-1000 steam generator.

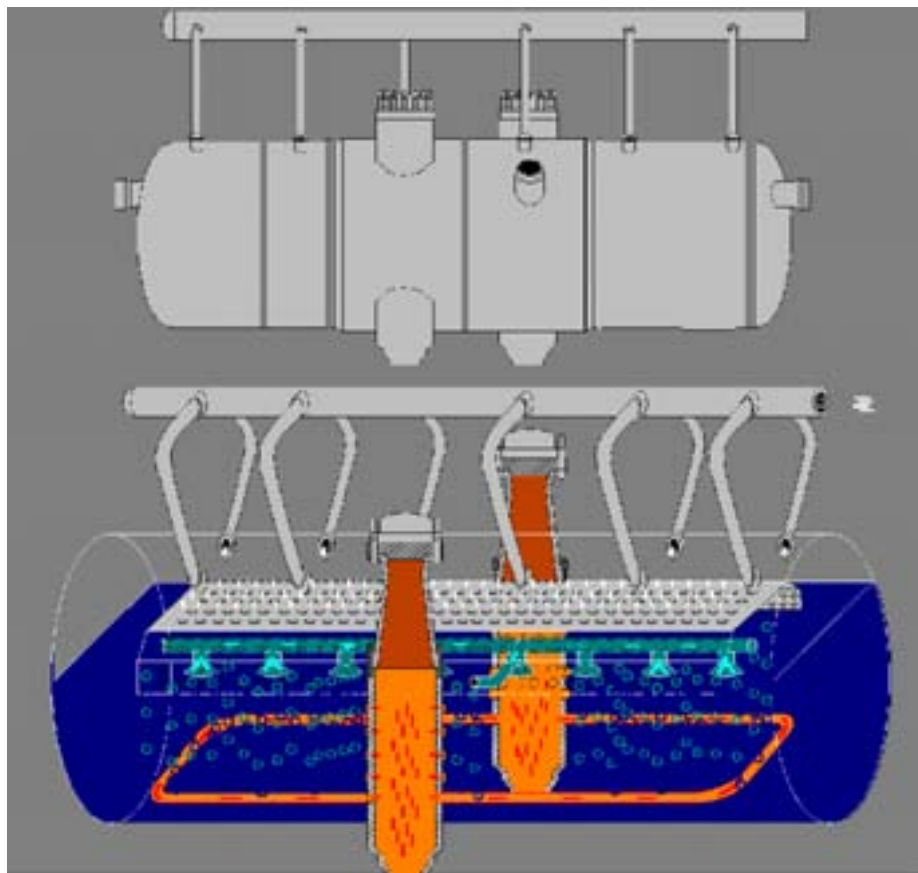


FIG. 8. WWER-1000 steam generator general view.

1.2.5. Pressure compensation system

The principal objective of pressure compensation in WWER reactors is to keep the primary circuit coolant in the liquid state. The WWER-1000 primary circuit pressure control equipment is shown in Figure 9.

The pressure compensation system is intended for:

- Primary circuit pressure control during steady state and transients;
- Protection of primary circuit equipment from exceeding the design base pressure;
- Increasing primary circuit pressure during start-up of the nuclear steam supply system;
- Decreasing primary circuit pressure during cool down of the nuclear steam supply system.

The following equipment is included in the pressure compensation system:

- Pressurizer;
- Bubbler-condenser;
- Steam dumping pipeline with impulse safety valves and relief valves;
- Pipeline for dumping steam-gas mixture from pressurizer with valves and constrictor;
- Injection pipeline;
- Pipeline connecting pressurizer with “hot leg” of main circulation pipeline (so-called connecting pipeline).

The pressurizer is a vertical tank connected to the “hot leg” of the primary circuit loop. To maintain a constant pressure in the primary circuit, the pressurizer is equipped with a spray system and electrical heaters. The spray system is located at the top of the pressurizer and is intended for injecting water into the steam volume to provide steam condensation and consequent pressure drop. Electric heaters, located in the lower part of the pressurizer, are intended for water heating and consequent pressure increase in the primary circuit. They are also used for coolant heating during reactor start-up.

The pressurizer size and volume of water are chosen to eliminate the regimes where steam from the pressurizer could appear in the main circulation pipelines.

The main parameters of the pressurizer are as follows:

Pressure	15,7 (160) MPa (kgs/cm ²);
Temperature	346 °C;
Volume (total)	79 m ³ ;
Water volume in rated state	55 m ³ ;
Steam volume in rated state	24 m ³ ;
Electric heaters power	2520 ±190 kW.

The bubbler-condenser is designed for accepting steam from the pressurizer. It is a horizontal tank filled with water for 2/3 of its total volume. Inside the bubbler-condenser there are three distributing steam collectors located near the bottom, and heat exchanger tubes for cooling the water in the bubbler-condenser. There is a safety membrane that breaks when the pressure exceeds the emergency setpoint, thus dumping steam to the reactor containment.

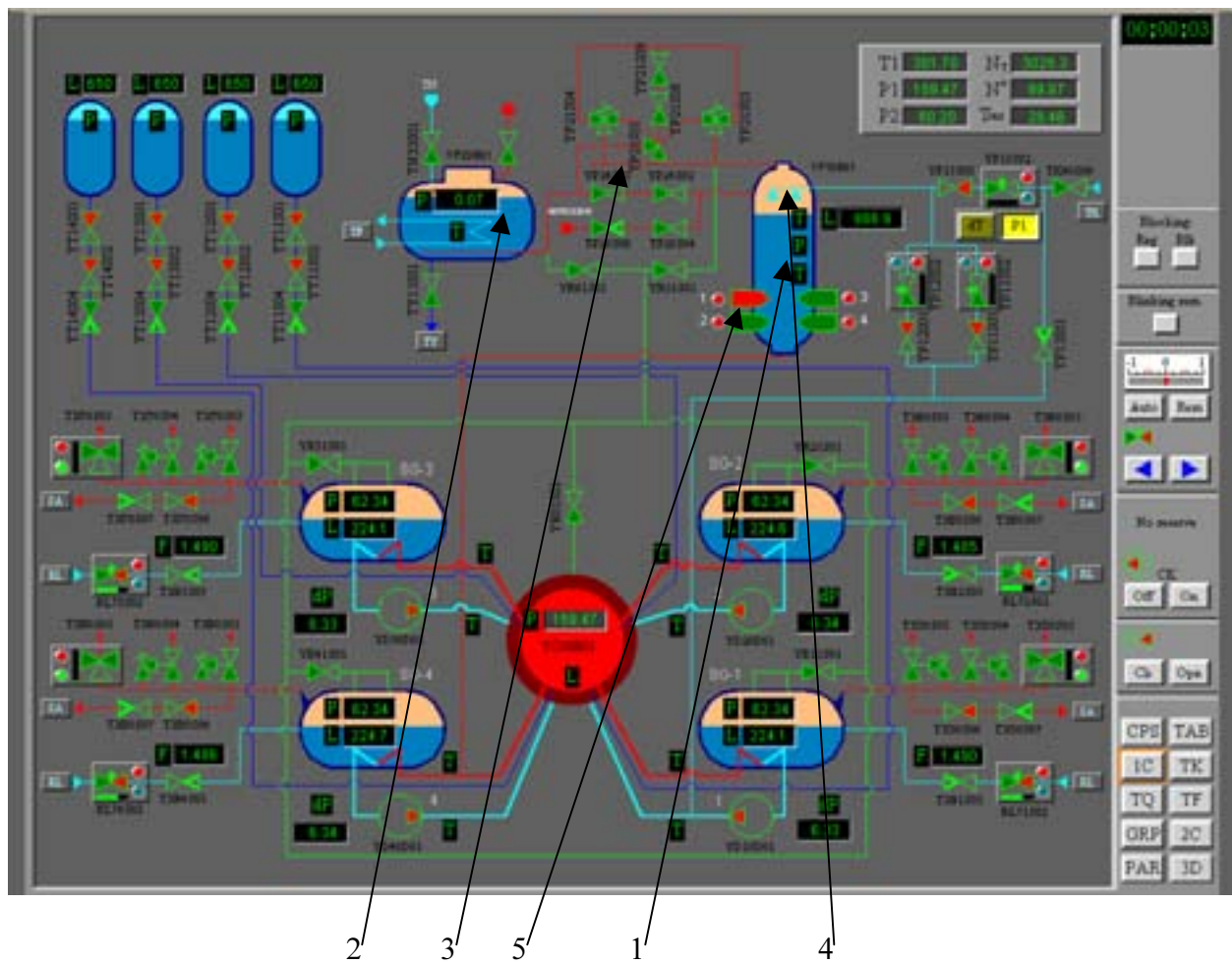


FIG. 9. WWER-1000 primary circuit pressure control equipment.

1. Pressurizer.
2. Bubbler.
3. Steam dumping pipeline with impulse safety valves and relief valves.
4. Spray system and injection pipeline.
5. Electrical heaters.
6. Connecting pipeline.

Pressure change compensation is performed as follows:

Water in the pressurizer is heated by electric heaters located in the bottom part of the pressurizer. The water boils and steam forms a steam cushion in the upper part of the pressurizer. In stationary conditions steam and water are in the saturated state. In transient regimes of the nuclear steam supply system, when the mean temperature of the coolant changes, coolant can flow through the connecting pipeline from or to the pressurizer. In such situations, the steam cushion dampens pressure changes due to evaporation of water and/or condensation of steam. Changes of coolant volume in the pressurizer also play an important role in the pressure compensation process.

In the case of decreasing water volume and respective increasing steam volume, the pressure starts decreasing, but for lower pressure, the water boiling margin decreases too. As a result, water becomes over saturated for the current pressure and water evaporation increases, leading to pressure increase.

When water volume increases and the steam volume shrinks, pressure starts increasing, but for increased pressure, the water saturation temperature is higher. As a result, steam becomes subcooled for the current pressure and steam condensation increases, leading to a pressure decrease. In the case of a significant pressure rise, coolant is injected through the spray system located in the upper part of the pressurizer. This injection of relatively cold water results in condensation of steam in the steam volume and a consequent decrease in pressure.

The level in the pressurizer is maintained by an automatic controller that considers inputs from the current level and the primary circuit mean temperature. Regulation is carried out with valves in the feed and bleed system.

The pressure in the pressurizer is maintained by an automatic controller that considers pressure input from the reactor vessel pressure. Regulation is carried out with electric heaters and with valves at injection pipelines.

In the stationary regime under normal operation conditions, one group of electric heaters is switched on. This group is needed for compensation for heat losses.

During reactor startup when the primary circuit is cold, instead of a steam cushion in the pressurizer, a nitrogen cushion is used. When the primary circuit and pressurizer are heated to operation temperature, nitrogen is replaced by steam.

In the process of reactor cool down, the automatic controller maintains the temperature difference between the primary circuit and the pressurizer by injection of cold water from the “cold leg” of the primary circuit loop or from the feed and bleed system.

1.2.6. Primary circuit feed and bleed system

The primary circuit feed and bleed system is designed for:

- Controlling the inventory of the primary circuit coolant;
- Changing boron concentration in the primary circuit coolant;
- Bleeding leakages from primary circuit equipment;
- Primary circuit coolant purification and return;
- Feeding water to MCP sealing;
- Feeding boron concentrate to the primary circuit in case of electric power loss.

The feed and bleed system equipment presented in the Simulator is shown in Figure 10. It includes:

- two trains for feed water supply to the primary circuit;
- automatic level control in the pressurizer;
- two primary coolant storage tanks;
- bleed deaerator;
- boron concentrate storage tank;

- boron concentrate feed pump;
- pure water storage tank.

Primary circuit inventory control is performed via feed and bleed valves. A pressurizer level below the setpoint indicates that there is insufficient coolant mass in the primary circuit and an additional amount of coolant is then fed into the primary circuit from the primary coolant storage tank until the level in the pressurizer reaches the setpoint. On the contrary, a pressurizer level above the setpoint indicates that there is excess coolant mass in the primary circuit, and the bleed flow increases while the feed flow decreases thus reducing coolant mass in the primary circuit.

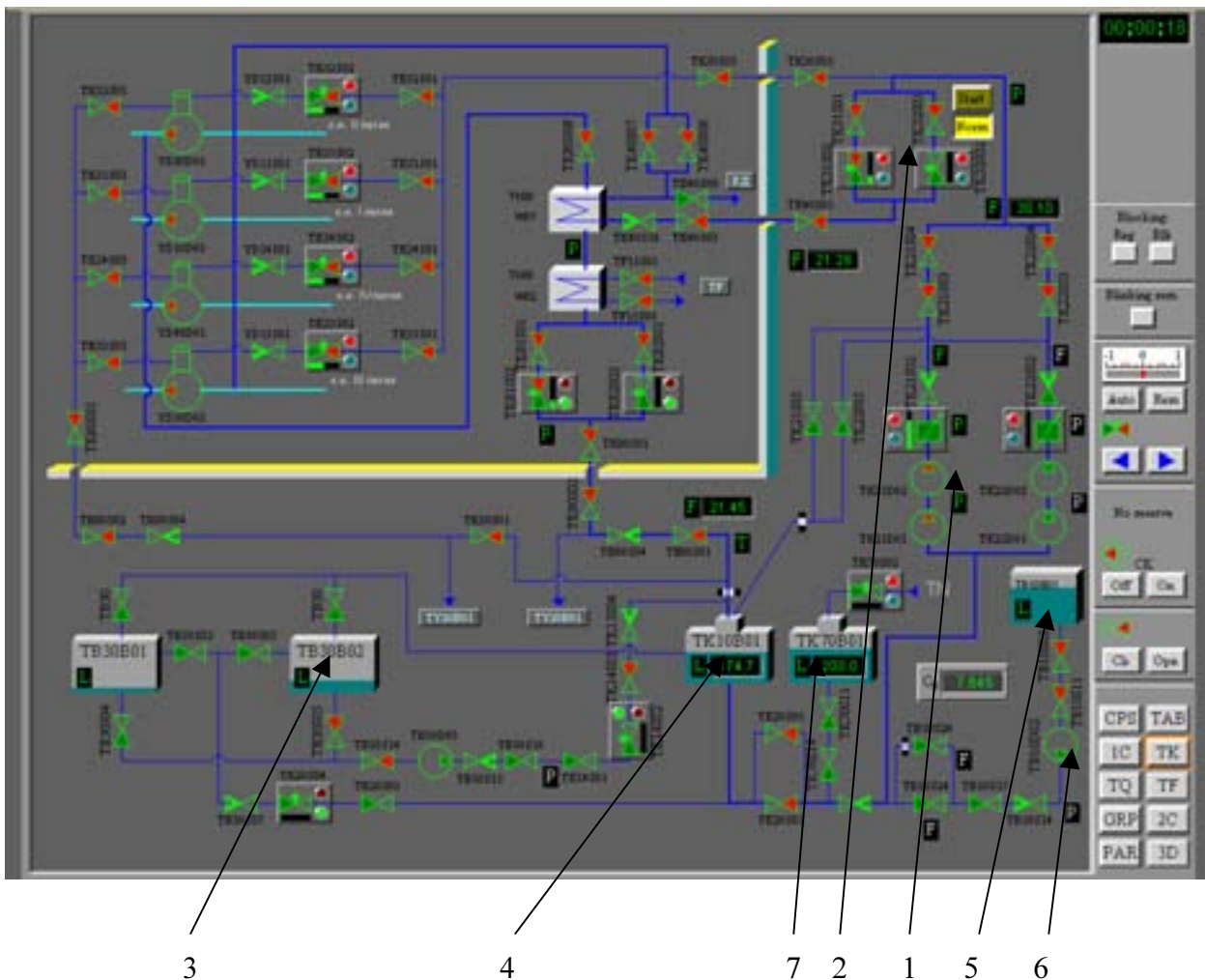


FIG. 10. WWER-1000 feed and bleed system equipment.

- Trains for feed water supply to primary circuit.
- Automatic level control in the pressurizer.
- Primary coolant storage tanks.
- Bleed deaerator.
- Boron concentrate storage tank.
- Boron concentrate feed pump.
- Pure water storage tank.

Bleed coolant from the primary circuit passes to the regenerative heat exchanger where it is cooled by feed water return flow to the primary circuit. After that it is cooled in the auxiliary heat exchanger down to 40-55 °C.

After cooling, the bleed coolant is purified in the low-pressure water purification system. After purification, the coolant flows to the feed water deaerator for degassing from which it is returned to the primary circuit by feed water pumps through the hot side of regenerative heat exchangers. Under normal operation conditions, only one train of feed water pumps is in operation. Control valves provide feed and bleed flows in the range 30-60 t/h.

Part of the feed water goes to MCP sealing. The pressure head of the feed water pumps is about 20 kgs/cm² higher compared to the MCP outlet pressure head. The MCP sealing flow is about 2 m³/h for each MCP.

1.2.7. Secondary circuit

The secondary circuit of the WWER-1000 includes 4 steam generators, steam isolation valves and steam discharge valves, main steam header, turbine, four condensers, feed water heating systems and the feed water supply system.

The steam formed in all steam generators is collected in the main steam header and distributed to a number of loads. Under normal operation conditions most of steam flow goes via turbine governor valves to the high-pressure cylinder of the turbine. Steam exiting the high-pressure cylinder enters the separator to remove extra moisture. After the separator, steam passes to the reheater where it is heated and enters the low-pressure cylinders. Each of the low-pressure cylinders is connected to a separate condenser, where steam is condensed by flowing over tube bundles through which cooling water is circulated. Starting from the condenser, the condensate is pumped through low-pressure and high-pressure reheating heat exchangers respectively, in which it is heated by extraction steam from the turbine. Then the condensate goes to the deaerator in which all non-condensable gases are removed. Then the resulting feed water is pumped to the steam generators by feed water pumps.

The inventory control of the water level in the steam generators is performed by the steam generators' inventory control valves.

The secondary circuit equipment, presented in the Simulator, is shown in Figure 11.

High and low pressure turbine cylinders, separator, reheater, condensers and feed water reheating system are presented in the Simulator in a simplified way.

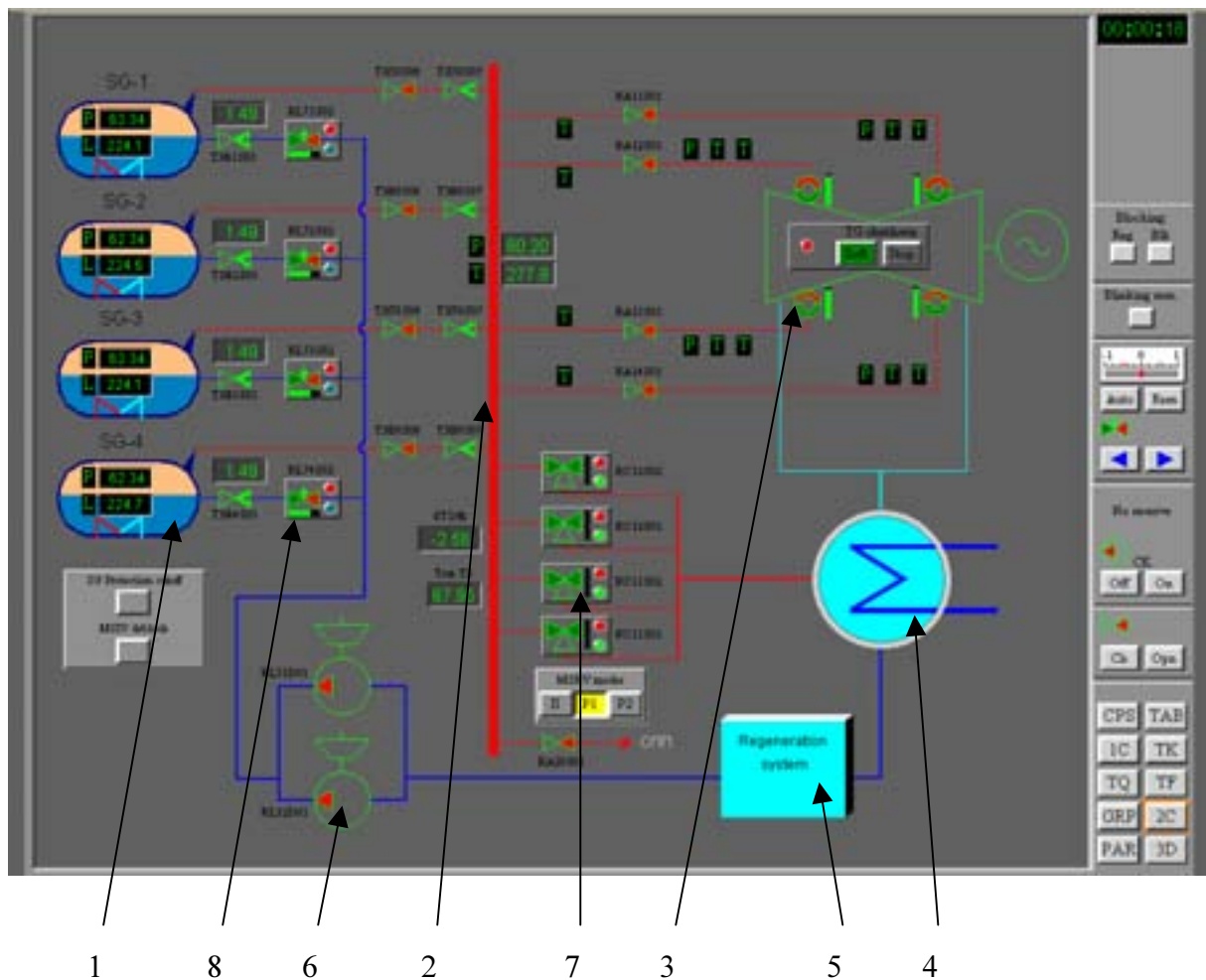


FIG. 11. WWER-1000 secondary circuit equipment presented in the simulator.

1. Steam generators.
2. Main steam header.
3. Turbine governor valves.
4. Condenser.
5. Feed water heating system (regeneration system).
6. Feed water pumps.
7. Condenser steam discharge valves.
8. Steam generator inventory control valves.

1.3. Reactor control and protection system

Regulation of reactor power and control of the fission chain reaction is carried out by two systems adjusting reactivity that are based on different principles:

- Insertion of solid absorbers - control rods system (control & protection system – CPS); and
- Injection of a liquid absorber - boron regulation system.

Reactivity regulation is based on changing the position of the control rods and on changing the boron concentration in the primary circuit coolant.

The reactor control and protection system includes the following components:

- Control Rod System;
- Technological parameters transmitters;
- System for processing signals and data from technological parameters transmitters;
- Neutron flux monitoring system;
- Emergency protection system;
- Preventive protection system including power limiting regulator;
- Control rods positions monitoring system;
- Control rods individual and group control system.

1.3.1. Control rod system

The Control Rod System is intended for:

- Maintaining a critical state at a stationary power level and control of power release distribution in the core;
- Changing reactor power;
- Providing preventive and emergency reactor protection.

In the WWER-1000 there are 61 control rods. Each control rod consists of 18 absorber elements, which move inside fuel assemblies in special channels and which have individual drives.

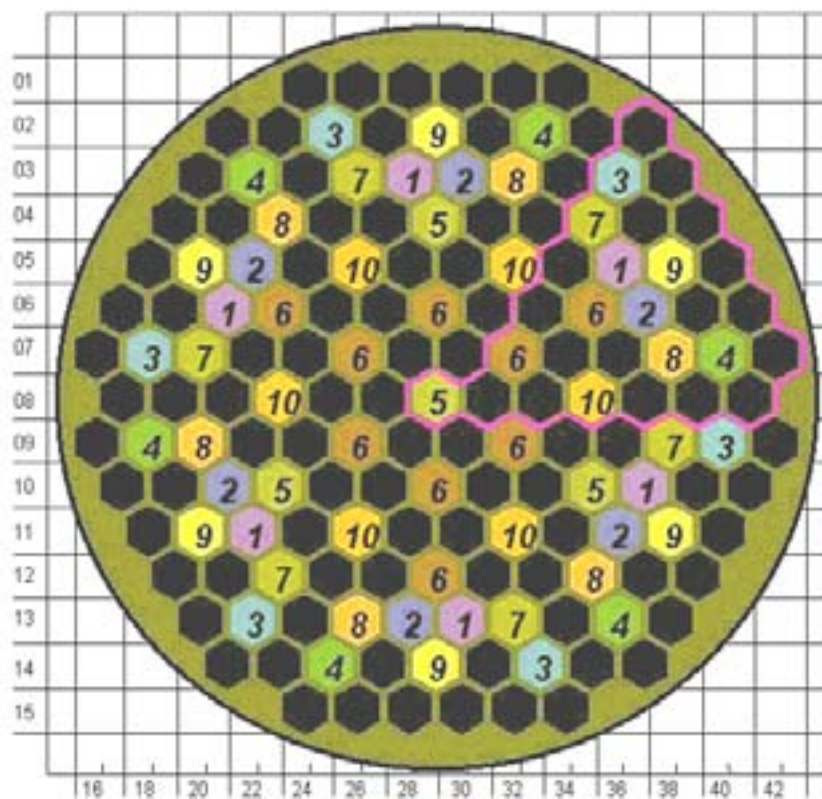
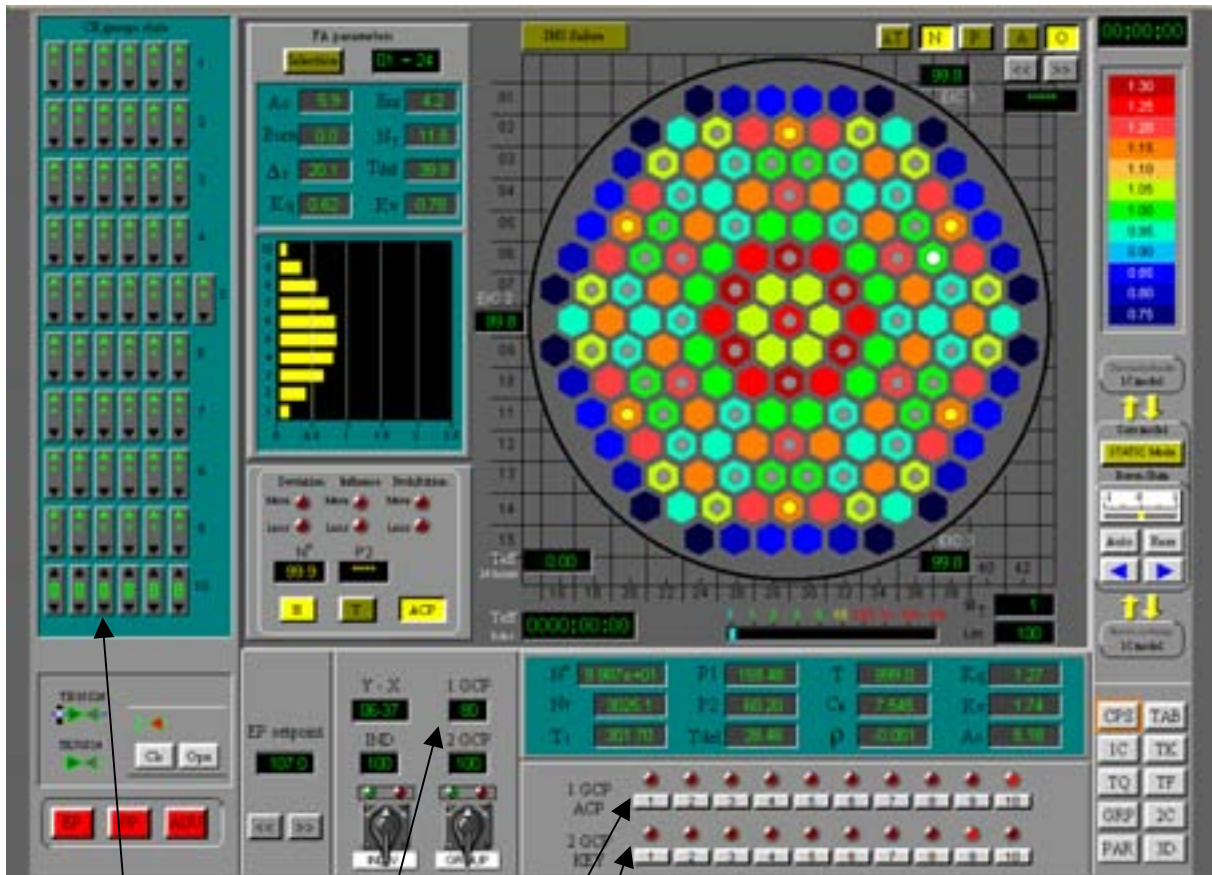


FIG. 12. WWER-1000 control rod map.

Control rod banks are marked with numbers 1-10.

All control rods are subdivided into 10 groups (banks). Distribution of control rods into groups and positions of control rods on the core plan view are shown in the control rod map in Figure 12. The number of control rods in a group, and the velocity of movement, are chosen so that maximal differential efficiency should be more than $0,035 \beta_{\text{eff}}/\text{cm}$ and the reactivity introduction rate in the process of withdrawing control rods should be less than $0,07\beta_{\text{eff}}/\text{s}$, where β_{eff} = efficient fraction of delayed neutrons.

When the reactor is operating at rated power all groups of control rods are in the top position above the core, except for group #10, which is a work group. Typically it is located at a height 70-90% from the bottom and serves for compensation of small changes of reactivity due to oscillations in temperature, boron concentration and electric load.,



1. Control rods' position panel.
2. Individual control rod bank position indicator.
3. Control rod bank selector for automatic power control.
4. Control rod bank selector for manual power control.

FIG. 13. Control rod position indicators.

Control of sparse power release distribution in the core is performed by the control rod group #5. It is used for maintaining power release non-uniformity in designed margins and for stabilization of field shape in maneuvering regimes according to the algorithms of reactor control for xenon transients.

All control rods are used for emergency protection (EP) and preventive protection (PP).

When the EP signal occurs all control rods are dropped into the core and the time taken to reach the bottom position is less than 4 seconds.

After signals of preventive protection occur (they can be of various types) groups of control rods are sequentially moved down into the core, or one group can be dropped for a fast decrease in reactor power, or prohibition of removing control rods from the core is implemented.

Control rods' positions are displayed on panels in the Control Room. In the Simulator, the control rods' positions are presented on the CPS display page (see Figure 13).

Positions of all control rods are displayed on the control rod position panel, while the position of the particular selected control rod bank is displayed on the individual control rod bank position indicator.

Any control rod bank can be selected for automatic or manual power control. This is done with the help of the control rod bank selector for automatic or manual power control respectively.

1.3.2. Boron regulation system

The boron regulation system is intended for compensation of slow reactivity changes and maintaining the reactor critical state during transients concerned with xenon poisoning, and during reactor startup and shutdown.

In normal operation regimes boron regulation provides:

- Compensation for slow reactivity changes associated with fuel burnup and xenon transient processes;
- Compensation for reactivity increase during xenon decay and reactor cool down;
- Required subcriticality during core refueling and planned maintenance work.

Boron regulation can also be used for changing reactor power. The absorption capabilities of boron regulation were chosen taking into account the following two requirements:

- To compensate for full reactivity in the cold state; and
- To provide subcriticality not less than 0.05%, without taking into account control rods.

This requirement is satisfied when the boron concentration in the coolant is 2.8 g/kg (16g/kg of boric acid).

The boron regulation system permits changes in boron concentration at a rate of about 15-20% per hour from the rated concentration.

A combination of boron regulation with the control rod movement improves the maneuvering characteristics of the reactor.

1.3.3. Neutron flux monitoring system

The neutron flux monitoring system (on WWER reactors it is called AKNP) is intended for monitoring neutron flux and determining neutron power and reactor period. It provides comparison with setpoints for neutron power and reactor period, generates signals for the control and protection system, for the power-limiting regulator (ROM), and for the control room panels.

The neutron flux monitoring system consists of three parts:

- Neutron flux monitoring system for Control and Protection System;
- Neutron flux monitoring system for core refueling;
- Neutron flux monitoring system for standby Control Panel.

The last two parts are not presented in the Simulator and are not considered in this manual.

The neutron flux monitoring system for CPS provides permanent monitoring of neutron flux beginning from the reactor subcritical state and at all power levels. The system consists of two independent trains, each having its own detectors in adjacent channels of ex-core ionizing chambers. The range of measurements is subdivided into three intervals: source range, intermediate range and power range, where

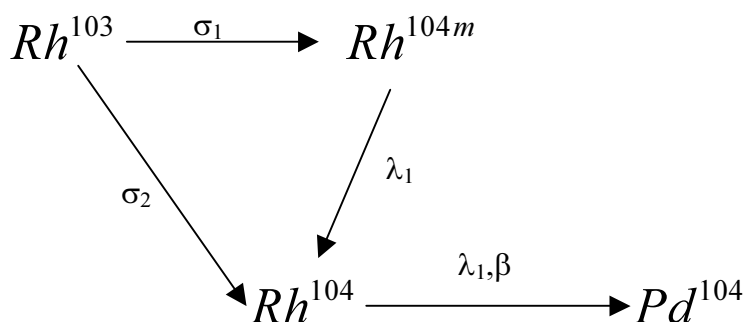
- Source range (SR) $(1,0 \times 10^{-7} - 0,1) \% N_{nom} / (3,0 \times 10^3 - 3,0 \times 10^9)$ neutron/s m^2 ;
- Intermediate range (IR) $(1,0 \times 10^{-3} - 100) \% N_{nom} / (3,0 \times 10^7 - 3,0 \times 10^{12})$ neutron/s m^2 ;
- Power range (PR) $(0,1 - 120) \% N_{nom} / (3,0 \times 10^9 - 3,0 \times 10^{12})$ neutron/s m^2 .

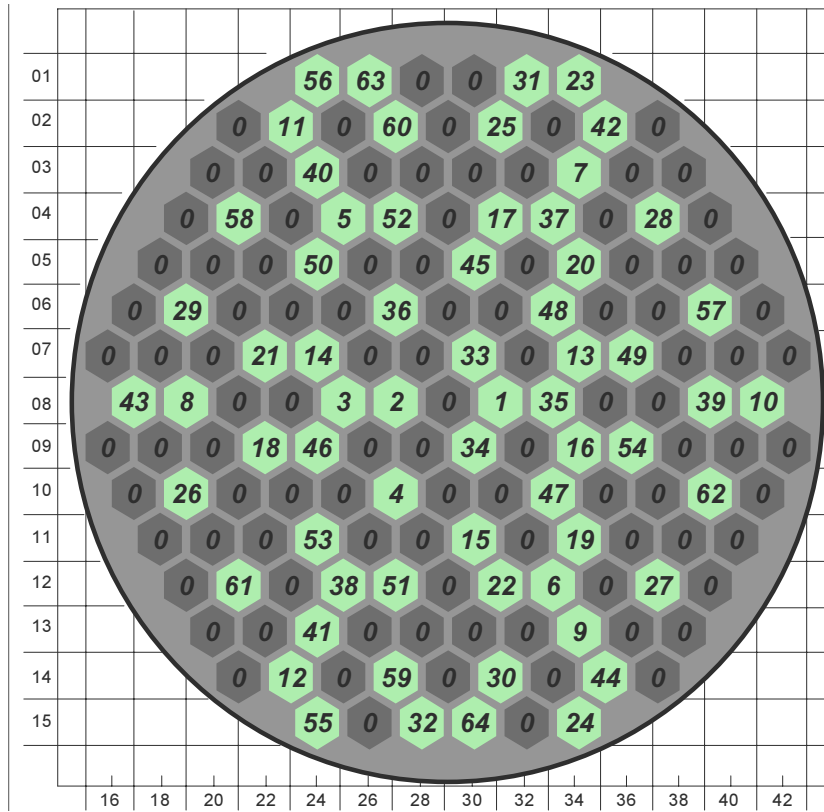
AKNP provides monitoring of the reactor period in the interval from 10 to 200 sec.

1.3.4. In-core instrumentation

Neutron flux measuring channels (NFMC) are intended for monitoring of neutron flux in radial and axial directions in the reactor core. The NFMC assembly is an assembly of neutron flux measuring instruments installed in a protective cover.

The NFMC assembly is placed within the fuel assembly central tube and guiding channel of the protective tube block. In the upper part of the NFMC, an electric connector with cable is mounted. Signals from direct-charge sensors made from rhodium are transmitted to measuring instrumentation. The transformation diagram from neutron capture to electric signal is presented below.





Neutron flux measuring channels are marked with numbers 1-64

FIG. 14. Map of neutron flux measuring channels in the core.

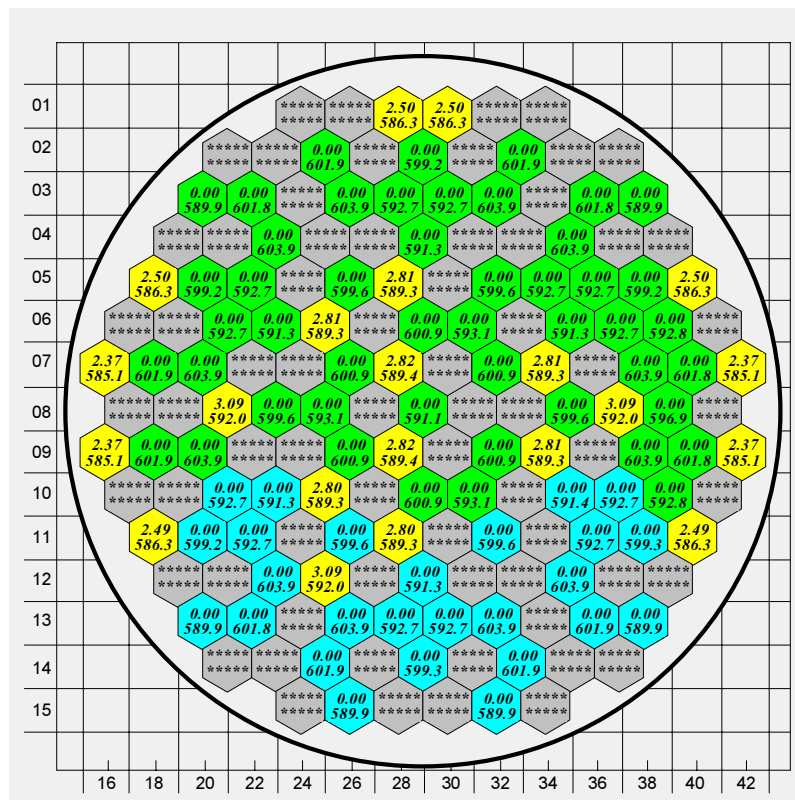


FIG. 15. Map of temperature transmitters location on the plan view of WWER-1000 core

Temperature transmitters are marked with blue, green and yellow colors.

The total number of NFMC assemblies in the WWER-1000 reactor core is 64. Signals from 64 NFMC assemblies give representative information on the neutron flux density distribution in the core in the axial and radial directions. The map of NFMC positions on the reactor core (plan view) is shown in Figure 14.

Besides neutron flux transmitters in the in-core measuring system, temperature monitoring transmitters are used. Ninety one transmitters are installed at the outlet of 91 fuel assemblies. Their readings are used for determination of individual fuel assembly power and for precise determination of reactor thermal power. The map of temperature transmitters location for measurement of coolant temperature at the outlet of the fuel assemblies is shown in Figure 15.

1.3.5. Control and protection system operation

For reactor protection and control the following systems are used: AKNP, ARM, ROM, URB, AZ. All of them adjust the reactivity via control rod systems.

All control rods are subdivided into 10 groups, having 6 control rods in each group, except group No. 5 which consists of 7 control rods. Control rods can be moved individually and in groups. There are predefined sequences of control rods movement by groups: “up” – from a lower number to a higher number, “down” – from a higher number to a lower number.

Electromagnetic forces suspend control rods above the reactor core. When an emergency protection signal occurs (in a WWER it is called AZ signal) all control rods drop down. The power supply to the electromagnetic locks is cut off, so the control rods drop due to gravity. The drop time for full insertion is less than 4 seconds.

The complete list of emergency protection signals and their possible reasons are listed in Table I.

TABLE I. LIST OF SIGNALS FOR EMERGENCY PROTECTION (REACTOR SCRAM)

Signal name	Action	Possible reason
Reactor period in source range is less than 10 sec.	Reactor scram	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods. 3. Ejection of control rod due to rupture of its cover.
Reactor period in intermediate range is less than 10 sec.	Reactor scram	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods. 3. Ejection of control rod due to rupture of its cover.

Signal name	Action	Possible reason
Reactor period in power range is less than 10 sec	Reactor scram	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods. 3. Ejection of control rod due to rupture of its cover.
Neutron flux density in power range. N _{set} : setpoint for neutron power – is set by operator according to technical regulations. It can vary. Setpoint N _{set} . is set discretely with step 1% in the range 3-107% N _{nom} .	Reactor scram if reactor neutron power is greater than 107%N _{set}	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods.
Neutron flux density in power range	Reactor scram if reactor neutron power is greater than 107%N _{set}	Uncontrolled withdrawal of control rods.
Neutron flux density in intermediate range Setpoint can vary. Set by operator.	Reactor scram if reactor neutron power is greater than setpoint	Wrong actions of operator.
Neutron flux density in source range. Setpoint can vary. Set by operator.	Reactor scram if reactor neutron power is greater than setpoint.	Wrong actions of operator.
Concurrence of signals: 1) Core pressure less than 14,7 MPa (150 atm) and reactor power is greater 75%N _{set} . or 2) Coolant temperature in hot legs is greater 330 ⁰ C and core pressure is less than 13,72 MPa (140 atm).	Reactor scram in case of primary circuit pressure lowering	1. Unintended injection to pressurizer. 2. Unintended actuation of pressurizer safety valve. 3. LOCA.
Margin to boiling in primary circuit is less than 10 ⁰ C for each loop	Reactor scram due to increasing coolant temperature	Various.
Lowering pressure drop at MCP from 0,39 MPa to 0,25 MPa for time less 5c	Reactor scram due to low coolant flow rate	1. MCP stuck. 2. Disconnection of MCP shaft.
Concurrence of signals for each of steam lines: 1) Pressure in steam line is less than 4,9 MPa (50 kgs/cm ²). 2) Difference of primary circuit boiling temperature and steam temperature in secondary circuit steam line is greater 75 C	Reactor scram in case of rapid lowering of pressure in steam line	1. Steam line rupture. 2. Feed water line rupture. 3. Non-closing of relief valves.

Signal name	Action	Possible reason
MCP loss of power supply: <ul style="list-style-type: none"> • 1 of 2 operating MCPs while reactor power greater than 5% N_{nom}; • 2 of 4 operating MCPs while reactor power greater than 75%N_{nom}.with delay 6 sec. 	Reactor scram in case of lowering coolant flow rate	MCP trip
Pressure in any of 4 SG greater than 7,84 MPa (80 kgs/cm ²)	Reactor scram	Turbine trip under conditions when relief valves cannot provide rated dump of steam.
Seism with intensity about 6	Reactor scram	Seism is more than designed basis
Loss of reliable power supply for CPS in 2 sections of 3	Reactor scram	CPS loss of power
SG level is lower than 650mm while appropriate MCP is operating.	Reactor scram due to loss of rated feed water flow.	1. Trip of feed water pumps. 2. Rupture of feed water pipelines.
Pressure in primary circuit greater 180 kgs/cm ²	Reactor scram due to unintended pressure rise in primary circuit.	Various
Coolant temperature in any of hot legs $t_{set}+8^{\circ}C$	Reactor scram due to unintended rise of coolant temperature.	Various
Level in pressurizer is less than 4600 mm.	Reactor scram	LOCA
Actuating by key (control room panel)	Reactor scram by operator	Technical specification requirements

The preventive protection system is intended for generation of signals for prohibition of increasing power or for reducing power to the safety margin. The preventive protection system may generate the following commands:

- Sequential movement of control rod groups with normal speed until disappearance of the signal that some parameter exceeded set margin;
- Prohibition of increasing power level until disappearance of the signal that some parameter exceeded set margin;
- Drop down of one group of control rods.

When preventive protection of level #1 signal occurs (in a WWER it is called PZ-1 signal) groups of control rods are moved down with a speed of 2cm/s in standard sequence until the PZ-1 signal disappears.

When a level #2 preventive protection signal occurs (in a WWER it is called a PZ-2 signal), plant automation prohibits control rod withdrawal, i.e. it is prohibited to raise reactor power until the PZ-2 signal disappears.

The complete set of preventive protection signals and actions is listed in Tables II and III.

TABLE II. LIST OF SIGNALS FOR PREVENTIVE PROTECTION PZ-1

Signal name	Action
Reactor period: <ul style="list-style-type: none"> • in source range, less than 20 s • in intermediate range, less than 20 s • in power range, less than 20 s 	Decrease of reactor period
Neutron flux in power range is greater than setpoint. Setpoint is in the range 104-107 from Nset	Decrease of reactor power in case of unintended loss of neutron power
Neutron flux in intermediate range is greater than setpoint. Setpoint is settled in a ratio 10:15 from Nset	Decrease of reactor power during reactor startup
Thermal power is greater than permitted for current number of operating MCPs	Decrease of reactor power with ROM
Pressure above reactor core is greater than 16,8 MPa (172 kgs/cm ²)	Decrease of reactor power due to unintended pressure rise
Coolant temperature in any loop $t_{nom} + 3^{\circ}C$	Decrease of reactor power due to unintended rise of coolant temperature
Pressure in main steam collector is greater than 6,85 MPa (70 kgs/cm ²)	Decrease of reactor power due to unintended steam pressure rise
Loss of power supply: <ul style="list-style-type: none"> • 1 MCP out of 4 in operation • 1 MCP out of 3 in operation 	Decrease of reactor power due to decreasing of flow rate through the core
PZ-1 from key (control room panel)	Decrease of reactor power by operator from control panel
Trip of 1 from 2 FWP in operation in the secondary circuit	Decrease of reactor power to 75% Nnom with ROM

A function of preventive protection is also carried out by the reactor power reducing and limiting device (ROM). Functionally, it is a component of the PZ-1 system. The permitted reactor power value is programmed as a function of several parameters. Reactor power is restricted in dependence on the number of main circulation pumps currently in operation, number of feed water pumps currently in operation in the BOP system, grid frequency and a number of other signals. Decrease of reactor power is carried out by movement of the selected work group of control rods down with normal speed.

TABLE III. LIST OF SIGNALS FOR PREVENTIVE PROTECTION PZ-2

Signal name	Action
Neutron flux value is in the source range. Setpoint variable is settled as a ratio 10:15 to the set value.	Limitation of reactor power rise during reactor startup.
Pressure above the core is greater than 16.2 MPa (165 kgs/cm ²)	Limitation of reactor power rise in case of raising pressure and temperature in primary circuit
Drop of one control rod	Limitation of reactor power rise to ensure permitted values of neutron flux non-uniformity.

ROM generates a signal for restriction of reactor power in the following cases:

- 102% N nom.- when 4 MCP and 2 FWP are in operation;
- 69% N nom. - when 3 MCP and 2 FWP are in operation;
- 52% N nom. - when 4 MCP and 1 FWP are in operation;
- 52% N nom. - when 2 MCP in opposite loops and 2 FWP are in operation;
- 42% N nom. - when 2 MCP in adjacent loops and 1 FWP are in operation;
- 7% N nom. – both FWP are not operating;
- Main circulation pump electric power supply frequency is less than 49 Hz.

In the case that the MCP electric power supply frequency is less than 49 Hz at 3 of 4 sections of power supply, a factor 0.9 is applied to the above margins.

Reduction of reactor power is carried out to 2% lower than the predefined power level.

The automatic regulator of reactor power (in a WWER it is called ARM) is intended for maintaining reactor power corresponding to the turbine-generator power and according to signals from the neutron flux monitoring system (AKNP). The automatic regulation of reactor power can be set in the following two operational modes:

- Mode of maintaining constant pressure before turbine regulating valves in the range 10 - 102% Nnom. (mode "T");
- Mode of maintaining constant neutron power in the core in the range 3 - 102% Nnom (mode "H").

Automatic change to mode "H" is carried out the following occurs

- neutron power rise 2% above the prescribed value;
- generation of any PZ signal.

Automatic change to mode "T" is carried out when the pressure in the main steam collector exceeds the set value by 2 kg/cm².

Prohibition for increasing reactor power by ARM is introduced when:

- neutron power reaches 102% from rated value;
- reactor period is less than 40 sec.

In WWER reactors there is also a specific protection which initiates a fast change to a lower power value (in a WWER it is called URB).

URB actuates while the reactor is operating at 75%Nnom in the following situations:

- Trip of 2 of 4 operating MCPs - with a delay of 1.4 sec, reactor power is reduced 50%Nnom, if tripped MCPs are in opposite loops, and to 40%Nnom, if tripped MCPs are in adjacent loops;
- Trip of 1 of 2 operating FWP (closing of FWP stop valves or reducing steam pressure after regulating valves to setpoint) - reactor power is reduced to 50%Nnom;
- Turbogenerator load reduces to zero (disconnection of generator from electric grid) - reactor power is reduced to 40%Nnom;
- Disconnection of turbine by steam (don't understand this? (closing of two turbine stop valves) reactor power is reduced to 40%Nnom.

Until the inserted CR group is withdrawn, these power margins should not be inserted. Rated positions of control rods should be restored within 3 hours Don't understand If it is not possible – hot shutdown of the unit should be carried out.

1.4. Safety systems

Safety systems of the WWER-1000 include a number of special systems which do not take part in normal plant operation These safety systems are poised to act. should the reactor control systems and the reactor operators be unable to maintain reactor operating parameters within in the prescribed limits.

In the Simulator the following safety systems are presented:

- high-pressure part of emergency core cooling system,
- primary circuit over-pressure protection system,
- secondary circuit over-pressure protection system,
- control and protection system,
- steam generators emergency feed water system.

The emergency core cooling system includes a high pressure part which is intended for fast feed of boron concentrate into the reactor when the reactor pressure drops below 5.9 MPa (60 kgs/cm²).

Boron concentrate (16 g/kg of boric acid) at a temperature between 60 - 70 °C is delivered to the reactor from high pressure tanks via four independent trains. Within the first 30 minutes no Reactor Operator actions are needed. For delivery of coolant to the reactor, the energy of compressed nitrogen is used. There are fast acting stop valves for prevention of nitrogen ingress into the reactor.

The primary circuit over-pressure protection system protects the reactor vessel, pressurizer and primary circuit equipment from pressure rise over specified limits. It includes steam discharge valves on the line from the pressurizer to a bubbler-condenser, and a safety membrane. The pressure in the primary circuit should not exceed the nominal value by more than 15%. If the pressure in the pressurizer rises, the coolant is dumped from the pressurizer to the bubbler through a relief valve. The steam is condensed in the bubbler, and the condensate is pumped out of the bubbler to maintain a constant level in it. If the pressure in the bubbler rises above the safety limit, a membrane breaks and the excess pressure is released to containment. A control valve permits to decrease primary circuit pressure to any defined value by operator command. Set points for impulse valves full opening are from 185 to 192 kg/cm², for closing – from 170 to 175 kgs/cm².

The secondary circuit over-pressure protection system includes steam discharge valves to atmosphere and to condensers, and fast actuating safety valves. Under normal condition the steam flows to the turbine via governor valves. The opening of the governor valves alters steam flow to the turbine, varies turbine load, and affects the steam pressure in the main steam header and steam generators. If the pressure rises above the specified value, the steam discharge valves to atmosphere open and limit the steam pressure rise. If the pressure rises further, the steam discharge valves to the condensers open and the steam bypasses the turbine to the condensers. If the pressure is still too high, the safety relief valves open to ensure that the steam pressure does not exceed the safety limit.

2. SIMULATOR MODEL FUNDAMENTALS

2.1. Mathematical model of core neutron kinetics

2.1.1 List of variables

\hat{A} - transfer operator matrix;

\vec{C}_i - vector of concentrations for predecessors of delayed neutrons $i=1,6$;

\hat{F} - fission operator matrix;

$\overline{F\Phi}$ - mean fission velocity for the reactor;

$\vec{\Phi}$ - neutron flux density;

β_k^B β_k^G λ_k^B λ_k^G - effective fractions and decay constants for groups of fission products;

β_i λ_i - fraction and decay constant for group i of delayed neutrons;

v - velocity of neutrons;

ω - inverse period from the solution of equation similar to equation of inverse clock.

2.1.2. Theoretical approach

The neutron kinetics model provides a dynamic calculation of the power release in the WWER-1000 reactor, and readings of in-core and ex-core measuring devices in real time and in fast modes.

The major contribution to the power release in the WWER reactor is made by the kinetic energy of fission nuclei originating from fission of uranium -235, -238, plutonium -239, -241; radiation capture of γ -quanta generated in the fission process; moderation of fast neutrons; radiation capture of γ -quanta and β -particles emitted from fission fragments (decay power release).

The spatial distribution of the power release and readings of measuring devices are defined by the neutron flux density in the reactor. Calculation of 3D spatial distribution of the neutron field is the basic part of the neutron kinetics model.

The spatial distribution of the neutron flux density is determined by the geometry of the WWER-1000 reactor and the distribution of construction materials.

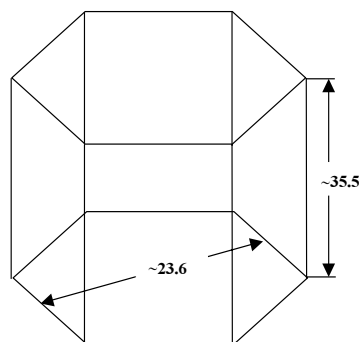


FIG. 16. Control node of kinetic model.

The influence of channels and core construction elements on the neutron flux distribution is simulated by introduction of neutron physics constants as parameters in the equations for the neutron field in the reactor.

Spatial discretization was determined according to the geometry of the WWER-1000 reactor. Average neutron flux density is calculated in 163×10 nodes of spatial mesh: 163 nodes in the radial plane coinciding with the centers of fuel assemblies and 10 nodes in the axial direction with a step of about 35.5 cm. Figure 16 shows the control node of the kinetic model.

The spatial distribution of neutron flux density is described within the frame of single velocity diffusion theory, i.e. all neutrons are combined in one group with an average effective velocity. Neutron physics constants for such an approach are calculated also using the one energy group approximation.

The library of neutron physics constants (NPhC) includes all major core construction materials and elements.

The value of NPhC is influenced by a number of factors. Dependence of NPhC on the following feedback parameters is simulated:

- fuel burnup,
- fuel temperature,
- water temperature,
- water density.

Absorption and fission cross-sections for the fuel assembly are presented in the form of four-dimensional tables with dependence on the above-mentioned parameters. Values of absorption and fission cross-sections are calculated for the node-wise values of feedback parameters.

NPhC between node values of feedback parameters are calculated using many-dimensional linear interpolation of table values.

Absorption is also influenced by the xenon and samarium poisoning of the reactor.

Xenon is a product of radioactive decay of iodine that is produced in the process of uranium decay. Samarium is produced similarly from promethium.

Concentrations of iodine, xenon, promethium and samarium are calculated for all nodes in fuel assemblies. The macroscopic cross-sections for neutron absorption by xenon and samarium are determined by multiplication of xenon or samarium concentration by the appropriate microscopic absorption cross-section. Microscopic cross-sections for xenon and samarium are considered constant.

For all control rod construction elements tables of absorption cross-sections were calculated depending on:

- coolant temperature,
- coolant density.

Absorption cross-sections for feedback parameter values other than in the defined nodes are calculated with the help of two-dimensional linear interpolation.

NPhC for each calculated cell with a control rod is defined as a linear combination of NPhC for control rod elements with the weighting fraction according to the weight of the element in the calculated cell.

2.1.3. Neutron flux density calculation

Neutron flux density versus time is described by the equations:

$$-\hat{A}\vec{\Phi} + (1-\beta)\hat{F}\vec{\Phi} + \sum_{i=1}^6 \lambda_i \vec{C}_i = \frac{1}{\nu} \frac{\partial \vec{\Phi}}{\partial \tau} \quad (2.1.1)$$

$$\frac{\partial \vec{C}_i}{\partial \tau} = \beta_i \hat{F}\vec{\Phi} - \lambda_i \vec{C}_i \quad (2.1.2)$$

Where :

- \hat{A} - transfer operator matrix,
- \hat{F} - fission operator matrix,
- \vec{C}_i - vector of concentrations for precursors of delayed neutrons $i=1,6$,
- β_i λ_i - fraction and decay constant for group i of delayed neutrons,
- ν - velocity of neutrons,
- $\vec{\Phi}$ - neutrons flux density.

Neutron flux density is calculated as a product of amplitude function by form-function:

$$\vec{\Phi}(r) = \exp(\omega\tau)\varphi(r, \tau) \quad (2.1.3)$$

Where :

ω - inverse period from the solution of equation similar to equation of inverse clock:

$$1 - \frac{\langle \hat{A}\vec{\Phi} \rangle}{\langle \hat{F}\vec{\Phi} \rangle} + \sum_{i=1}^6 \lambda_i \frac{\langle \vec{C}_i \rangle}{\langle \hat{F}\vec{\Phi} \rangle} \exp[-(\lambda_i + \omega)\Delta\tau] - \sum_{i=1}^6 \beta_i \exp[-(\lambda_i + \omega)\Delta\tau] = \omega \left[\frac{\langle \frac{1}{\nu}\vec{\Phi} \rangle}{\langle \hat{F}\vec{\Phi} \rangle} + \sum_{i=1}^6 \beta_i \frac{1 - \exp[-(\lambda_i + \omega)\Delta\tau]}{\lambda_i + \omega} \right] \quad (2.1.4)$$

The form-function is calculated at every time step using upper relaxation method.

2.1.4. Power release calculation

Energy released during fission of nuclei consists of the following fractions:

- kinetic energy of fission fragments, \mathcal{E}_{kin}
- kinetic energy of neutrons generated during the fission process, \mathcal{E}_n
- energy of γ -quantum, irradiated during fission, $\mathcal{E}_{\gamma 1}$

- energy of γ -quantum, irradiated by fission products, $\mathcal{E}_{\gamma 2}$
- energy of β -particles, irradiated by fission products, \mathcal{E}_{β}
- energy of anti-neutrino \mathcal{E}_{ν} .

It is considered that the following energy is not released in the reactor: energy entrained by anti-neutrinos and energy released by the fission products with very long half life (more than 3 years - it is considered that this energy is released after unloading of fuel). For U^{235} these parameters are presented in Tables IV, V and VI.

TABLE IV.

Power release fractions	Energy, MeV
\mathcal{E}_{kin}	166.2
\mathcal{E}_n	4.8
$\mathcal{E}_{\gamma 1}$	8.
$\mathcal{E}_{\gamma 2}$	5.29
\mathcal{E}_{β}	5.53
\mathcal{E}_{ν}	0.6

Also, $\nu - 1$ neutron is captured in the reactor to release radioactive capture energy w , the values of capture energy are presented in the table below.

TABLE V.

Material	w , MeV.
Uranium-235	6.54
Hydrogen	2.22
Oxygen	3.38

The kinetic energy of fission fragments is released in the fuel assemblies:

$$F_k \Phi_k \mathcal{E}_{kin}, \quad k=1, 163 \times 10 \quad (2.1.5)$$

The energy of γ -quanta is released everywhere. The mass coefficient of γ -quanta relaxation is the following:

TABLE VI.

Material	Energy of γ -quantum, MeV		
	1	3	6
Uranium	0.0779	0.0435	0.0471
Water	0.0636	0.0357	0.0251

The energy release due to the delayed γ -quanta is considered uniform for the whole reactor.

The kinetic energy of neutrons is released during the process of slowing down neutrons. It is not possible to calculate this value within the frame of the one energy group approximation. That is why some estimation calculations for different channels were carried out using the GETERA code in the two energy groups approximation. As a result, approximate relations for velocities of moderation and absorption have been obtained for the whole cell and for the individual elements as well: for coolant, fuel and absorber element. Having these results, the energy release due to moderation and absorption of neutrons in fuel, coolant, control rods, etc is calculated as follows:

$$\text{for fuel (absorber)} \quad \Sigma_a \Phi \left[\frac{S_{1 \rightarrow 2}^{f(ABS)}}{S_{ABS}} \frac{\epsilon_n}{\nu} + \frac{S_C^{f(ABS)}}{S_{ABS}} w \right] \quad (2.1.6)$$

$$\text{for coolant} \quad \Sigma_a \Phi \left[\frac{S_{1 \rightarrow 2}^{water}}{S_{ABS}} \frac{\epsilon_n}{\nu} + \frac{S_C^{water}}{S_{ABS}} w \right] \quad (2.1.7)$$

Where $S_{1 \rightarrow 2}$, S_{ABS} , S_C - velocities of moderation, absorption and radioactive capture respectively, are estimated with the GETERA code.

Residual energy release consists of residual energy release of γ -quantum E_G and β -particles E_B . It is assumed that energy of γ -quanta is released uniformly in the whole reactor core, and energy of β -particles is released uniformly in the fuel.

Dependence of residual energy release on time is described with equations:

$$\begin{aligned} \frac{\partial E^k_B}{\partial \tau} &= \beta_k^B \overline{F\Phi} - \lambda_k^B E^k_B \\ \frac{\partial E^k_G}{\partial \tau} &= \beta_k^G \overline{F\Phi} - \lambda_k^G E^k_G \end{aligned} \quad (2.1.8)$$

Where

$\overline{F\Phi}$ - means fission velocity for the reactor,

β_k^B , β_k^G , λ_k^B , λ_k^G - effective fractions and decay constants for MB and MG groups of fission products.

2.1.5. Main equations and methods of solving

Neutron flux density is calculated using the following equations:

$$-\hat{A}\vec{\Phi} + (1-\beta)\hat{F}\vec{\Phi} + \sum_{i=1}^6 \lambda_i \vec{C}_i = \frac{1}{\nu} \frac{\partial \vec{\Phi}}{\partial \tau} \quad (2.1.9)$$

$$\frac{\partial \vec{C}_i}{\partial \tau} = \beta_i \hat{F}\vec{\Phi} - \lambda_i \vec{C}_i \quad (2.1.10)$$

Where :

- \hat{A} - matrix of transfer operator,
- \hat{F} - matrix of fission operator,
- \vec{C}_i - vector of concentrations for precursors of delayed neutrons $i=1,6$,
- $\beta_i \lambda_i$ - fraction and decay constant for group I of delayed neutrons,
- ν - velocity of neutrons,
- $\vec{\Phi}$ - neutron flux density.

In finite difference form:

$$\sum_{n=1}^{N_R} (\Phi_n(\tau + d\tau) - \Phi_k(\tau + d\tau)) + \frac{h_R^2}{h_Z^2} \sum_{m=1}^{N_Z} (\Phi_m(\tau + d\tau) - \Phi_k(\tau + d\tau)) - \frac{h_R^2}{D_{eff_k}} \Sigma_A^k \Phi_k(\tau + d\tau) + (1 - \beta) \frac{h_R^2}{D_{eff_k}} \nu \Sigma_F^k \Phi_k(\tau + d\tau) + \sum_{i=1}^6 \lambda_i C_i(\tau + d\tau) = \frac{h_R^2}{D_{eff_k}} \frac{1}{\nu} \frac{\Phi_k(\tau + d\tau) - \Phi_k(\tau)}{d\tau} \quad (2.1.11)$$

To simplify the equation (2.2.11) new parameters are introduced:

$$\Sigma_A^{k*} = \frac{h_R^2}{D_{eff_k}} \Sigma_A^k \quad (2.1.12)$$

$$\nu \Sigma_F^{k*} = \frac{h_R^2}{D_{eff_k}} \nu \Sigma_F^k \quad (2.1.13)$$

$$\frac{1}{\nu} = \frac{h_R^2}{D_{eff_k}} \frac{1}{\nu} \quad (2.1.14)$$

Values of effective absorption and fission cross-sections are put in the table for all construction elements. The symbol * is omitted below, assuming that effective parameters are used in equations.

Equation (2.1.10) for precursors of delayed neutrons is solved in the following form:

$$C_i^k(\tau + d\tau) = C_i^k(\tau) \exp(-\lambda_i d\tau) + \beta_i \nu \Sigma_F^k \Phi_k(\tau + d\tau) \frac{1 - \exp(-\lambda_i d\tau)}{\lambda_i} \quad (2.1.15)$$

The solution for neutron flux density is obtained in the following form:

$$\Phi_k(\tau + d\tau) = \exp(\omega \cdot d\tau) \cdot \varphi_k(\tau + d\tau) - \text{multiplication of amplitude function by form-function.}$$

Equations for amplitude function (inverse period):

$$1 - \frac{\langle \hat{A} \vec{\Phi} \rangle}{\langle \hat{F} \vec{\Phi} \rangle} + \sum_{i=1}^6 \lambda_i \frac{\langle \vec{C}_i \rangle}{\langle \hat{F} \vec{\Phi} \rangle} \exp[-(\lambda_i + \omega) \Delta \tau] - \sum_{i=1}^6 \beta_i \exp[-(\lambda_i + \omega) \Delta \tau] = \omega \left[\frac{\langle \frac{1}{\nu} \vec{\Phi} \rangle}{\langle \hat{F} \vec{\Phi} \rangle} + \sum_{i=1}^6 \beta_i \frac{1 - \exp[-(\lambda_i + \omega) \Delta \tau]}{\lambda_i + \omega} \right] \quad (2.1.16)$$

Equation for form function:

$$\hat{A}\varphi(\tau + d\tau) + (1 - \beta)\hat{F}\varphi(\tau + d\tau) - \sum_{i=1}^6 \lambda_i \beta_i \hat{F}\varphi(\tau + d\tau) \frac{1 - \exp[-(\lambda_i + \omega) \cdot d\tau]}{\lambda_i + \omega} + \frac{1}{\nu} \left(\omega + \frac{1}{d\tau} \right) \varphi(\tau + d\tau) = \sum_{i=1}^6 \lambda_i C_i(\tau) \exp[-(\lambda_i + \omega) \cdot d\tau] + \frac{1}{\nu} \frac{\varphi(\tau)}{d\tau} \quad (2.1.17)$$

In vector-matrix form, equation for form-function:

$$\hat{M}\vec{\varphi} = \vec{b} \quad (2.1.18)$$

$$\hat{M} = -\hat{L} - \hat{U} + \hat{D} \quad (2.1.19)$$

Numerical solution of equation (2.1.18) is found by iteration:

$$\vec{\varphi}^{n+1} = \hat{D}^{-1}(\hat{L} + \hat{U})\vec{\varphi}^n \quad (2.1.20)$$

Equations for xenon and samarium:

$$\frac{\partial \mathfrak{I}(\tau)}{\partial \tau} = \gamma_{\mathfrak{I}} \nu \Sigma_F \Phi(\tau) - \lambda_{\mathfrak{I}} \mathfrak{I}(\tau) \quad (2.1.21)$$

$$\frac{\partial X(\tau)}{\partial \tau} = \gamma_X \nu \Sigma_F \Phi(\tau) + \lambda_{\mathfrak{I}} \mathfrak{I}(\tau) - \sigma_X \Phi(\tau) X(\tau) - \lambda_X X(\tau) \quad (2.1.22)$$

Equations (2.1.21) -(2.1.22) are solved in the following form:

$$\mathfrak{I}(\tau + d\tau) = \mathfrak{I}(\tau) \exp(-\lambda_{\mathfrak{I}} \cdot d\tau) + \frac{1 - \exp(-\lambda_{\mathfrak{I}} \cdot d\tau)}{\lambda_{\mathfrak{I}}} \gamma_{\mathfrak{I}} \nu \Sigma_F \Phi(\tau + d\tau) \quad (2.1.23)$$

$$X(\tau + d\tau) = X(\tau) + dt \cdot \{ \gamma_X \nu \Sigma_F \Phi(\tau + d\tau) + \lambda_{\mathfrak{I}} \mathfrak{I}(\tau + d\tau) - (\sigma_X \Phi(\tau + d\tau) + \lambda_X) \cdot X(\tau) \}$$

Equations for concentrations of promethium and samarium:

(2.1.24)

$$\frac{\partial \text{Pr}(\tau)}{\partial \tau} = \gamma_{\text{Pr}} \nu \Sigma_F \Phi(\tau) - \lambda_{\text{Pr}} \text{Pr}(\tau) \quad (2.1.25)$$

$$\frac{\partial \text{Sm}(\tau)}{\partial \tau} = \lambda_{\text{Pr}} \text{Pr}(\tau) - \sigma_{\text{Sm}} \Phi(\tau) \text{Sm}(\tau) \quad (2.1.26)$$

Equations (2.1.25) -(2.1.26) are solved in the form:

$$\text{Pr}(\tau + d\tau) = \text{Pr}(\tau) \exp(-\lambda_{\text{Pr}} \cdot d\tau) + \frac{1 - \exp(-\lambda_{\text{Pr}} \cdot d\tau)}{\lambda_{\text{Pr}}} \gamma_{\text{Pr}} \nu \Sigma_F \Phi(\tau + d\tau) \quad (2.1.27)$$

$$\text{Sm}(\tau + d\tau) = \text{Sm}(\tau) + dt \cdot \{ \lambda_{\text{Pr}} \text{Pr}(\tau + d\tau) - \sigma_{\text{Sm}} \Phi(\tau + d\tau) \cdot \text{Sm}(\tau) \} \quad (2.1.28)$$

Equations for calculation residual power release:

$$\frac{\partial E^k_B}{\partial \tau} = \beta_k^B \overline{F\Phi} - \lambda_k^B E^k_B \quad (2.1.29)$$

$$\frac{\partial E^k_G}{\partial \tau} = \beta_k^G \overline{F\Phi} - \lambda_k^G E^k_G \quad (2.1.30)$$

Equations (2.1.29) - (2.1.30) are solved in the form:

$$E_B^k(\tau + d\tau) = E^k(\tau) \exp(-\lambda_B \cdot d\tau) + \frac{1 - \exp(-\lambda_B \cdot d\tau)}{\lambda_B} \beta_B \overline{F\Phi} \quad (2.1.31)$$

$$E_G^k(\tau + d\tau) = E^k(\tau) \exp(-\lambda_G \cdot d\tau) + \frac{1 - \exp(-\lambda_G \cdot d\tau)}{\lambda_G} \beta_G \overline{F\Phi} \quad (2.1.32)$$

2.1.6. Neutron flux instrumentation model

The neutron flux instrumentation model includes the following components:

- fission chambers,
- side ionizing chambers,
- in-core detectors.

The fission chambers are used for monitoring neutron flux density during reactor start up from zero power level to the minimally controlled power level. Readings of fission chambers are simulated by multiplication of the function interpolating neutron flux density in the fission chamber channel by the normalization coefficient of the model. The interpolation function is equal to the calculated neutron flux density in the centers of calculated cells and is equal to zero at the distance of effective addition from the upper edge of the core.

$$F_{FCh} = \alpha_{FCh} * \varphi(z_{FCh}) \quad (2.1.33)$$

Side ionizing chambers (SIC): it is assumed that readings of every side ionizing chamber are influenced by the limited number of fuel assemblies located nearby. Readings of all SIC sections are calculated as a sum of power multiplications in the calculated cell with the fuel assembly, located in the region of particular SIC, by the individual weight coefficient for this cell and by the general normalization coefficient of the model:

$$F_{ICh} = \alpha_{ICh} * \sum_{k \in \Omega} K_k W_k \quad (2.1.34)$$

The in-core detectors model takes into account radiation decay and activation of the sensor material. Processes in the sensor material are presented on Fig 17.

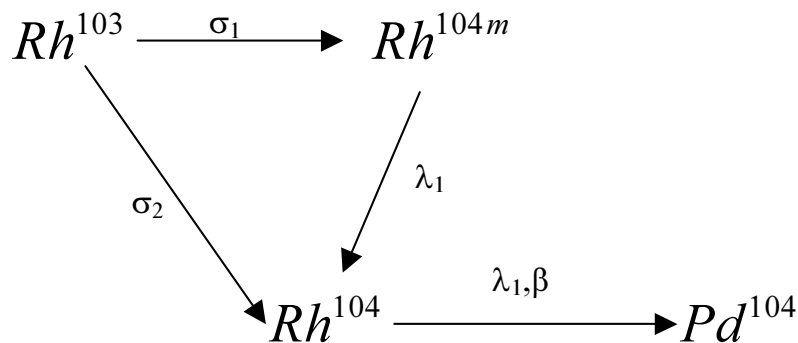


FIG. 17. Rh-detectors activation diagram.

The following system of equations is used to describe the activation of in-core sensors:

$$\begin{aligned}\frac{dC_0}{dt} &= -(\sigma_1 + \sigma_2)\varphi C_0 \\ \frac{dC_1}{dt} &= -\lambda_1 C_1 + \sigma_1\varphi C_0 \\ \frac{dC_2}{dt} &= -\lambda_1 C_1 - \lambda_2 C_2 + \sigma_2\varphi C_0\end{aligned}\tag{2.1.35}$$

where

C_0 – RH₁₀₃ concentration,

C_1 – RH_{104m} concentration,

C_2 – RH₁₀₄ concentration.

The resulting sensor current is calculated with regard to sensor burn-up data.

2.2. Mathematical model of thermal hydraulics

2.2.1. List of variables

Parameters:

A	cross section area , m ²
C_p	heat capacity, J/(kg K)
D_g	hydraulic diameter , m
D_T	thermal diameter, m
I	enthalpy , J/kg
I'	saturated water enthalpy , J/kg
LOS	term associated with losses of pressure head, Pa/(kg/s) ²
M	mass , kg
P	pressure , Pa
P_{cr}	critical pressure , Pa
Q_f	heat generation in fuel node , W
R_{cl_in}	heat resistance of inner part of cladding , m ² K/ W
R_{cl_out}	heat resistance of outer part , m ² K/ W
R_f	heat resistance of fuel , m ² K/ W
R_{gap}	heat resistance of fuel cladding gap , m ² ·K/ W
S_{cl}	outer surface of cladding for node , m ²
S_{fu}	outer surface of fuel for node , m ²
T	coolant temperature, K
T_{cl}	average cladding temperature , K
T_f	average fuel temperature , K
V	volume of node, m ³
W	mass flow rate , kg/s
g	acceleration of gravity, m/s
h	thermal conductivity, W / (m·K)
h_{clad}	thermal conductivity of cladding, W / (m·K)
k_{cc}	heat transfer coefficient (for node) from cladding to coolant , W/K

k_{fc}	heat transfer coefficient (for node) from fuel to cladding , W/K
q_v	volume energy generation , W/m ³
q	heat flux , W/m ²
r_{av}	specific heat of evaporation , J/kg
s	coefficient of phase slipping
w_o	specific velocity of circulation , m/s
w_m	mixture (one phase) velocity , m/s
x	real mass vapor quality
z	height position, m
α	heat transfer coefficient for coolant , W / (m ² K)
Δ	absolute roughness , m
Δz	length of node, m
λ_{fr}	friction coefficient
μ	dynamic viscosity , Pa·s
ξ_{loc}	local resistances hydraulic losses coefficient
ρ	one phase density , kg/m ³
ρ'	saturated water density , kg/m ³
ρ''	saturated vapor density , kg/m ³
ρ_m	mixture (one phase) density , kg/m ³
φ	volume vapour quality

Time indexes:

Δt time step
without index (W) - moment $t + \Delta t$
index ' (W ') - moment t .

Sparse indexes:

i average in node
 l at node boundary

Criteria:

Fr Frude criterion
 Pr Prundtl criterion
 Re Reynolds criterion

2.2.2. Basic equations for coolant

Calculation of coolant parameters in primary and secondary circuits is based on the model of two-phase mixture with phase slipping.

Main equations are one-dimensional conservation equations:

for mass

$$\frac{\partial \rho_m}{\partial t} + \frac{\partial}{\partial z} (\rho_m w_m) = 0; \quad (2.2.1)$$

for momentum

$$\frac{\partial(\rho_m w_m)}{\partial t} + \frac{\partial}{\partial z}(\rho_m w_m w_m) + \frac{\partial P}{\partial z} = -\tau_m + \rho_m g_z; \quad (2.2.2)$$

where τ_m describes losses by friction and local resistances

for energy

$$\frac{\partial(\rho_m I)}{\partial t} + \frac{\partial}{\partial z}(\rho_m w_m I) - \frac{\partial P}{\partial z} = q_v; \quad (2.2.3)$$

and state equation:

$$\rho_m = \rho_m(P, I, w_m, q). \quad (2.2.4)$$

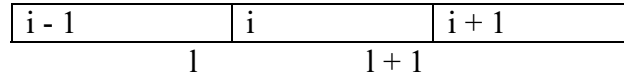
2.2.3. Spatial discretization

To obtain spatial discretization expressions, the initial equations 2.1.1-2.1.3 are integrated by control volume.

Unknown values of pressure P_i , mass flow rate $W_i = A\rho_i w_i$, enthalpy H_i , density ρ_{m_i} at limit points (control volume boundaries) are taken as basics.

Average parameters for the flow paths are written as follows:

$$Y_i = 0,5(Y_i + Y_{i+1})$$



2.2.4. Time discretization

Numerical expressions for time discretization are based on the following completely implicit scheme:

Mass

$$\frac{0,5V_i}{\Delta t}(\rho_{m_{i+1}} + \rho_{m_i} - \rho_{m_{i+1}}^t - \rho_{m_i}^t) + W_{i+1} - W_i = 0 \quad (2.2.5)$$

Momentum

$$\frac{1}{L_{spi}\Delta t}0,5(W_{i+1} + W_i - W_{i+1}^t - W_i^t) + P_{i+1} - P_i = -0,5 \cdot LOS_i(W_{i+1}|W_{i+1}| + W_i|W_i|) + \Delta HGR_i \quad (2.2.6)$$

where

gravitational head

$$\Delta HGR_i = 0,5g(\rho_{m_{i+1}} + \rho_{m_i})(zh_{i+1} - zh_i)$$

zh_i - height position

$$\text{specific length } L_{spi} = \sum_{V_i} \frac{A_k}{\Delta Z_k}$$

The coefficient for hydraulic losses LOS takes into account losses by friction, local resistances and losses by coolant acceleration.

Energy:

$$\frac{0,5 V_i}{\Delta t} (\rho_{m_{l+1}} I_{l+1} + \rho_{m_l} I_l - \rho'_{m_{l+1}} I'_{l+1} - \rho'_{m_l} I'_l) + W_{l+1} I_{l+1} - W_l I_l - 0,5 \frac{V_i}{\Delta t} (P_{l+1} + P_l - P'_{l+1} - P'_l) = Q_{tot_i} \quad (2.2.7)$$

where Q_{tot_i} - total thermal flow, transferred to the coolant at i -control volume and including 2 components:

- energy release in coolant Q_{cool_i} ;
- external heat flow from fuel assemblies/pipelines Q_i :

$$Q_{tot_i} = Q_{cool_i} + Q_i$$

$$Q_i = k_{cc_i} (T_{cl_i} - T_i)$$

2.2.5. Closure correlation equations

Hydraulic losses on calculated segment i include:

- losses by friction;
- losses by local resistances;
- losses by coolant acceleration.

And total pressure drop is calculated as:

$$\Delta P_i = -LOS_i W_i^2 \quad (2.2.8)$$

Hydraulic losses by friction

Hydraulic losses by friction take into account:

- dependence of friction factor on coolant velocity;
- correction factor for non-homogeneity of two-phase mixture;
- design features via local friction factor values.

Losses by friction are calculated as follows:

for a single-phase coolant flow

$$\Delta P_{fr} = \lambda_{fr} \frac{\Delta z}{D_g} \frac{W^2}{2A^2 \rho} \quad (2.2.9)$$

for two-phase coolant flow

$$\Delta P_{fr} = \lambda_{fr} \frac{\Delta z}{D_g} \frac{W^2}{2A^2 \rho'} \left[1 + x \left(\frac{\rho'}{\rho''} - 1 \right) \right] \Psi \quad (2.2.10)$$

where friction factor λ_{fr} is calculated as follows:

for pipes

$$\lambda_{fr} = 0,11 \left[\left(\Delta/D_g \right) + \frac{68}{\text{Re}} \right]^{0,25} \quad (2.2.11)$$

for tube bundles

$$\lambda_{fr} = 0,10 \left[1,46 \left(\Delta/D_g \right) + \frac{100}{\text{Re} \chi} \right]^{0,25} \quad (2.2.12)$$

Δ -absolute roughness;

χ -factor of form.

$$\chi = \frac{2\varepsilon}{(1-\varepsilon)^2} \left(\frac{\varepsilon}{2} - \frac{3}{2} - \frac{\ln \varepsilon}{1-\varepsilon} \right)$$

where ε — relative step of tubes in a bundle.

The non-homogeneity coefficient for two-phase mixture is

$$\Psi = 1 + 0,57 x^{0,125} (1-x)^2 \left[\frac{1}{0,2 + \frac{w_0}{\sqrt{g D_g}} \frac{\rho''}{\rho'}} - 5,2 x^2 \right]; \quad (2.2.13)$$

where circulation speed $w_0 = \frac{W}{A \rho'}$

Hydraulic losses by local resistances

Hydraulic losses by local resistances take into account:

- correction coefficient due to non-homogeneity of two-phase mixture;
- design data for WWER-1000.

Losses by local resistances are calculated as follows:

for single-phase coolant flow

$$\Delta P_{loc} = \xi_{loc} \frac{W^2}{2 A^2 \rho} \quad (2.2.14)$$

for two-phase coolant flow

$$\Delta P_{loc} = \xi_{loc} \left[1 + x \left(\frac{\rho'}{\rho''} - 1 \right) \right] \frac{W^2}{2 A^2 \rho'} \quad (2.2.15)$$

Appropriate approximation tables are used for calculation of local hydraulic resistance coefficients

$$\xi_{loc} = f(\text{Re}) \quad (2.2.16)$$

Hydraulic losses by flow acceleration

Hydraulic losses by acceleration are calculated as follows:

$$\Delta P_{acc} = \left\{ \left[\frac{x^2}{\varphi \rho''} + \frac{(1-x)^2}{(1-\varphi) \rho'} \right]_{out} - \left[\frac{x^2}{\varphi \rho''} + \frac{(1-x)^2}{(1-\varphi) \rho'} \right]_{in} \right\} \frac{W^2}{A^2} \quad (2.2.17)$$

Calculation of hydraulic losses term LOS

Taking into account the structure of expressions (2.2.10) and (2.2.15), the hydraulic losses coefficient LOS is calculated in the following way:

For friction

$$LOS_{fr} = KSI_{fr} \cdot \frac{1}{2A\rho'}$$

Reduced friction losses coefficient

$$KSI_{fr} = \lambda_{fr} \frac{\Delta Z}{D_g} gmix_corr ,$$

$$\text{where } gmix_corr = \begin{cases} \rho' / \rho & \text{- for water} \\ \Psi \cdot \left[1 + x \left(\frac{\rho'}{\rho''} - 1 \right) \right] & \text{- for two-phase area} \\ \frac{\rho'}{\rho} & \text{- for superheated steam} \end{cases}$$

where ρ - water or vapour density, respectively,

Ψ - is calculated with equation (2.2.13).

For local resistances

$$LOS_{loc} = KSI_{loc} \cdot \frac{1}{2A\rho'}$$

Reduced local resistance losses coefficient

$$KSI_{loc} = \xi_{loc} \cdot g2mix_corr ,$$

where

$$g2mix_corr = \begin{cases} \rho' / \rho \\ 1 + x \left(\frac{\rho'}{\rho''} - 1 \right) \\ \frac{\rho'}{\rho} \end{cases}$$

Hydraulic losses due to the acceleration of the flow in terms of (2.2.17) for two phase flow

$$LOS_{acc} = \frac{1}{A^2} \left\{ \left[\frac{x^2}{\varphi \rho''} + \frac{(1-x)^2}{(1-\varphi)\rho'} \right]_{out} - \left[\frac{x^2}{\varphi \rho''} + \frac{(1-x)^2}{(1-\varphi)\rho'} \right]_{in} \right\}$$

for a single phase flow

$$LOS_{acc} = \frac{1}{A^2} \left\{ \left[\frac{1}{\rho} \right]_{out} - \left[\frac{1}{\rho} \right]_{in} \right\}$$

Thermal hydraulic parameters of two-phase mixture

Thermal hydraulic non-equilibrium, which may be caused by high heat rates in the reactor core, is taken into account when thermal hydraulic parameters of the two-phase mixture are calculated.

For enthalpies $I < I_{stb}$ a single-phase coolant (water) is considered.

For enthalpies $I > I''$ a single-phase coolant (steam) is considered.

In the interval $I_{stb} < I < I''$ two-phase mixture flow is considered taking into account absolute mass steam quality and thermal non-equilibrium of the flow.

Enthalpy of the subcooled boiling is calculated as:

$$I = I' - \frac{q}{\rho_m} 7,5 \left(\frac{D_g}{\mu'} \right)^{0,28} \left(\frac{q}{r_{av}} \frac{\rho'}{\rho''} \right)^{0,08} \rho_m^{0,2} \quad (2.2.18)$$

Mass steam quality is:

$$x = (I - I' + dI) / (r_{av} + dI), \text{ where} \quad (2.2.19)$$

$$dI = (I - I_{gst} + I' - I_{gst}) \exp \left(-2 \frac{I - I_{gst}}{I' - I_{gst}} \right)$$

Phase slipping coefficient

$$s = \left[1 + (0,6 + 1,5 \beta^2) \left(1 - \frac{P}{P_{cr}} \right) Fr^{-0,25} \right], \quad (2.2.20)$$

$$\beta = \left[1 + \frac{\rho''}{\rho'} \frac{1-x}{x} \right]^{-1} \quad (2.2.21)$$

Volumetric steam quality:

$$\varphi = \left[1 + S \frac{\rho''}{\rho'} \frac{1-x}{x} \right]^{-1} \quad (2.2.22)$$

Density of two-phase mixture:

$$\rho_m = \varphi \rho'' + (1-\varphi)\rho' \quad (2.2.23)$$

Heat transfer coefficients

Heat transfer to single-phase coolant (convection) is calculated as:

$$\alpha_c = \frac{Nu \cdot h}{D_T}, \text{ where} \quad (2.2.24)$$

$$Nu = 0,023 \cdot Re^{0,8} \cdot Pr^{0,4}$$

D_T - thermal diameter.

Heat transfer, taking into account two phase flow velocity:

$$\alpha_b = \alpha_{pk} \sqrt{1 + 7 \cdot 10^{-9} (w_{mix} \cdot r_{av} \cdot \rho' / q) (\alpha_{00} / \alpha_{pk})^2} \quad (2.2.25)$$

where α_{00} - heat transfer coefficient for bubble boiling in large volume.

$$\alpha_{00} = 4,34q^{0,7} (p^{0,14} + 1,35 \cdot 10^{-2} p^2), \quad [p] = MPa$$

$$\alpha_{pk} = \sqrt{\alpha_c^2 + (0,7 \cdot \alpha_{00})^2}$$

α_c - heat transfer coefficient for one-phase flow, calculated for circulation rate

$$w_0 = \frac{\rho w}{\rho'}$$

Heat transfer for boiling of subcooled water:

$$\alpha = \alpha_b \left\{ 1 + \left[\alpha_b \left(\frac{1}{\alpha_c} - \frac{t_s - t}{q} \right) \right]^{-3/2} \right\}^{2/3} \quad (2.2.26)$$

where $(t_s - t)$ - temperature margin to saturation,

α_c - is calculated according to (2.2.21),

α_b - is calculated according to (2.2.22).

Critical thermal flow is calculated according to following:

$$q_{cr} = \frac{2,899 \cdot r \rho' \left[g D_T (g \alpha')^{1/3} \right]^{1/3} Pr^{1/3} \left(\frac{\rho w D_T}{\mu'} \right)^{0,2} \cdot (0,94 - 0,286X)}{1 + \frac{1}{3,54} \cdot \frac{1}{\Phi'(z)} \int_0^z \Phi'(z) dz} \cdot 10^6 \quad (2.2.27)$$

$$\text{where } X = x \cdot \left[\frac{\rho w^2 \cdot D_T}{\rho' \cdot \sigma} \right]^{0,2};$$

x - mass steam quality;

g - acceleration of gravity;

$\Phi'(z)$ - relative energy release.

Heat transfer coefficient in the over crisis area:

$$\alpha = 0,0811 \cdot \lambda^{0,4376} \cdot Pr^{2,307} \cdot Re^{pow} / \left(D_T^{0,7842} \cdot (1+x)^{2,59} \right) - 13,89 \cdot \exp(1,4504 \cdot 10^{-7} \cdot P \cdot Pr), \quad (2.2.28)$$

$$\text{where } pow = 0,6004 + 0,2456 \cdot \ln(1+x) \quad [p] = MPa$$

2.2.6. Fuel assembly model

For a description of the fuel assembly heat parameters, the following approach is utilized:

- Completely implicit numerical scheme;
- Two geometric zones (first zone - cladding, second - fuel rod).

Figure 18 illustrates the geometric zones of the fuel assembly.

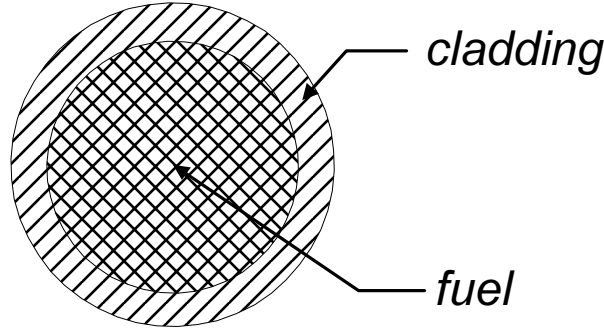


FIG. 18. Geometric zones of fuel assembly.

For correct consideration of thermal conductivity coefficient dependence via temperature, the results of precise stationary thermal model calculations are used (30 radial zones). In this model a two-parameters function for average fuel thermal resistance R_f is calculated taking into account the geometry of the fuel rod. The parameters are: heat flow χ and temperature T_{sf} on the surface of the fuel rod. The average coefficient of fuel thermal resistance at the preliminary stage of data preparation is calculated so that the value of average fuel temperature for the precise model coincides with the one for the simulator model in stationary states. Calculated data for the simulator are stored as a two-dimensional interpolation table of thermal resistance for average fuel temperature $R_f = f(q_{sf}, T_{sf})$. Fuel heat capacity, cladding heat capacity and cladding specific thermal conductivity coefficient dependencies via temperature are taken into account.

Heat balances for the model of fuel assembly are presented as follows:

$$M_f C_{P_f} \frac{dT_f}{dt} = Q_f - k_{fc} (T_f - T_{cl}) \quad (2.2.29)$$

$$M_{cl} C_{P_{cl}} \frac{dT_{cl}}{dt} = k_{fc} (T_f - T_{cl}) - k_{cc} (T_{cl} - T), \quad (2.2.30)$$

where

Q_f - total energy release in fuel rod,

$$k_{fc} = S_{fu} [R_f + R_{gap} + R_{cl_in}]^{-1}$$

$$k_{cc} = S_{cl} \left[\frac{1}{\alpha} + R_{cl_out} \right]^{-1},$$

$$R_{cl_in} = 0.5 D_{fuel} \ln \left[0.5 (D_{fuel} + D_{clad}) / D_{fuel} \right] \frac{1}{h_{clad}},$$

h_{clad} - heat conductivity of cladding,

$$R_{cl_out} = 0.5 D_{clad} \ln \left[D_{clad} / 0.5 (D_{fuel} + D_{clad}) \right] \frac{1}{h_{clad}},$$

D_{fuel} - diameter of fuel rod,

D_{clad} - diameter of cladding.

Fuel heat capacity, cladding heat capacity and cladding specific thermal conductivity coefficient dependencies via temperature are calculated using linear interpolation.

For further use in thermal hydraulic calculations the equations (2.2.29) and (2.2.30) are transformed to:

$$T_f = \Phi 1 + \Phi 2 T_{cl} \quad (2.2.31)$$

$$\Phi 1 = \left(T_f^t + \frac{Q_f \Delta t}{M_f C_{P_f}} \right) / \left(1 + \frac{k_{fc} \Delta t}{M_f C_{P_f}} \right)$$

$$\Phi 2 = \frac{k_{fc} \Delta t}{M_f C_{P_f}} / \left(1 + \frac{k_{fc} \Delta t}{M_f C_{P_f}} \right)$$

$$T_{cl} = \Upsilon 1 + \Upsilon 2 \cdot T \quad (2.2.32)$$

$$AC = 1 + \frac{k_{fc} \Delta t}{M_{cl} C_{P_{cl}}} (1 - \Phi 2) + \frac{k_{cc} \Delta t}{M_{cl} C_{P_{cl}}}$$

$$\Upsilon 1 = \left(T_{cl}^t + \frac{k_{fc} \Delta t}{M_{cl} C_{P_{cl}}} \Phi 1 \right) / AC$$

$$\Upsilon 2 = \frac{k_{cc} \Delta t}{M_{cl} C_{P_{cl}}} / AC$$

To obtain implicit scheme temperature is considered as follows:

$$T_i = T_i^t + \frac{I_i - I_i^t}{C_{P_i}} = T_i^t + 0.5 \frac{I_{l+1} + I_l - I_{l+1}^t - I_l^t}{C_{P_i}}$$

2.2.7. Final form of equations

Finally the equations are written as follows:

Mass

$$W_{l+1} = W_l + S W_{_prev_i} + S W_{_old_i}; \quad (2.2.33)$$

$$S W_{_prev_i} = - \frac{0.5 V_i}{\Delta t} (\rho_{m_{l+1}} + \rho_{m_l});$$

$$S W_{_old_i} = \frac{0.5 V_i}{\Delta t} (\rho_{m_{l+1}}^t + \rho_{m_l}^t)$$

Momentum

$$P_{l+1} = P_l + SP_{prev_i} + SP_{old_i} \quad (2.2.34)$$

$$SP_{prev_i} = DI_i - 0,5 LOS_i (W_{l+1}^2 + W_l^2) - \frac{0,5}{L_{SP_i} \Delta t} (W_{l+1} + W_l)$$

$$SP_{old_i} = \frac{0,5}{L_{SP_i} \Delta t} (W_{l+1}^t + W_l^t)$$

Energy

$$I_{l+1} \left(\rho_{m_{l+1}} + \frac{2 \Delta t}{V_i} W_{l+1} + \mathcal{DT}_i \Delta t \right) = \frac{2 \Delta t}{V_i} \tilde{Q}_{tot_i} + SI_{prev_i} + SI_{old_i} \quad (2.2.35)$$

$$SI_{prev_i} = I_l \left(\frac{2 \Delta t}{V_i} W_l - \rho_{m_l} - \mathcal{DT}_i \Delta t \right) + P_{l+1} + P_l,$$

$$SI_{old_i} = \rho_{m_{l+1}}^t I_{l+1}^t + \rho_{m_l} I_l^t - P_{l+1}^t - P_l^t + \mathcal{DT}_i \cdot (I_{l+1}^t + I_l^t),$$

Where $\mathcal{DT}_i = k_{cc_i} (1 - \mathcal{U}2_i) / (C_{p_i} \cdot V_i),$

$$\tilde{Q}_{tot} = Q_{gr_i} + Q_{cool_i} + k_{cc_i} (\mathcal{U}1_i - (1 - \mathcal{U}2_i) T_i^t).$$

The completely implicit scheme improves regularity and accuracy of the numerical solution.

3. SIMULATOR USER GUIDE

3.1. Introduction

The WWER-1000 Reactor Department Simulator was originally developed for personnel training. It is executed on a personal computer in real time and provides a dynamic response with sufficient fidelity. After reducing the scope of modelling to the systems essential for overall correct response and fidelity and cutting out a number of auxiliary systems the Simulator becomes suitable for educational and information purposes. The present configuration of the Simulator is able to respond to operating conditions normally encountered in WWER-1000 power plant operation.

The interaction between the user and the Simulator is organized through a set of display screens and a mouse. The Simulator display screens partially mimic the actual control panel instrumentation as well as provide additional instruments for analysis. Control panel devices (buttons, switches, keys) are represented as simplified pictures and are operated via individual panels in response to user inputs.

The Simulator User Guide assumes that the user is familiar with the main features of thermal nuclear power reactors and characteristics of the Water-Water Energy Reactor (WWER).

The Simulator can be utilized for achieving various kinds of training objectives. Three groups of training tasks are presented in the task list on the CD. They provide examples of objectives which can be achieved with WWER-1000 Reactor Department Simulator training:

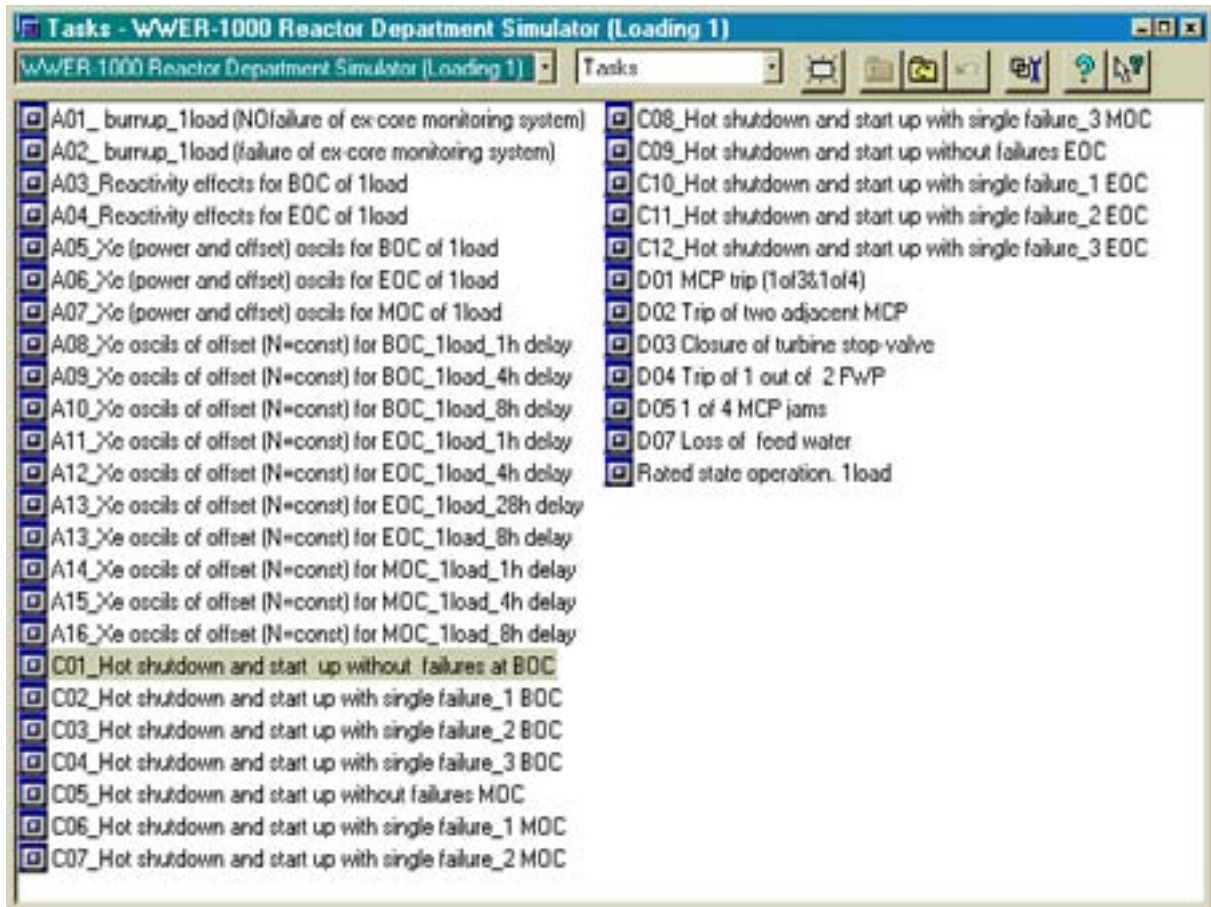
- Group 1 demonstrates various physics phenomena of the reactor core and are intended for understanding of reactor core neutronics.
- Group 2 provides the opportunity to train user diagnostic skills.
- Group 3 demonstrates various transients and is intended for user familiarization with WWER-1000 operation experience.

The present configuration of the Simulator is intended for information and educational purposes and shall not be used for any kind of personnel training.

3.2. Installing, uninstalling and simulator startup

On the CD containing workshop material, please consult the readme.txt file in the *WWER\Simulator* folder.

When the Simulator starts, the following window (or similar) appears:



The main elements of the window are:

1. List of training tasks.

The available list of training tasks is presented in this window.

2. Core loading selector.

Two types of core loading are available in the Simulator: loading 1 with lower enrichment and loading 5 with higher enrichment.

3. File type selector.

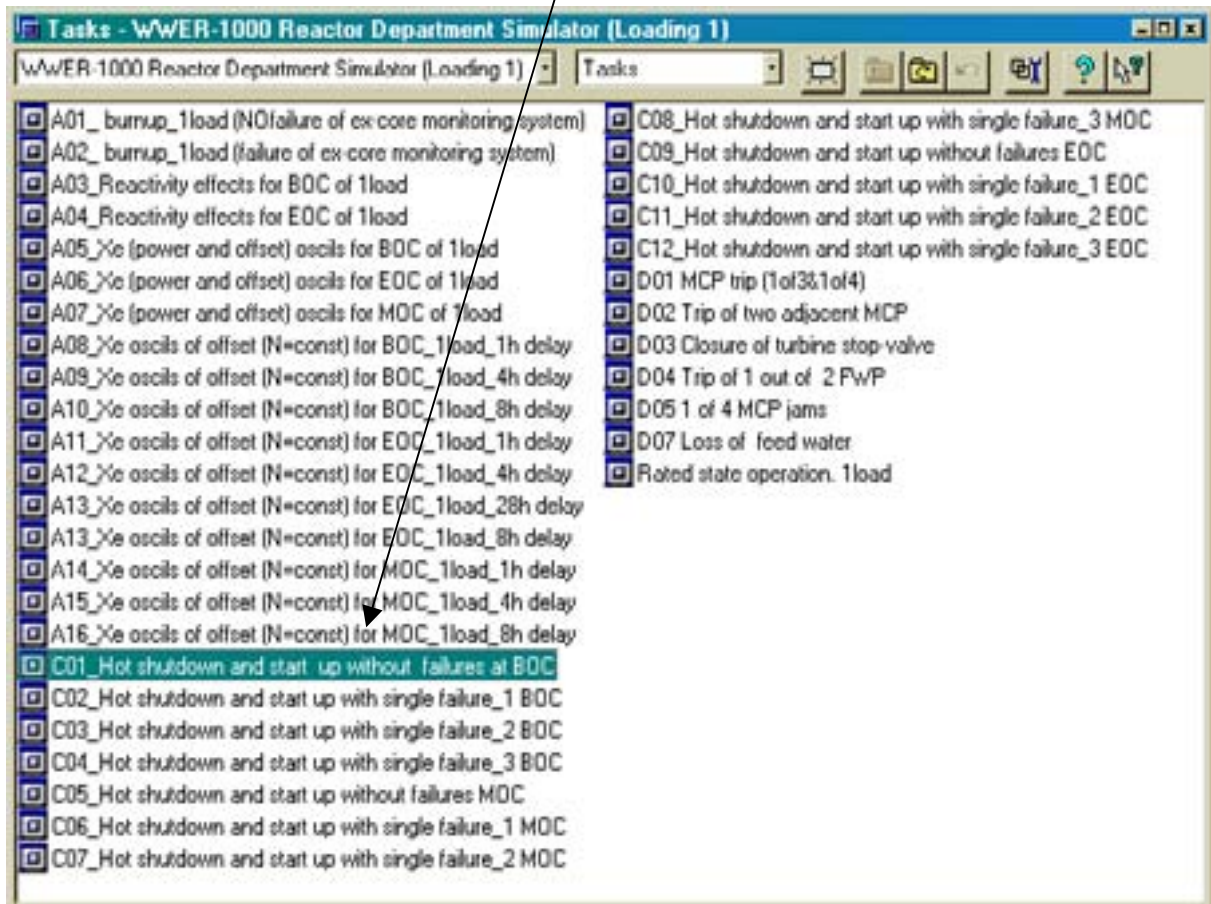
This selector helps to organize the work with different files involved in the Simulator. Selecting between Tasks/Packages/Protocols/Initial conditions/Scripts allows to access correspondingly the lists of available training tasks, predefined packages of tasks, training protocols, Simulator initial conditions and script files.

4. Tool bar.

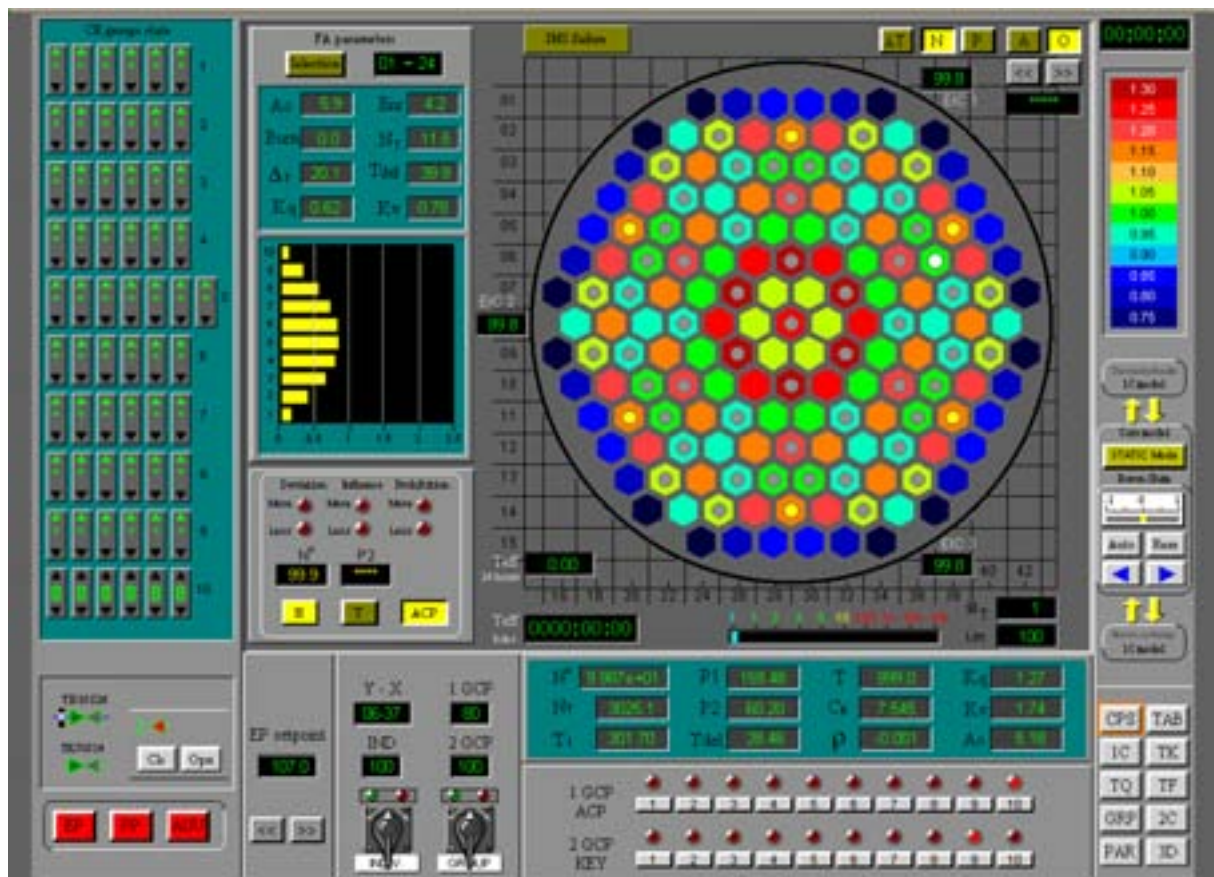
Every button in the tool bar has a tool tip.

3.3. Training tasks selection

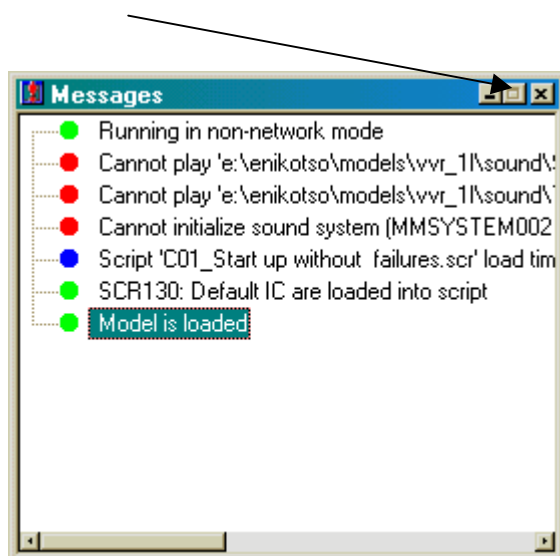
To select the training task for execution double-click by mouse on the item in the list of training tasks.



After that the Simulator display pages will be loaded, the display page will look like the following:



The system messages will be presented in the **Messages** window. The message window can be minimized.



Now the training task is selected, model is loaded, Simulator is ready for operation.

3.4. Simulator start, stop and exit

To get access to the Simulator control bar, move cursor to the top of the screen and the control bar will appear. Every button in the control bar has a tool tip.

To start the simulation, press the button **Start/Stop**.



To stop the simulation, depress the button **Start/Stop**.



To exit the training task, press the button **Exit** and confirm exiting.



3.5. Main simulator controls

3.5.1. Control rod operation

To operate a CR bank by key it is required to perform the following steps:

1. On the enunciator page (TAB) check that it is not prohibited to move a CR. Quit and deblock signals if they exist.
2. Select a CR bank on the CR banks selector panel by clicking on one of the buttons in the lower row. When the particular CR bank is selected the lamp above the button is highlighted in red. Control rods of the selected bank are highlighted by yellow circles on the assembly-wise diagram.
3. To insert a CR bank into the core, click at the lower right corner of the control rod bank operation key and keep the left mouse button pressed. When the CR bank is moving into the core, the lamp above the key is highlighted in green.
4. To withdraw a CR bank out of the core, click at the lower left corner of control rod bank operation key and keep the left mouse button pressed. When the CR bank is moving out of the core the lamp above the key is highlighted in red.
5. Monitor the position of the CR bank on the CR position panel and in the window above the key.

To operate an individual CR by key it is required to perform the following steps:

1. On the enunciator page (TAB) check that it is not prohibited to move a CR. Quit and deblock signals if they exist.
2. Select an individual CR on the assembly-wise diagram by clicking on it. When the particular CR is selected it is highlighted by a white circle on the assembly-wise diagram.
3. To insert the CR into the core, click at the lower right corner of individual control rod operation key and keep the left mouse button pressed. When CR is moving into the core the lamp above the key is highlighted in green.

4. To withdraw a CR out of the core, click at the lower left corner of the individual control rod operation key and keep the left mouse button pressed. When the CR is moving out of the core, the lamp above the key is highlighted in red.
5. Monitor the position of the CR on the CR position panel and in the window above the key.

3.5.2. Pump operation

To operate any pump it is required to do the following:

1. Select a pump on the Simulator display page by clicking on it. After the pump is selected the triangle on the pump sign will be highlighted in white.
2. Find the pump control group on the right of display page. The pump control group includes: pump sign, pump name, buttons **On/Off** (Open/Close).
3. Press button **On** to start the selected pump.
4. Press button **Off** to stop the selected pump.

When the pump is switched on the triangle on the pump sign is highlighted in red.

When the pump is switched off the triangle on the pump sign is highlighted in green.

3.5.3. Valve operation

To operate any motorized valve it is required to do the following:

1. Select the valve on Simulator display page by clicking on it. After the valve is selected it will be highlighted in white.
2. Find the valve control group on the right of display page. The valve control group includes: motorized valve sign, valve name, buttons **Opn/Cls** (Open/Close).
3. Press button **Opn** to open the selected valve.
4. Press button **Cls** to close the selected valve.

A motorized valve can not be put in an intermediate position.

When the valve is completely open, it is highlighted in red.

When the valve is opening, it is blinking in red.

When the valve is completely closed, it is highlighted in green.

When the valve is closing, it is blinking in green.

3.5.4. Automatic controller operation

Typically, under normal operation conditions, all automatically controlled valves are in AUTO mode. To operate any automatically controlled valve (ACV) manually it is required to do the following:

1. Select an automatically controlled valve on the Simulator display page by clicking on it. After the valve is selected it will be highlighted in white.
2. Find the ACV control group on the right of display page. The ACV control group includes: ACV sign, ACV name, buttons **Auto/Rem** (Automatic/Manual), buttons **<</>>**, bar position indicator.
3. Press button **Rem** to switch ACV to MANUAL.
4. Press button **>>** to open the selected ACV.
5. Press button **<<** to close the selected ACV.

The ACV can be put in an intermediate position (partially opened).

When the ACV is completely open, it is highlighted in red.

When the ACV is opening, it is blinking in red.

When the ACV is completely closed, it is highlighted in green.

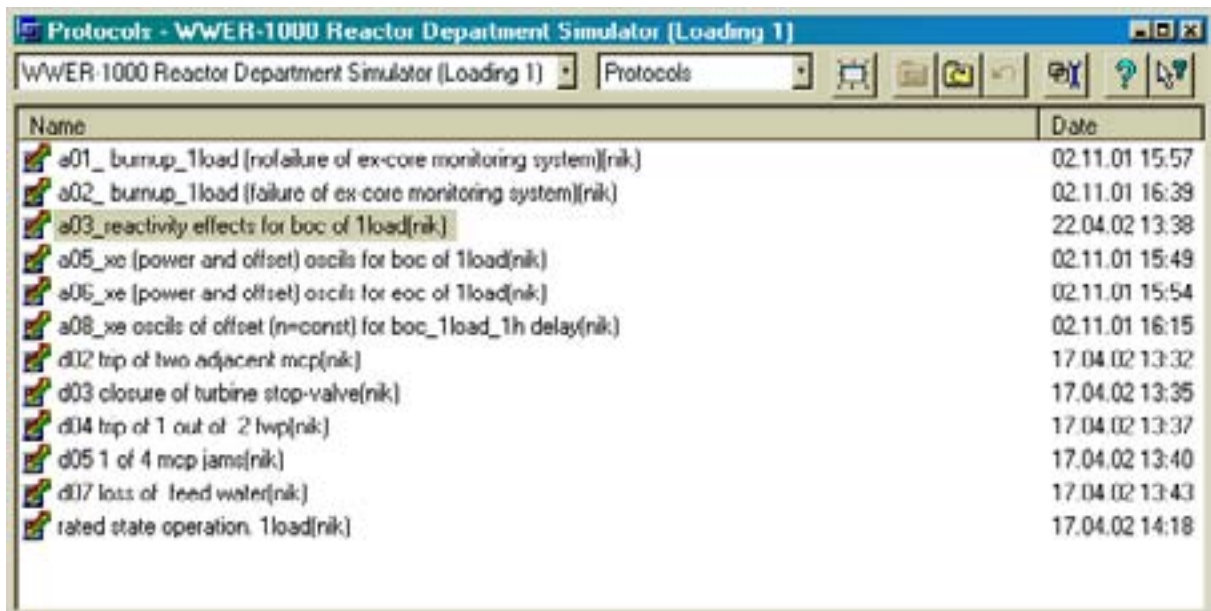
When the ACV is closing, it is blinking in green.

3.6. Protocol analysis

3.6.1. Protocol selection

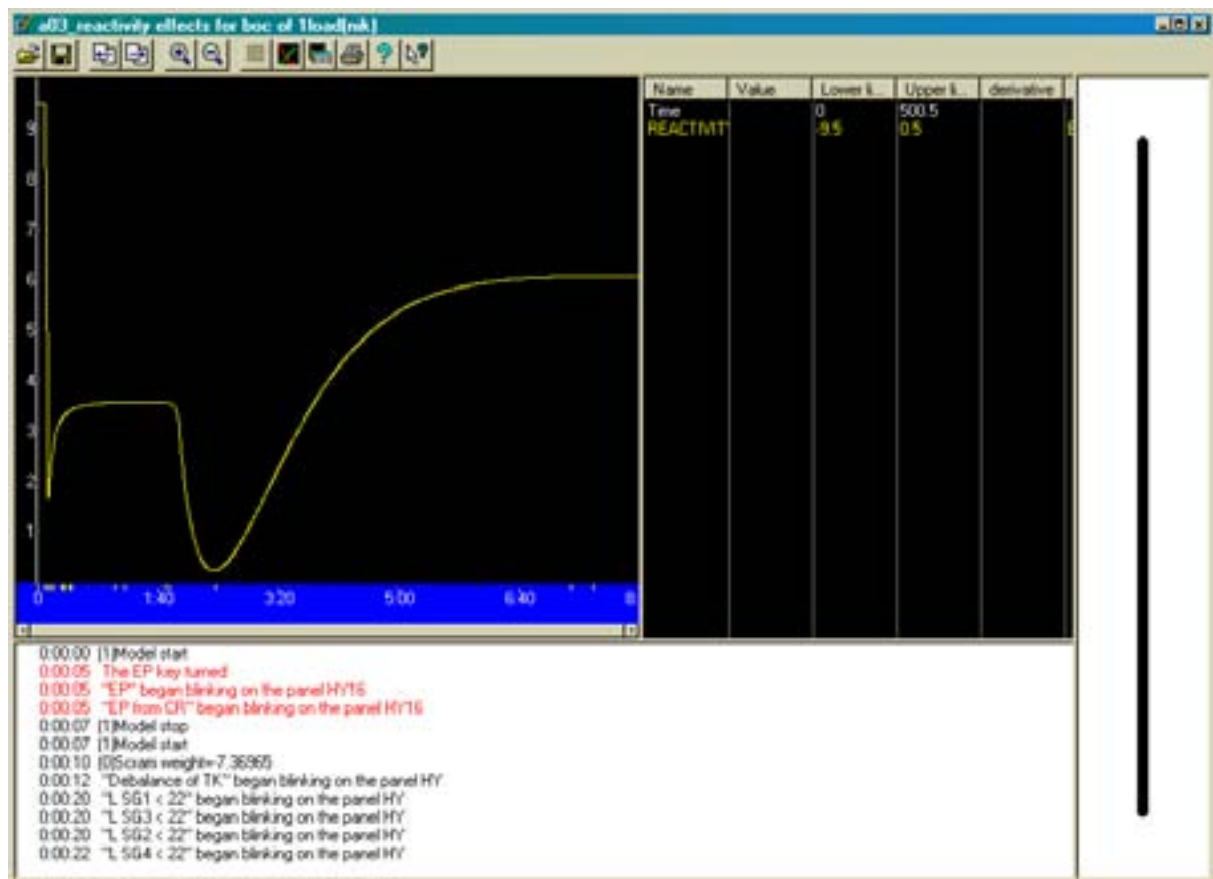
To view and analyze the simulation results, the Protocol Viewer is used. To go to the protocol window select the <Protocols> option in a drop down menu of File Type Selector.

The list of saved protocols will be displayed like this:



Name	Date
a01_burnup_1load (nofailure of ex-core monitoring system)(nik)	02.11.01 15:57
a02_burnup_1load (failure of ex-core monitoring system)(nik)	02.11.01 16:39
a03_reactivity effects for boc of 1load(nik)	22.04.02 13:38
a05_xe (power and offset) oscils for boc of 1load(nik)	02.11.01 15:49
a06_xe (power and offset) oscils for eoc of 1load(nik)	02.11.01 15:54
a08_xe oscils of offset (n=const) for boc_1load_1h delay(nik)	02.11.01 16:15
d02 tip of two adjacent mcp(nik)	17.04.02 13:32
d03 closure of turbine stop-valve(nik)	17.04.02 13:35
d04 tip of 1 out of 2 twp(nik)	17.04.02 13:37
d05 1 of 4 mcp (ame)(nik)	17.04.02 13:40
d07 loss of feed water(nik)	17.04.02 13:43
rated state operation. 1load(nik)	17.04.02 14:18

Double click on the protocol in the list to select it for analysis. The following window will appear:



3.6.2. Protocol viewer window functions

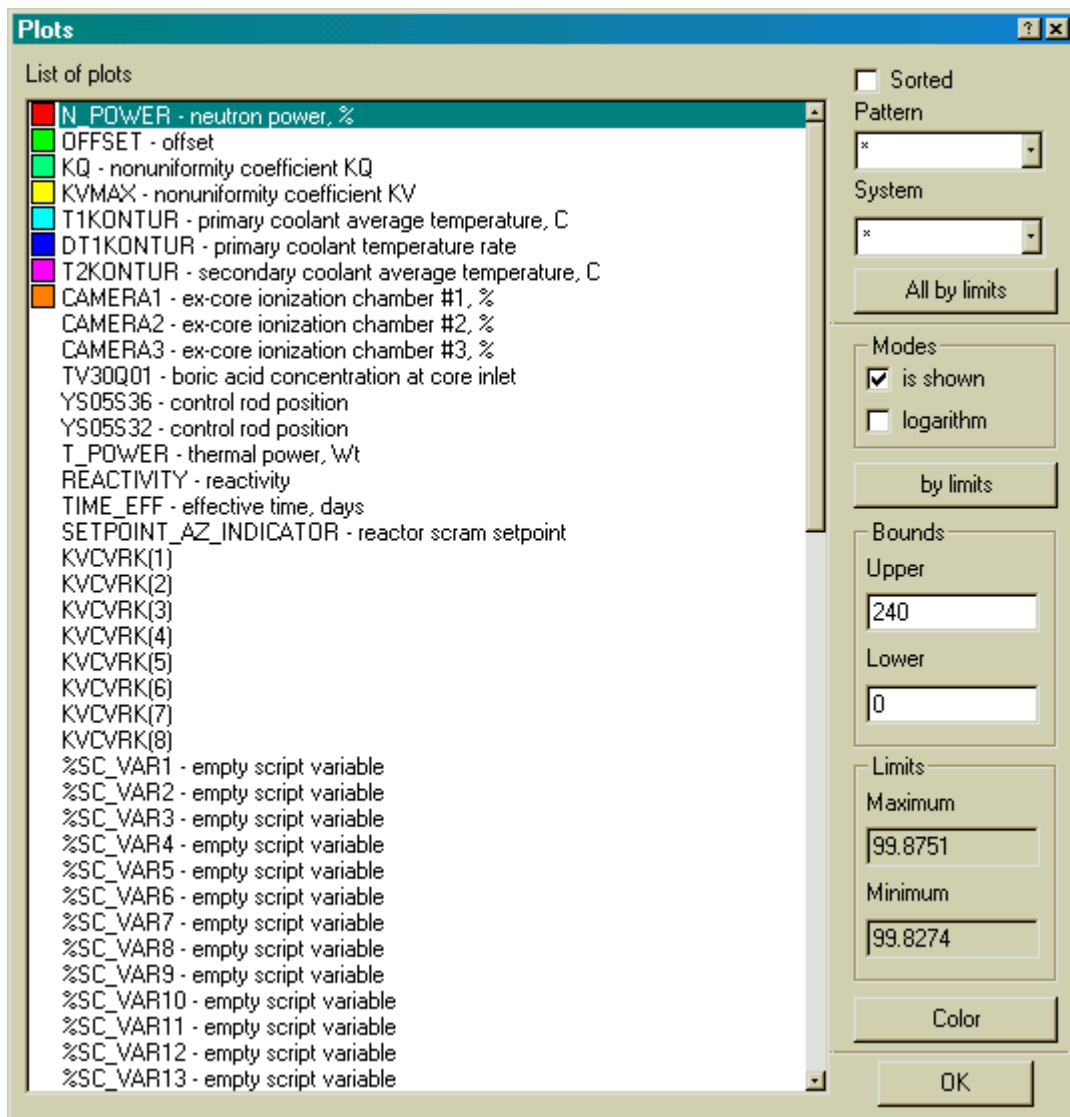
The protocol window includes the following components:

- Toolbar,
- Plot diagram,
- Displayed variables,
- Text window,
- Training trajectory.

The toolbar contains tool buttons for various functions. Every button in toolbar has its own tool tip. The main functions include the following:

- Open and save protocol,
- Zoom in and Zoom out,
- Select variables to be displayed on the plot,
- Save data to text file,
- Print plot.

The plot diagram shows the plots of variables, recorded during the training session. To edit plot parameters, press the <Plots> button in the Tool bar. The following window appears:



To select a variable from the list to be displayed on the plot - double click on the variable's name. (see Appendix A for the list of variables). To remove a variable from the plot – double click on the variable's name. It is also possible to change color for the plot lines and define lower and upper boundary values for the Y-axis

To the right of the plot there is the plot parameters description. It includes the variable name, current value, lower and upper limit on Y-axis, derivative (if it exists) and description.

The text window contains list of events that occurred during the training session. The list includes the following type of events:

- Simulator model start and stop,
- Enunciator signals occurrence,
- Changes of equipment status, e.g. pumps start and stop, opening and closing valves, etc,
- Model acceleration,
- Parameter values, defined in the script.

The training trajectory line is displayed on the right side of the window. When the sequence of simulated events is linear, the trajectory is presented in a solid black line. If, during training session, a simulator state has been saved and then the model has been returned to the saved state, the trajectory will have point and branches.

3.7. Simulator display pages

Ten display pages are available in the Simulator. They are

1. Reactivity control page (CPS),
2. Enunciator page (TAB),
3. Primary circulation loop page (1C),
4. Feed and bleed system page (TK),
5. Process support systems page (TQ),
6. Extraction water cooling system page (TF),
7. Secondary circulation loop page (2C),
8. Trends page (GRP),
9. Reactor core parameters page (PAR),
10. Three dimension diagram page (3D).

Display page selector buttons are used to switch between simulator display pages. Display page selector buttons are available on each display page in the right lower corner of the screen.

On all simulator display pages, instrumentation readings are presented in the following units:

Flow rates	- t/h,
Pressure	- kg/cm ² ,
Temperature	- °C,
Concentration	- g/kg,
Position (valve)	-
Reactivity	- %ΔK/K.

3.7.1. Reactivity control page (CPS)

This is the main screen of the Simulator. The screen shows

- Reactivity control equipment presented in the simulator,
- Reactor core parameters and individual fuel assembly parameters,
- Assembly-wise distribution of reactor core parameters.

Reactivity control equipment presented in Simulator includes:

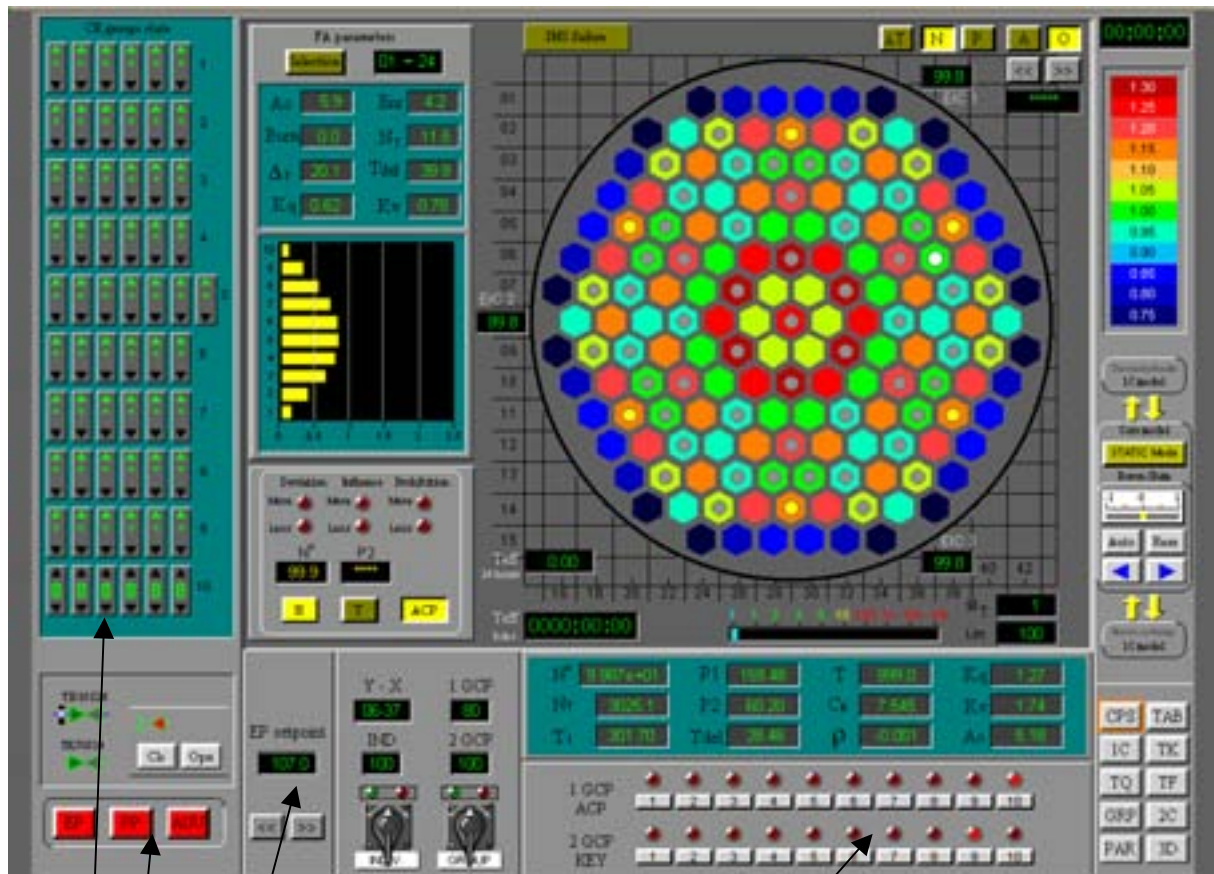
- Control rod bank operation key,
- Individual control rod operation key,
- Reactor protection system keys,
- Emergency protection set point selector,
- Two groups of CR banks selectors.

To operate a CR bank by key it is required to perform the following steps:

1. On the enunciator page (TAB) check that it is not prohibited to move a CR. Quit and deblock signals if they exist.
2. Select a CR bank on CR banks selector panel by clicking on one of the buttons in the lower row. When the particular CR bank is selected the lamp above the button is highlighted in red. Control rods of the selected bank are highlighted by yellow circles on the assembly-wise diagram.
3. To insert a CR bank into the core click at the lower right corner of control rod bank operation key and keep the left mouse button pressed. When the CR bank is moving into the core, the lamp above the key is highlighted in green.
4. To withdraw a CR bank out of the core, click at the lower left corner of control rod bank operation key and keep the left mouse button pressed. When the CR bank is moving out of the core, the lamp above the key is highlighted in red.
5. Monitor the position of CR bank on the CR position panel and in the window above the key.

To operate an individual CR by key it is required to perform the following steps:

1. On the enunciator page (TAB) check that it is not prohibited to move a CR. Quit and deblock signals if they exist.
2. Select an individual CR on the assembly-wise diagram by clicking on it. When the particular CR is selected it is highlighted by a white circle on the assembly-wise diagram.
3. To insert a CR into the core, click at the lower right corner of the individual control rod operation key and keep the left mouse button pressed. When the CR is moving into the core, the lamp above the key is highlighted in green.
4. To withdraw a CR out of the core, click at the lower left corner of the individual control rod operation key and keep the left mouse button pressed. When the CR is moving out of the core, the lamp above the key is highlighted in red.
5. Monitor the position of CR on the CR position panel and in the window above the key.



1. Control rod bank operation key.
2. Individual control rod operation key.
3. Reactor protection system keys.
4. Emergency protection set point selector.
5. Two groups of CR banks selectors.
6. Control rod position panel.

Overall reactor parameters are displayed under the assembly wise diagram. The following integral parameters are shown:

N	neutron power (%)
Nt	reactor thermal power (Mw)
T1	primary circuit average coolant temperature (°C)
P1	pressure in reactor vessel (kg/cm ²)
P2	secondary circuit pressure (kg/cm ²)
Burn	fuel burnup value (Mw*day)
Tdel	temperature margin to boiling (°C)
T	reactor period (sec)
Cb	boron concentration (g/kg)
ρ	reactivity (%ΔK/K)
Kq	non-uniformity coefficient for FAs
Kv	sparse non-uniformity coefficient
Ao	axial offset (%)

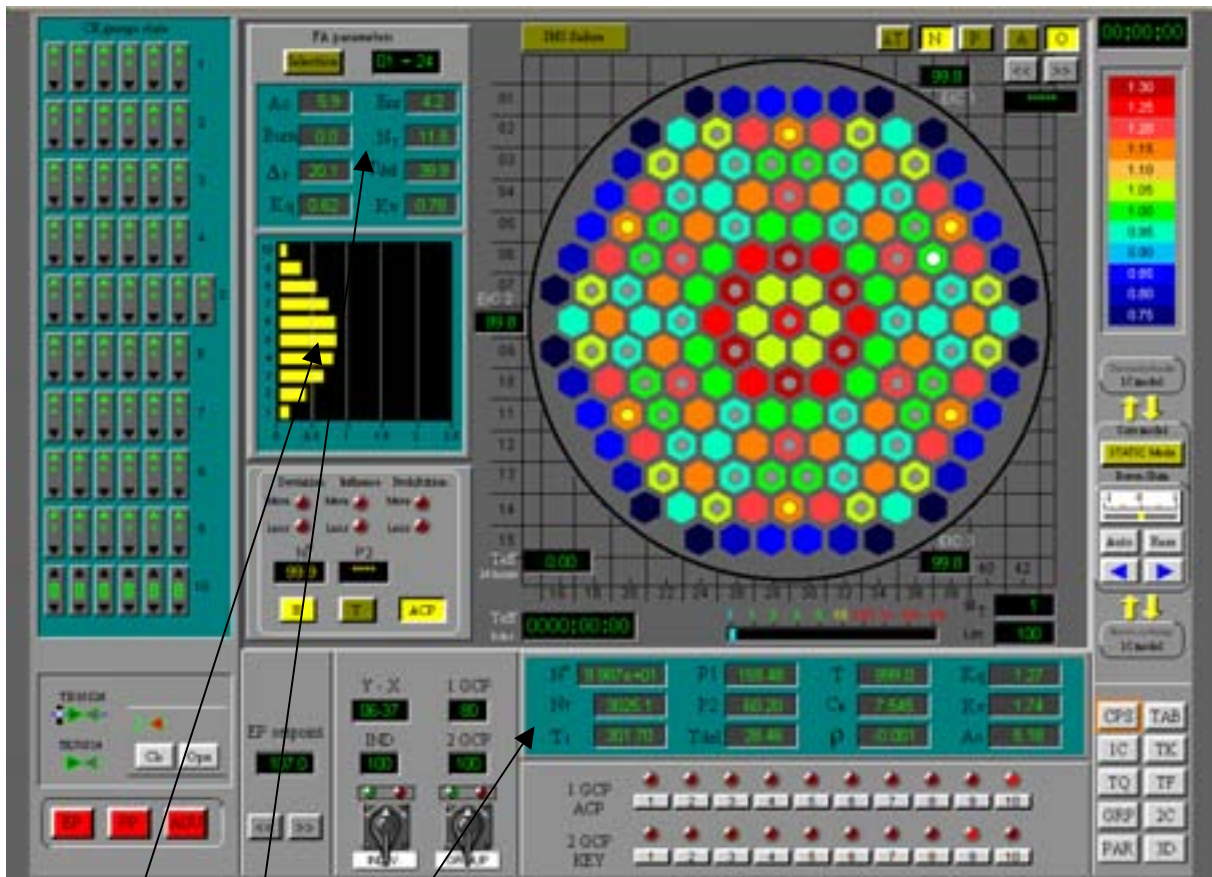
The axial distribution of relative power release in a fuel assembly as well as individual fuel assembly parameters are displayed to the left of the assembly-wise diagram. The following FA parameters are shown:

Ao	axial offset in FA(%)
Burn	fuel burnup value (Mw*day)
Δt	outlet – inlet temperature difference (°C)
Kq	non-uniformity coefficient for FA
Enr	fuel enrichment (%U ₂₃₅)
Nt	FA thermal power (Mw)
Tdel	temperature margin to boiling (°C)
Kv	sparse non-uniformity coefficient

To select a FA for review it is required to perform the following steps:

1. Press button <Selection>. When pressed it will be highlighted in yellow.
2. Click on the FA to be reviewed. The FA coordinates and parameters will be displayed.

To review another FA repeat items 1 and 2.



1. Overall reactor parameters.
2. Individual fuel assembly parameters.
3. Axial distribution of relative power release in fuel assembly.

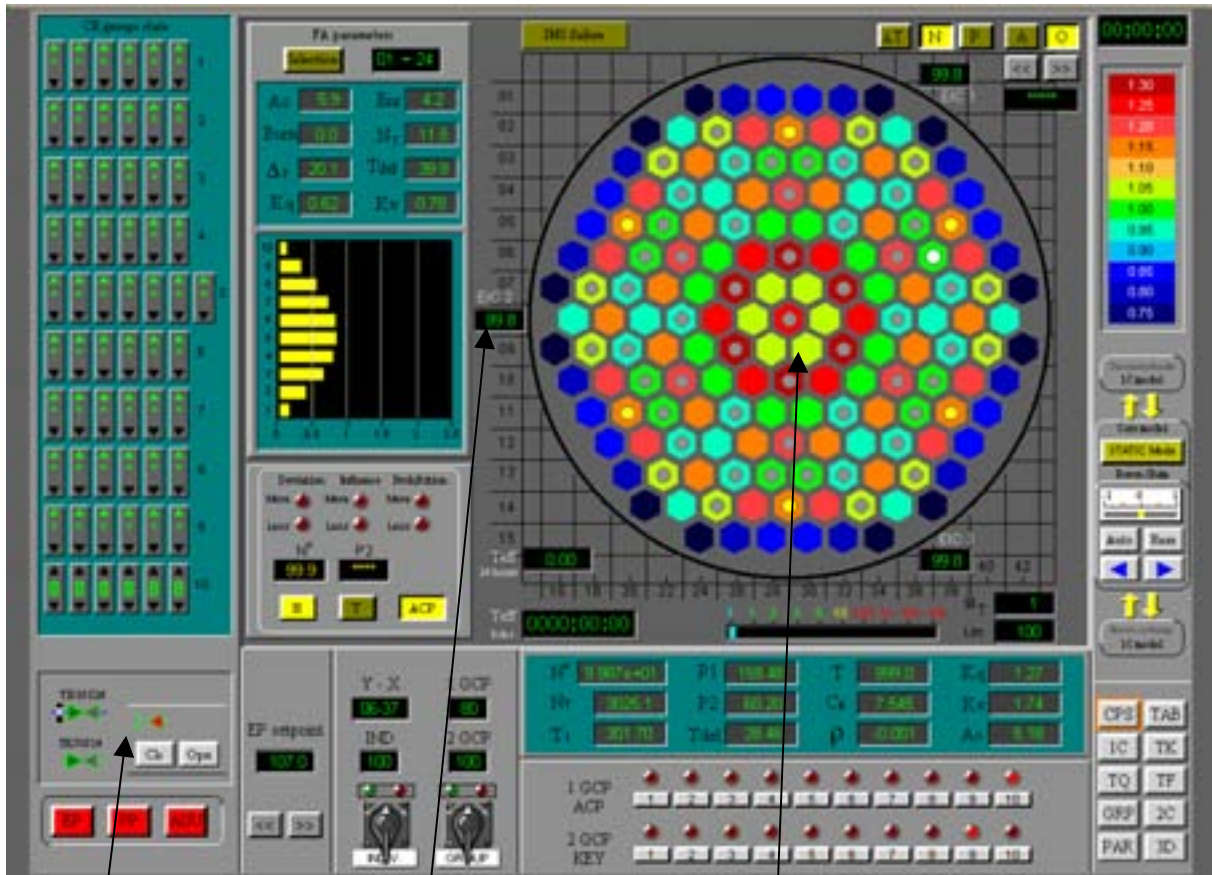
The assembly-wise distribution for three simulated reactor core parameters is presented in the right upper part of the screen. By pressing the buttons at the top of the screen the following parameters can be displayed on the diagram:

- N - fuel assembly neutron power,
- P - fuel assembly thermal power,
- dT - outlet – inlet coolant temperature difference in fuel assemblies.

Around the assembly-wise diagram three ex-core ionizing chambers (EIC) are displayed. On the right side of the screen there is a color scale. It changes according to the parameter selected for display on the diagram.

The pressed button is highlighted in yellow.

Valves presented in the left lower part of the screen can be operated by means of control panels located near the valves. Opening the upper valve results in boron injection into the primary circuit. Opening the lower valve results in boron dilution with pure water. These are the same valves as presented on the TK page.



1. Assembly-wise distribution of reactor core parameters.
2. Ex-core ionizing chambers.
3. Boron regulating valves with control panel.

Along the right side of the screen there are

- Simulator clock (at the top),
- Color scale. It changes according to the parameter selected for display on the diagram,
- Boron shim control panel,
- Model connectors for primary circuit model and for boron exchange model,
- Static neutronic model button,
- Display page selector buttons (at the bottom).

The simulator clock shows the actual simulation time in hh:mm:ss.

The color scale is used for displaying selected parameter values on the assembly-wise diagram.

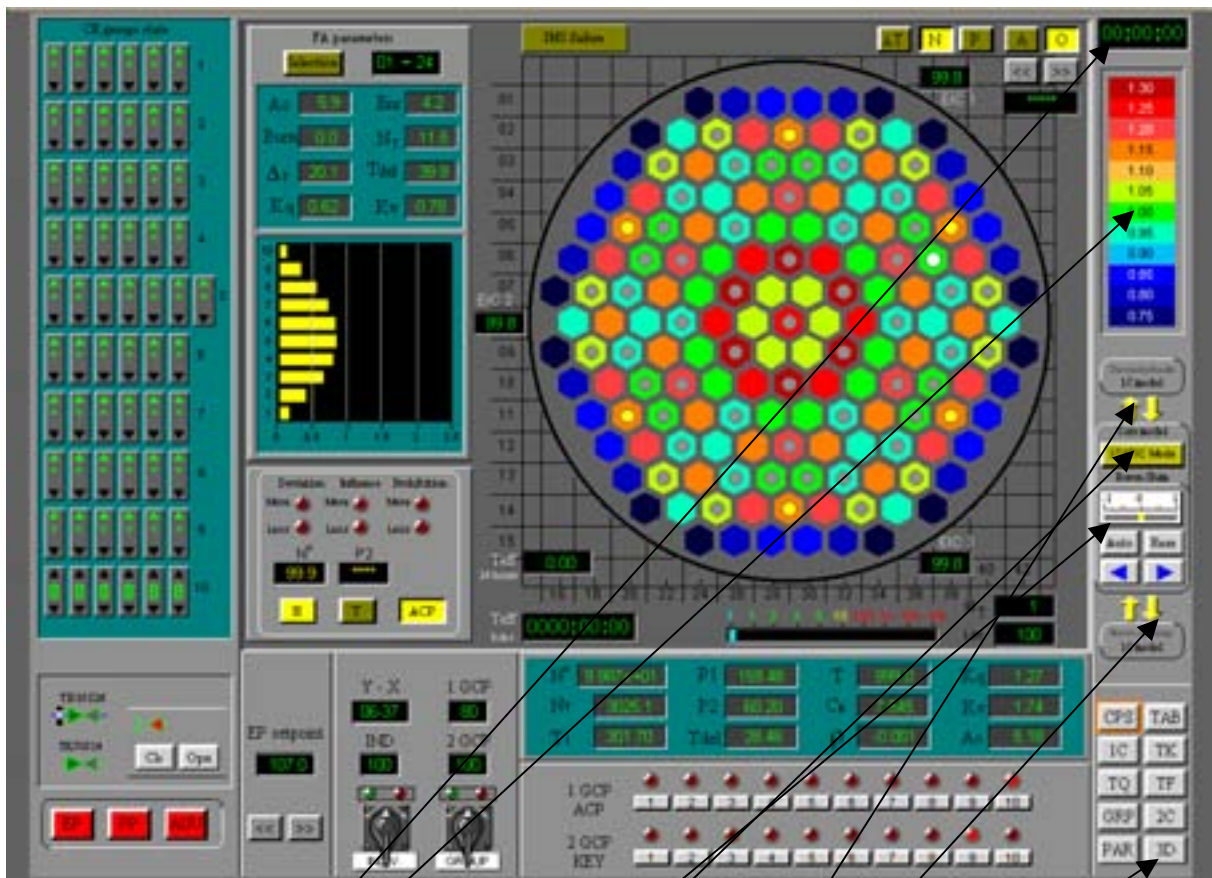
The boron shim control model works when boron exchange between the core model and the primary circuit model is disconnected. To disconnect boron exchange click on the two arrows highlighted in yellow. When the boron exchange model is disconnected, the boron shim

control panel is highlighted by a blue frame. It is possible to set the boron shim controller into AUTO or REM (manual) mode.

Model connectors allow for disconnection of the reactor core model from the primary circuit model. To disconnect, click on the two arrows highlighted in yellow. When the reactor core and primary circuit models are disconnected, the core inlet and outlet parameters are fixed and there is no data exchange between the two models. This is used for simulation of fuel burn up and other processes when the reactor core model has to be accelerated significantly.

Models are disconnected automatically when the acceleration multiplier exceeds 10. If the multiplier is returned to 1, the two models are reconnected.

Display page selector buttons are used to switch between simulator display pages. They are available on each display page.



1. Simulator clock.
2. Color scale.
3. Boron shim control panel.
4. Model connectors for primary circuit model and for boron exchange model.
5. Static neutronic model button.
6. Display page selector buttons.

3.7.2. *Enunciators page (TAB)*

The screen shows the simulator enunciator panel. All enunciator signals are presented in the following six groups:

- EP - emergency protection,
- PP-1 - preventive protection 1 (PZ-1),
- PP-2 - preventive protection 2 (PZ-2),
- AUU - accelerated unit unloading (URB),
- SG - steam generators,
- MSC - main steam collector,
- 1C - primary circuit.

While the simulator is in operation, if any of the signals are generated, the box on the panel will be highlighted in blinking yellow. For example, if the pressure in the reactor vessel is greater than 165 atm, the box **P1 > 165** on the panel **PP-2** will be highlighted in blinking yellow.

If a signal is activated, the correct response should be the following:

1. Remove the blinking by pressing <Blinking rem> button. The signal box will stop blinking and will be highlighted in solid yellow.
2. Deblock the signal by pressing <Deblocking> button. If the reason for the signal does not exist any more, the signal will be deblocked and the signal box will become dark. If the reason for the signal still exists, the signal will not be deblocked. In this case it is necessary first to eliminate the reason and then to deblock the signal.

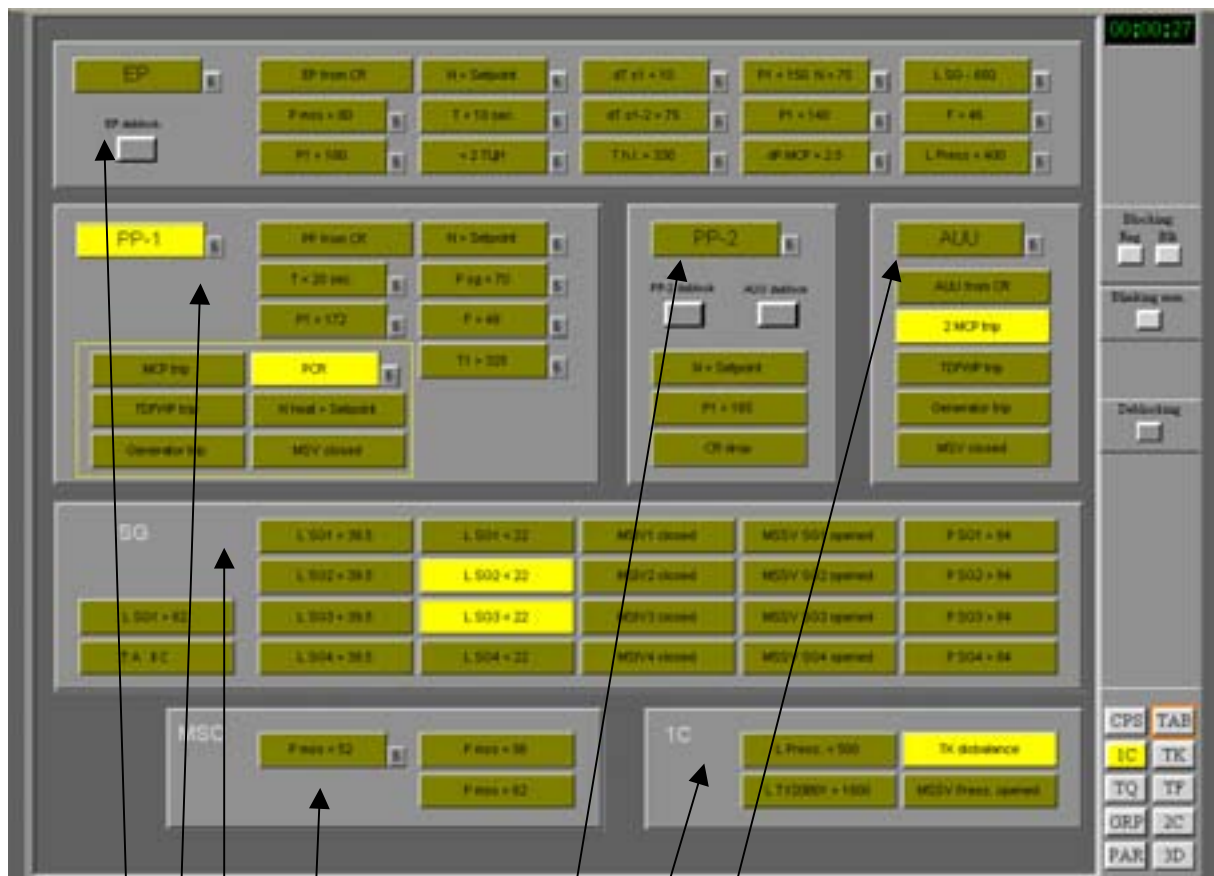
If emergency protection was activated, press the <EP deblock> button to enable further reactor parameters control.

If preventive protection was activated, press <PP deblock> button to enable further reactor parameters control.

If accelerated unit unloading was activated, press <AUU deblock> button to enable further reactor parameters control.

The screen allows to block all signals in the groups EP, PP-1, PP-2, AUU as well as to block individual signals from any group by pressing button <Б> located to the right of the signal box. Blocking any signal or group of signals will result in elimination of protection system actuation due to the signal reason.

A complete list of cues for enunciator signals is presented in Appendix B.



1. Emergency protection group of signals (EP).
2. Preventive protection 1 group of signals (PP-1).
3. Preventive protection 2 group of signals (PP-2).
4. Accelerated unit unloading group of signals (AUU).
5. Steam generators group of signals (SG).
- 6 Main steam collector group of signals (MSC).
7. Primary circuit group of signals (1C).

3.7.3. Primary circulation circuit page (1C)

The screen shows the main equipment of the primary circuit presented in the simulator. It includes:

- reactor vessel,
- four primary circulation loops with hot leg marked in red and cold leg marked in blue,
- four main circulation pumps,
- pressurizer with primary circuit pressure automatic control and four groups of electrical heaters,
- pressurizer spray trains,
- pressurizer safety relief valves,
- bubbler-condenser for emergency steam dumping from reactor vessel, pressurizer and SGs,
- high pressure part of emergency core cooling system, including four trains and four high pressure water storage tanks,
- four steam generators (horizontal boilers),
- SG level control motorized valves,
- four main steam release valves (MSRV).

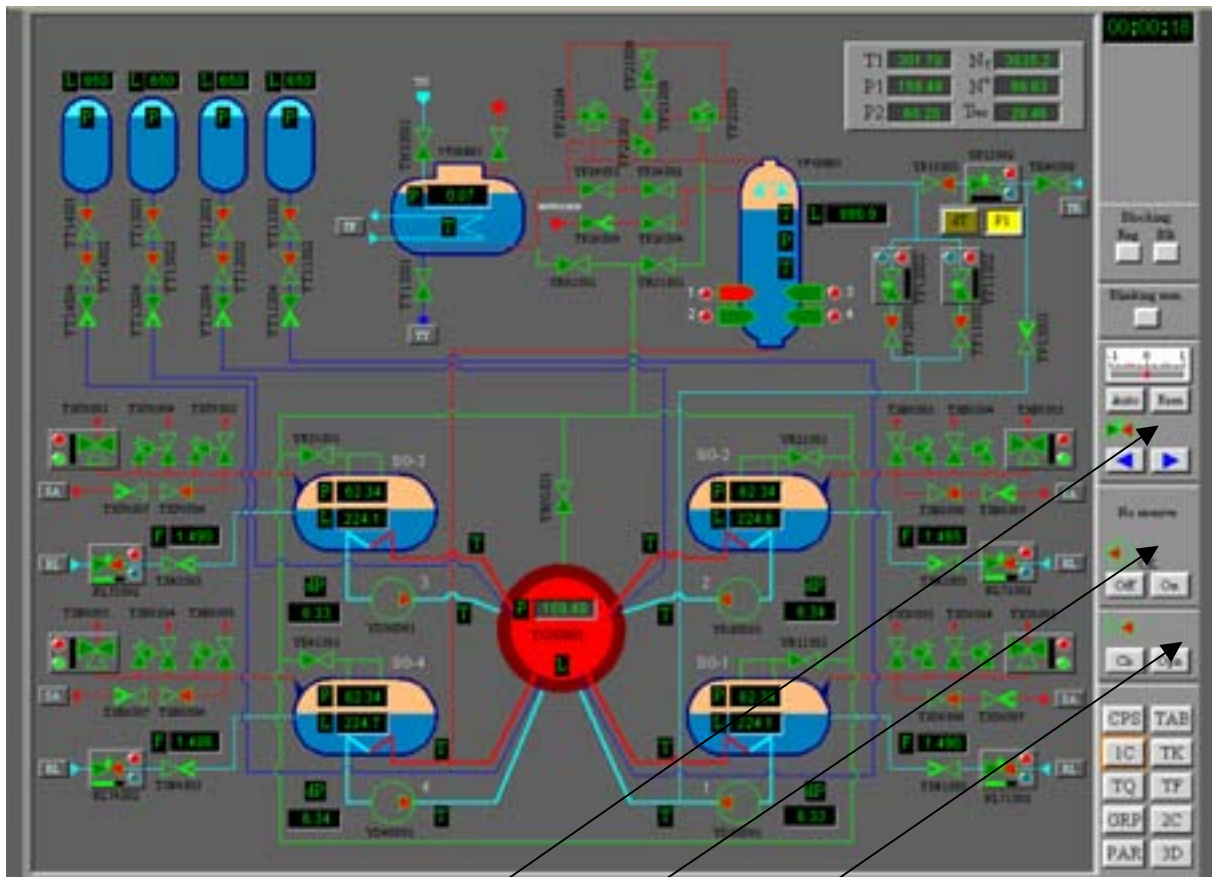
Under normal operating conditions, all automatic controls are in Auto mode. The screen allows to switch automatic controllers to manual mode and to control manually the following parameters:

- pressure in pressurizer,
- levels in every SG.

Also, all valves and pumps presented on the screen can be operated by means of control panels, located on the right side of the screen.

Along the right side of the screen there are:

- display page selector buttons (at the bottom),
- simulator clock (at the top),
- control panel for valves, pumps and automatic controllers operation (in the middle).



1. Control panel for automatic controllers operation.
2. Control panel for pumps operation.
3. Control panel for valves operation.

3.7.4. Feed and bleed system page (TK)

The screen shows the equipment of three different systems presented in the simulator. They are:

- Primary circuit feed system,
- Primary circuit bleed system,
- Boron concentrate feed and dilution system.

The yellow wall on the screen is the containment wall. Equipment below the wall is installed outside the containment.

The primary circuit feed system is presented in the right upper part of the screen. The system includes two trains with two pumps in each train, two automatic controllers, a heat exchanger for feed heating and four feed lines connected to cold legs of each of the four primary loops.

The primary circuit bleed system is presented in the left part of the screen. The system includes the bleed line from the cold leg of loop #2, two heat exchangers for bleed cooling, two automatic controllers and a bleed deaerator. From the bleed deaerator, the coolant goes to the feed system and back to the primary circuit.

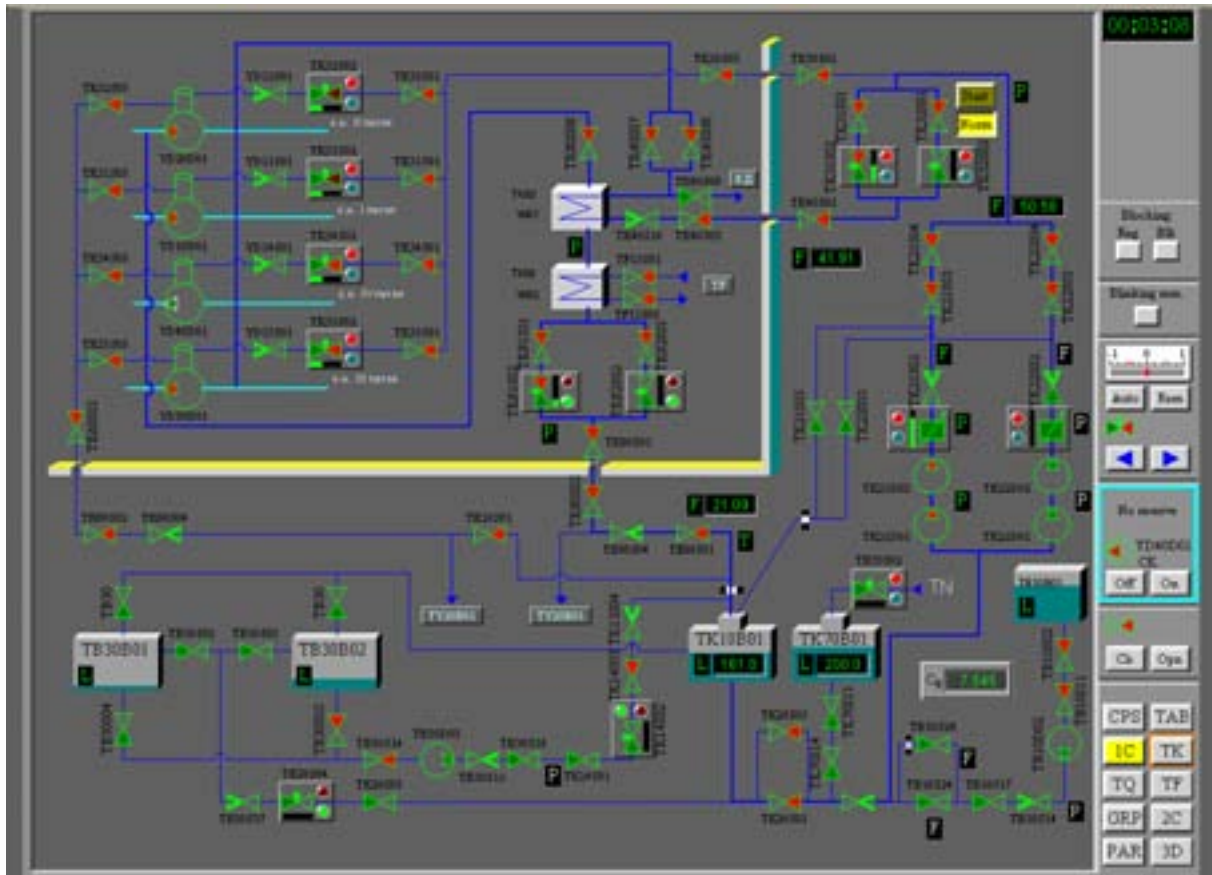
The primary circuit coolant can be drained from the bleed deaerator to two reserved water tanks, presented in the left lower corner of the screen. Coolant can be added to the bleed deaerator from the same two tanks.

The boron concentrate feed and dilute system is presented in the right lower corner of the screen. It includes a boron concentrate tank, boron concentrate feed line with pump, and a clear water tank for boron dilution. During boron injection, boron concentrate is taken from the boron concentrate tank and pumped to the feed lines. During boron dilution clear water (distillate) goes to the feed lines and then to the primary circuit. Dilution and injection can be done manually.

All valves and pumps presented on the screen can be operated by means of control panels, located on the right side of the screen.

Along the right side of the screen there are:

- display page selector buttons (at the bottom),
- simulator clock (at the top),
- control panel for valves, pumps and automatic controllers operation (in the middle).



3.7.5. Process support systems page (TQ)

The screen shows the equipment of three different systems presented in the simulator. The three systems are:

- Primary circuit cooling system,
- Boron concentrate feed system,
- Auxiliary steam generators level control system.

The yellow wall on the screen is the containment wall. Equipment below the wall is installed outside the containment.

The primary circuit cooling system is presented in the left part of the screen. The system includes a heat exchanger, pump and two automatic controllers. It has a connection line to the boron concentrate feed system.

The boron concentrate feed system is presented in the left part of the screen. It includes 2 tanks with boron concentrate. The low pressure tank is installed inside the containment, and the high pressure tank is outside the containment.

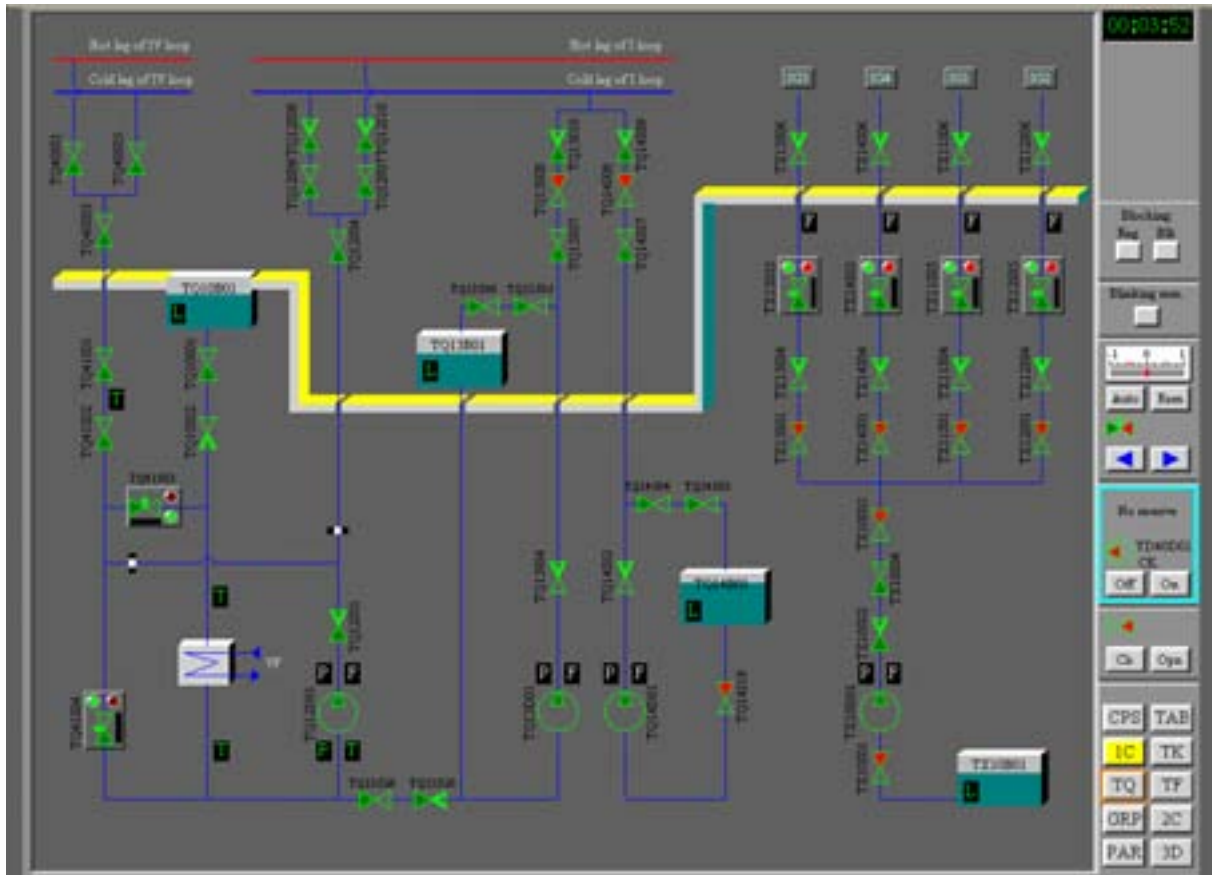
The auxiliary steam generators level control system is presented in the right part of the screen. The system includes the following components: water storage tank, one train with pump, four individual trains to four steam generators, and four automatic controllers.

The screen allows to switch automatic controllers to manual mode and to control levels in any SG manually.

Also all valves and pumps presented on the screen can be operated by means of control panels, located on the right side of the screen.

Along the right side of the screen there are:

- Display page selector buttons (at the bottom);
- Simulator clock (at the top);
- Control panel for valves, pumps and automatic controllers operation (in the middle).



3.7.6. Extraction water cooling system page (TF)

The screen shows the equipment of the extraction water cooling system presented in the simulator. It includes:

- Primary circuit drainage cooling system,
- Bubbler-condenser cooling heat exchanger,
- MCP drains cooling system.

The yellow wall on the screen is the containment wall. Equipment below the wall is installed outside the containment.

The primary circuit drainage cooling system is presented in the right part of the screen. The system includes drainage lines from primary circuit loops, a heat exchanger for drainage

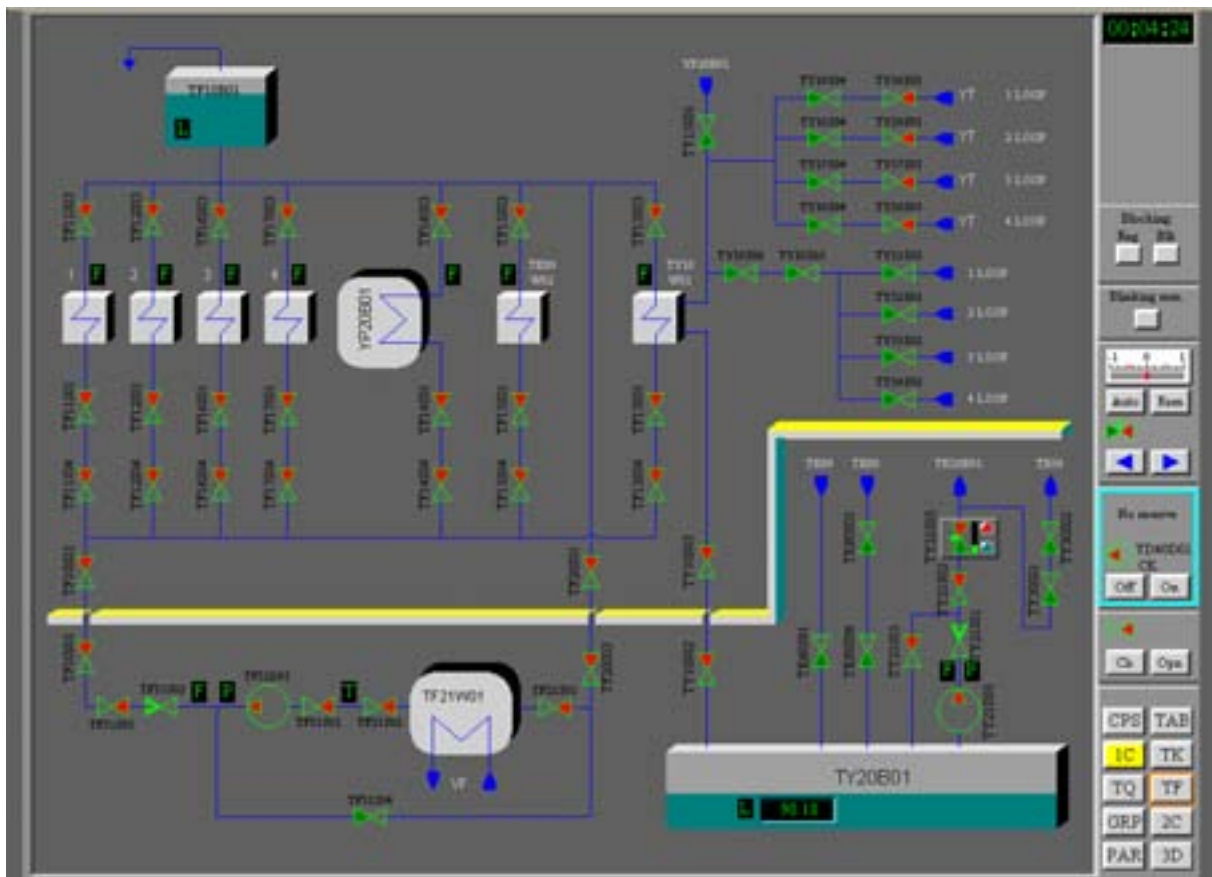
cooling, a drain water storage tank, a train to supply drain water to the bleed deaerator (page TF) with a pump and automatic controller.

The bubbler-condenser cooling heat exchanger and MCP drains cooling system is presented in the left part of the screen.

All valves and pumps presented on the screen can be operated by means of control panels, located on the right side of the screen.

Along the right side of the screen there are:

- Display page selector buttons (at the bottom),
- Simulator clock (at the top),
- Control panel for valves, pumps and automatic controllers operation (in the middle).



3.7.7. Secondary circuit page (2C)

The screen shows the main equipment of the secondary circuit presented in the simulator. It includes:

- four steam generators (horizontal boilers),
- SG level control motorized valves,
- main steam header,
- four main steam valves,

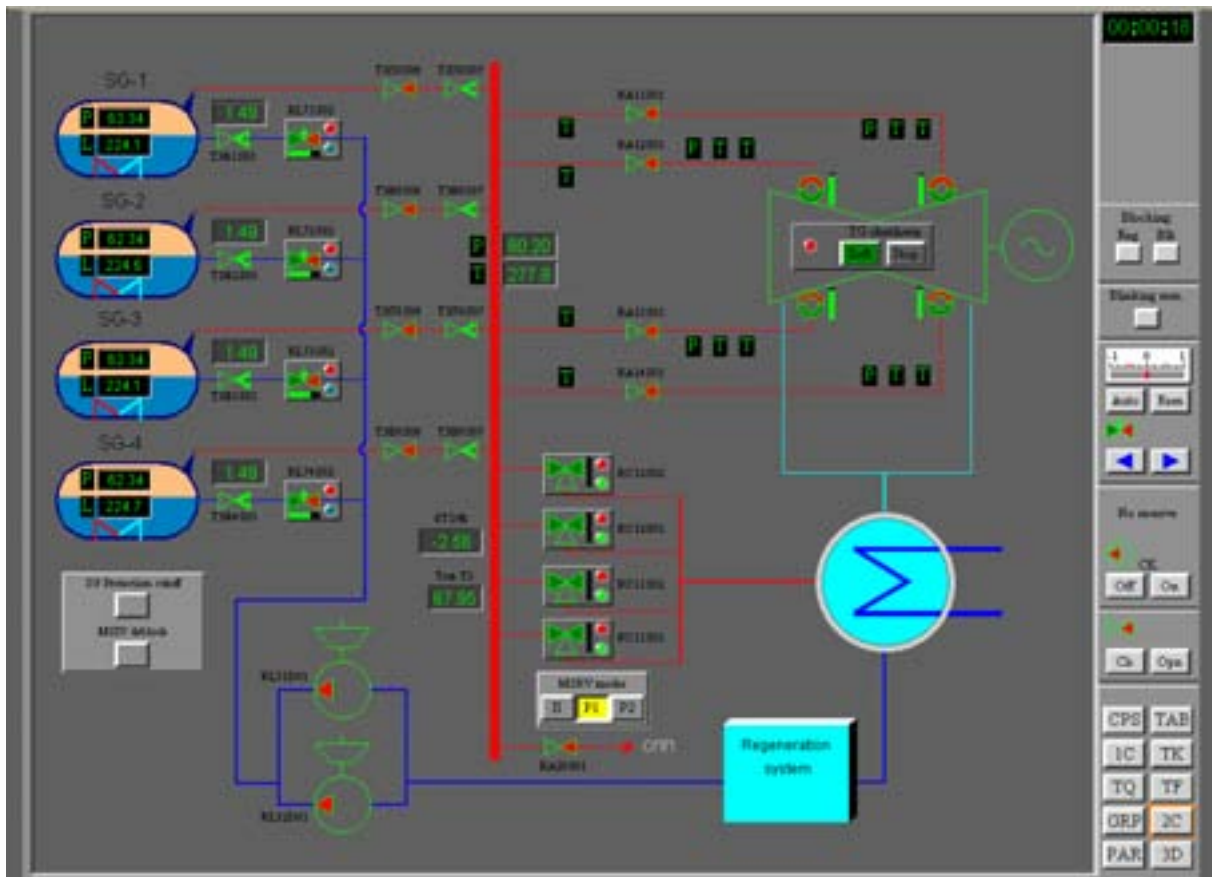
- turbine and generator,
- main condensate system,
- two SG feed pumps,
- four main steam release valves (MSRV),
- steam release valves mode selector.

Under normal operating conditions, all automatic controls are in Auto mode. The screen allows to switch automatic controllers to manual mode and to control levels in all four steam generators manually.

Also, all valves and pumps presented on the screen can be operated by means of control panels, located on the right side of the screen.

Along the right side of the screen there are:

- display page selector buttons (at the bottom),
- simulator clock (at the top),
- control panel for valves, pumps and automatic controllers operation (in the middle).



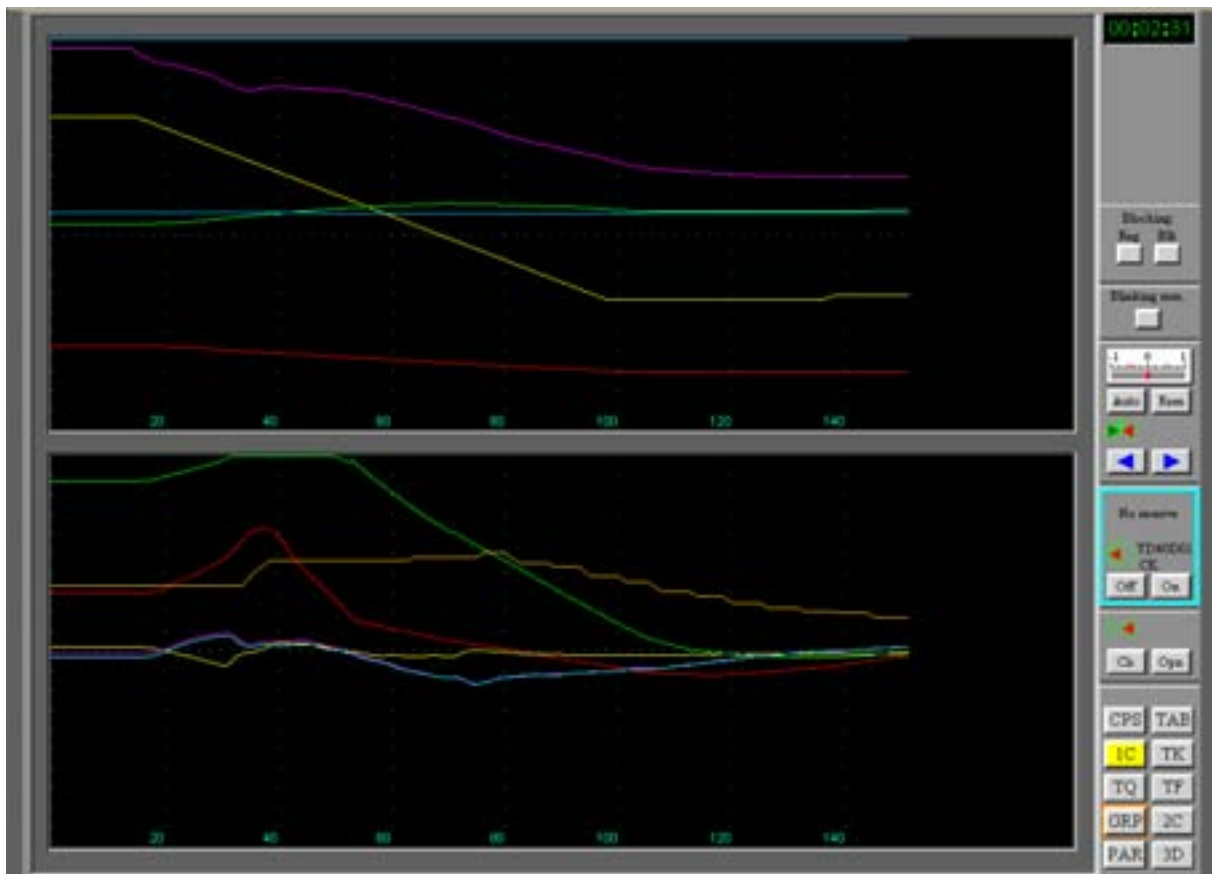
3.7.8. Trends page (GRP)

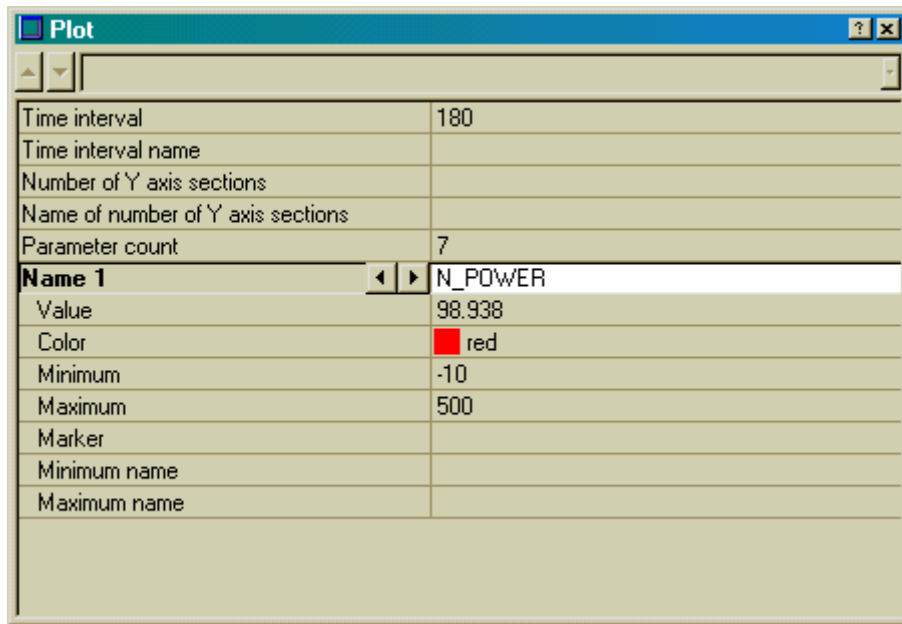
This screen shows two trend plots for simulated parameters. The horizontal axis represents simulation time, the vertical axis represents the selected parameter. Up to seven parameters can be selected for each of two plots.

To select the parameter for the plot click the right mouse button on the plot, select the item 'Edit Plot' and type the name of the variable in the field <Name> of the plot menu (for a detailed list of simulator variables see Appendix A: List of main variables for WWER-1000 Reactor Department Simulator). Then select the color for the line, and the minimum and maximum for the vertical axis. The current parameter value will be displayed in the field <Value>.

Note that the vertical axis on each plot can be adjusted to correspond to maximum and minimum values of the parameter during the time segment indicated by horizontal axis.

It is also possible to save plot data to an ASCII text file.





3.7.9. Reactor core parameters page (PAR)

This screen shows the assembly-wise distribution for eight simulated reactor core parameters. By pressing the buttons at the bottom of the screen the following parameters can be displayed on the diagram:

- Nt - fuel assembly power release (MW),
- KQ - fuel assembly relative power,
- dT - outlet – inlet coolant temperature difference in fuel assemblies (°C),
- Tout - outlet coolant temperature in fuel assemblies (°C),
- Tзап - margin to boiling in fuel assemblies (°C),
- ENR - fuel enrichment (%U₂₃₅),
- BUR - fuel burnup (Mw*day),
- OFF - offset value in fuel assemblies (%).

The pressed button is highlighted in yellow.

The maximum and minimum values for the selected parameter are highlighted in pink and green correspondingly. The FA coordinates are displayed in the lower right corner of the screen.

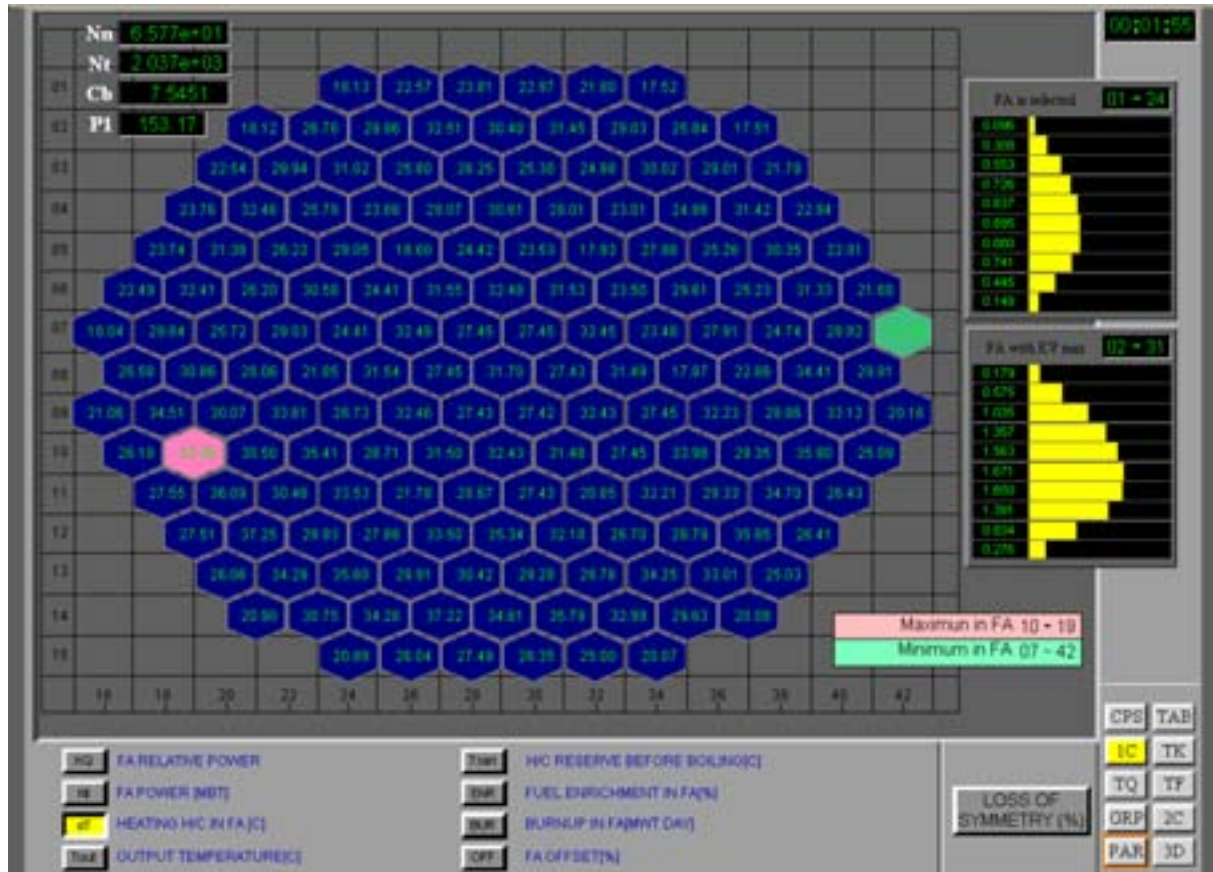
In the upper left corner of the screen the following four parameters are presented:

- Reactor neutron power (%),
- Reactor thermal power (Mw),
- Boron concentration in primary circuit (g/kg),
- Primary circuit pressure (Pa).

The axial distribution of relative power release is presented at the right of the screen. The upper diagram presents the distribution for the selected FA (highlighted by yellow border), the lower diagram presents axial data for the FA with maximum KQ value.

The FA can be selected for review by clicking the left mouse button.

LOSS OF SYMMETRY button should be used when there is a significant deterioration of core parameters from symmetrical distribution, e.g. one or two MCP trips, etc. When pressed the loss of symmetry in relative units (%) is displayed.



3.7.10. Three dimensional diagram page (3D)

This screen shows the three dimensional diagram for eight simulated reactor core parameters. By pressing the buttons at the bottom of the screen the following parameters can be displayed on the 3D diagram:

- Nt - fuel assembly power release (Mw),
- KQ - fuel assembly relative power,
- dT - outlet – inlet coolant temperature difference in fuel assemblies (°C),
- Tout - outlet coolant temperature in fuel assemblies (°C),
- Tзап - margin to boiling in fuel assemblies (°C),
- ENR - fuel enrichment (%U₂₃₅),
- BUR - fuel burnup (Mw*day),
- OFF - offset value in fuel assemblies (%).

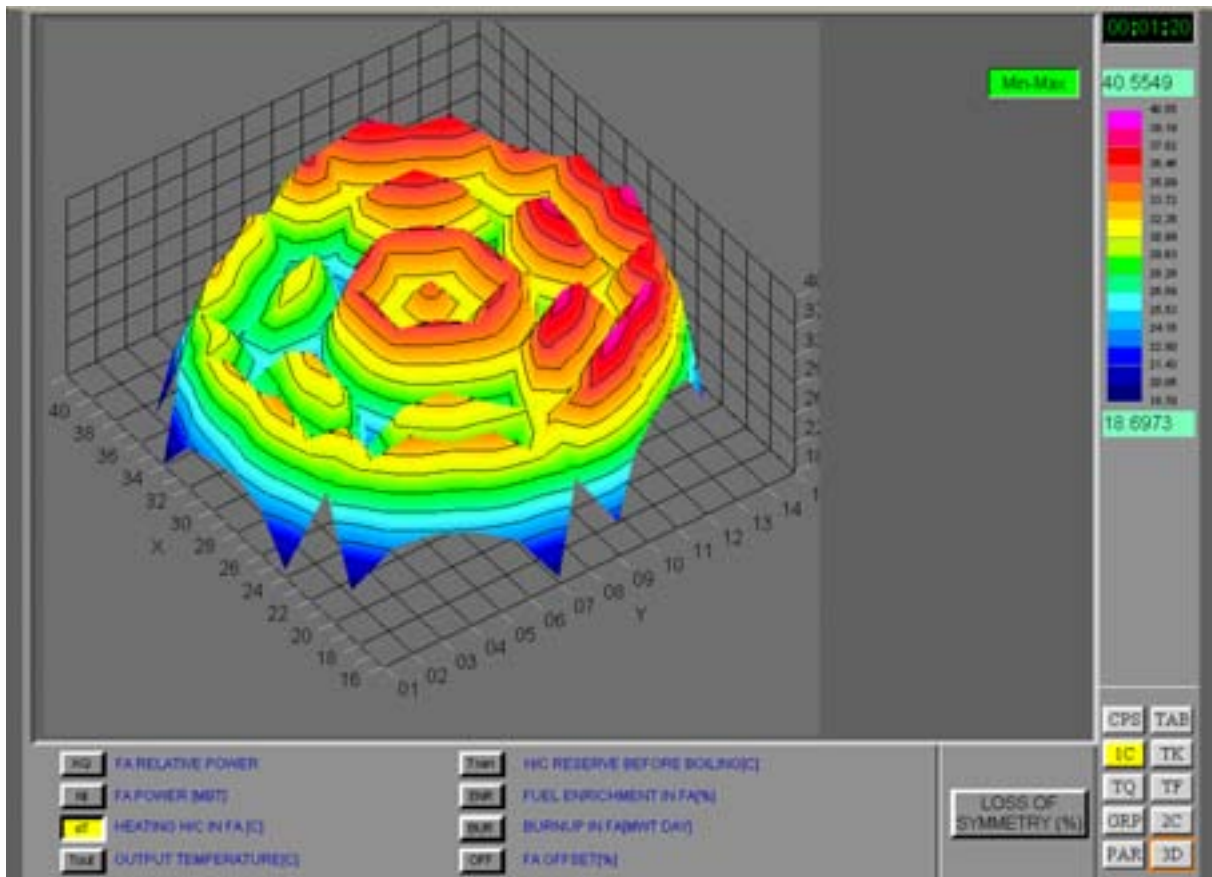
The pressed button is highlighted in yellow.

On the right of the screen there is a color scale. It changes according to the parameter selected for display on 3D diagram. When the button Min-Max is pressed (highlighted in green), the upper and lower limits of the scale will follow the changes of the parameter selected for display on the 3D diagram.

The 3D diagram can be rotated in any direction by pressing the left mouse button and moving the mouse in the desired direction.

The 3D plot options can be changed through the 3-Dimensional Plot menu. To open the Three Dimensional Plot menu click the right mouse button while the cursor is on the 3D diagram and select item “3-Dimensional Plot”.

The LOSS OF SYMMETRY button should be used when there is a significant deterioration of core parameters from a symmetrical distribution, e.g. one or two MCP trips, etc. When the distribution is close to symmetrical the use of this button will not result in a useful picture.



4. SIMULATOR EXERCISES

4.1. Overview

This section provides a brief description of training tasks included in the Task Manager program in the form of scripts. Every task has an individual scenario - the sequence of events written in the form of a script. While the task is executed, major parameters are recorded to a protocol file. After the task is completed the protocol can be analyzed in the Protocol Viewer Window.

It is also possible to perform actions that are carried out automatically from the script manually.

The Simulator can be utilized for achieving various kinds of training objectives. Three groups of training tasks are presented in the task list. They represent examples of objectives which can be achieved with the WWER-1000 Reactor Department Simulator training:

- Group 1. Tasks A01- A16 demonstrate various physics phenomena of the reactor core and are intended for understanding of reactor core neutronics.
- Group 2. Tasks C01-C12 provide the opportunity to train user diagnostic skills.
- Group 3. Tasks D01-D07 demonstrate different transients and are intended for user familiarization with WWER-1000 operation experience.

There are 2 core loading patterns presented in the simulator – 1st loading and 5th loading. The list of tasks for both loadings is essentially the same. The difference between these two core loadings is reviewed in Exercise 01.

Actions performed within similar tasks for the 1st and for the 5th core loadings are the same.

Every training task may be executed automatically by double clicking the task's icon from the Task Manager window. After analysis of the results it is possible to repeat the same sequence of events manually. For that it is necessary to select the training task "Rated state operation. 1st core-loading" (for the 1st core loading) or "Rated state operation. 5th core-loading" (for the 5th core loading). These are steady states with the reactor operating at rated power without any script.

4.2. Training exercises

4.2.1. Exercise 01

The objective is to understand the difference between the two core loading patterns presented in the simulator – 1st loading and 5th loading.

Select the task "Rated state operation. 1st core-loading" (for the 1st core loading).

Start the simulation.

Go to the Trend page (GRP).

Press button <ENR> to display the fuel enrichment in the fuel assemblies.

Watch the enrichment and fuel loading pattern.

Stop the simulation and exit.

Select the task "Rated state operation. 5th core-loading" (for the 5th core loading).

Repeat steps from 2 to 6. Compare fuel enrichment and fuel loading pattern.

The difference in fuel enrichment between two loadings and the respective difference in water-uranium-235 ratio results in different behavior in transients.

4.2.2. Exercise 02

The objective is to understand the difference between beginning of cycle and end of cycle core states.

Task “A03_Reactivity effects for BOC”

Initial condition (IC) – beginning of fuel cycle (BOC). The training scenario includes a reactor scram – due to emergency protection system (EP) actuation by key. After that the following parameters are determined:

- scram (EP) control rods “weight,”
- power reactivity factor (fuel temperature + coolant temperature),
- iodine poisoning value,
- xenon unpoisoning value.

After the simulation is completed, go to the protocol window and select variable REACTIVITY to be displayed on the plot. The resulting values of the four parameters listed above are written in the protocol text window.

Task “A04_Reactivity effects for EOC”

Initial condition – end of cycle (EOC). The sequence of events is exactly the same as for the Task A03.

Repeat the same tasks for 5th core loading. Record the results in the table below:

TABLE VII.

Parameter name	1 st core loading		5 th core loading	
	BOC	EOC	BOC	EOC
Scram (EP) control rods “weight”				
Power reactivity factor				
Iodine poisoning value				
Xenon unpoisoning value				

4.2.3. Exercise 03

The objective is to understand the possibility of neutron power and offset oscillations via reactor core characteristics.

Task “A05_Xenon (power and offset) oscillations for BOC”.

The IC is 100% power steady state BOC. The training scenario includes actuation of the URB (AUU) system (automatic fast reducing of reactor power) and blocking the reactor protection systems. After that, oscillations of offset and neutron power occur. Watch OFFSET and N_POWER on the trend page (GRP). Upon completion of the scenario perform protocol analysis. Select variables OFFSET and N_POWER to be displayed on the plot and adjust the limits.

Tasks “A06_Xenon (power and offset) oscillations for EOC” and “A07_Xenon (power and offset) oscillations for MOC” have the same sequence of events as the Task A05.

Repeat the same three tasks for the 5th core loading. Compare the amplitude of offset oscillations and damping. Explain the reasons.

4.2.4. Exercise 04

The objective is to understand the possibility of neutron power and offset oscillations due to control rod bank movement in and out of the core.

Task “A08_Xenon oscillations of the axial offset at constant reactor power for BOC 1h delay”.

Initial condition – 70% reactor power, steady state. Control rod bank #10 is inserted into the core from 80% to 40% of reactor height; it stays inserted for 1 hour and then it is withdrawn from the core to the initial position. Constant reactor power is maintained by boron, using the plant boron regulation system or boron regulator of the simulator. Thus, offset oscillations are initiated.

Watch OFFSET on the trend page (GRP). Upon completion of the scenario, perform protocol analysis. Select variable OFFSET to be displayed on the plot and adjust the limits.

Tasks A09 - A16 have a similar sequence of events as for Task A08. The only difference is in the time duration of keeping the control rod bank in the inserted position.

Repeat the same three tasks for the 5th core loading. Compare the amplitude of offset oscillations and damping. Explain the reasons.

4.2.5. Exercise 05

The objective is to return reactor into operation at 100% power level after hot shutdown.

Task “C01_Hot shutdown and start up without failures at BOC”.

The IC is 100% power steady state BOC. The training scenario includes emergency protection system actuation. After reactor scram, the user actions shall be as follows:

1. Hot shutdown after reactor scram (EP).
2. After stabilization of pressure in the secondary circuit, BRU-K (relief valve of steam dump to condenser) switch to «II» mode. Introduce boron up to 12 – 16 g/kg. After that start withdrawal of control rod groups beginning from 1 to 9. Leave group 10 at height about 80%. Start rising reactor power level by boron dilution. Reach reactor criticality (reactivity = 0, reactor period is positive). Save this simulator state. Analyze parameters.
3. Start further increase of reactor power. At 1-5% power level save state. Further rise power up to 35%. Stabilize at this power level by switching on ARM (ACP) (automatic regulator of power). Switch BRU-K to “watch” regime (P1). Switch turbine stop valves to automatic regime. Analyze parameters.
4. Switch off ARM(ACP). Control rod group 10 should not be lower than 70%. If it is lower, it will be need to be withdrawn up to 70-80%. Further increase of power up to 75% is achieved by boron dilution. Stabilize offset oscillations.

5. Further rise power level up to 100%. Stabilize offset oscillations at this level. Analyze parameters. Save simulator state.

4.2.6. Exercise 06

The tasks listed below provide the opportunity to train diagnostic capabilities:

C02 “Hot shutdown and start up with single failure_1 at BOC”;

C03 “Hot shutdown and start up with single failure_2 at BOC”;

C04 “Hot shutdown and start up with single failure_3 at BOC”.

The objectives are to observe the response of the reactor systems, find out the equipment that failed, try to bring it back to operation and start up the reactor as in the Task C01.

User actions are the same as for Exercise 05.

4.2.7. Exercise 07

The tasks listed below demonstrate different transients and are intended for user familiarization with WWER-1000 operation experience.

TABLE VIII.

Task Name	Description
D01 MCP trip (1 of 3, 1 of 4)	Trip of main circulation pump, 1 of 4 in operation
D02 Trip of two adjacent MCPs	Trip of two MCPs in two adjacent loops
D03 Closure of turbine stop-valve	Closure of main turbine stop valve
D04 Trip of 1 out of 2 FWP	Trip of 1 of 2 feed water pumps in operation in secondary circuit
D05 1 of 4 MCP jam	Jam of 1 MCP of 4 MCPs in operation
D07 Loss of feed water	Complete loss of feed water on secondary side

The objectives are to observe the triggering event, to watch the response of the reactor systems and to monitor the main parameters via the trend page.

Appendix A

List of main variables for WWER-1000 Reactor Department Simulator

BURNUP_CAS	fuel burnup for selected FA, Mw*day
CAMERA1	readings of ionizing chamber 1, %
CAMERA2	readings of ionizing chamber 2, %
CAMERA3	readings of ionizing chamber 1, %
DT_CAS	coolant heating for monitoring FA, °C
KQ	maximal assembly-wise non-uniformity factor
KQ_CAS	non-uniformity factor for monitoring FA
KV	maximal control volume-wise non-uniformity factor
KV_CAS	control volume-wise non-uniformity factor for monitoring FA
N_POWER	reactor neutron power, %
OFFSET	offset, %
OFFSET_CAS	offset for selected FA, %
PERIOD	reactor period, sec
PN_AX1	axial non-uniformity of neutron flux Kz (section 1)
PN_AX10	axial non-uniformity of neutron flux Kz (section 10)
PN_AX2	axial non-uniformity of neutron flux Kz (section 2)
PN_AX3	axial non-uniformity of neutron flux Kz (section 3)
PN_AX4	axial non-uniformity of neutron flux Kz (section 4)
PN_AX5	axial non-uniformity of neutron flux Kz (section 5)
PN_AX6	axial non-uniformity of neutron flux Kz (section 6)
PN_AX7	axial non-uniformity of neutron flux Kz (section 7)
PN_AX8	axial non-uniformity of neutron flux Kz (section 8)
PN_AX9	axial non-uniformity of neutron flux Kz (section 9)
RA11P02	pressure before Main steam valve 1, kg/cm ²
RA11S03	turbine stop & regulating valves
RA11T01	steam temperature before Main steam valve 1, °C
RA12P02	pressure before Main steam valve 2, kg/cm ²
RA12T01	steam temperature before Main steam valve 2, °C
RA13P02	pressure before Main steam valve 3, kg/cm ²
RA13T01	steam temperature before Main steam valve 3, °C
RA14P02	pressure before Main steam valve 4, kg/cm ²
RA14T01	steam temperature before Main steam valve 1, °C
RC11P01	pressure in Main Steam Collector, kg/cm ²
REACTIVITY	reactivity, %
RL71F01	feed water flow rate to SG-1, t/h
RL71S02	position of feed water regulator SG-1
RL72F01	feed water flow rate to SG-2, t/h
RL72S02	position of feed water regulator SG-2
RL73F01	feed water flow rate to SG-3, t/h
RL73S02	position of feed water regulator SG-3
RL74F01	feed water flow rate to SG-4, t/h
RL74S02	position of feed water regulator SG-4
SORT_CAS	fuel enrichment for selected FA, %
T_POWER	reactor thermal power, Mw
T1KONTUR	primary circuit mean temperature, °C
T2KONTUR	mean temperature of secondary circuit, °C

TB10F04	flow rate of boron concentrate 1, t/h
TB10F06	flow rate of boron concentrate 2, t/h
THOUR	system time, h
TIME EFF	effective simulation time
TK21F02	feed water & sealing water flow rate (pump 1) , t/h
TK22F02	feed water & sealing water flow rate (pump 2) , t/h
TK30F02	sealing water flow rate, t/h
TK40F01	feed water flow rate to primary circuit, t/h
TK80F01	purging water flow rate, t/h
TMIN	system time, min
TPOW_CAS	thermal power for selected FA, Mw
TQ12F01	flow rate from reactor cooling down system (planned) , t/h
TQ13F01	flow rate from reactor cooling down system (emergency) , t/h
TQ14F01	flow rate from high pressure emergency reactor cooling down system, t/h
TSEC	system time, sec
TV30Q01	boron concentration, g/kg
TX10F01	flow rate from SG emergency feed water system, t/h
TZAP	temperature margin up to boiling, °C
TZAP_CAS	temperature margin up to boiling for selected FA, °C
YA11T01	coolant temperature in hot leg 1, °C
YA12T01	coolant temperature in cold leg 1, °C
YA21T01	coolant temperature in hot leg 2, °C
YA22T01	coolant temperature in cold leg 2, °C
YA31T01	coolant temperature in hot leg 3, °C
YA32T01	coolant temperature in cold leg 3, °C
YA41T01	coolant temperature in hot leg 4, °C
YA42T01	coolant temperature in cold leg 4, °C
YB10L14	level in SG-1, mm
YB10P10	pressure in SG-1, kg/cm ²
YB20L14	level in SG-2, mm
YB20P10	pressure in SG-2, kg/cm ²
YB30L14	level in SG-3, mm
YB30P10	pressure in SG-3, kg/cm ²
YB40L14	level in SG-4, mm
YB40P10	pressure in SG-4, kg/cm ²
YC00P01	pressure in reactor (above the core) ,kg/cm ²
YP10L01	level in pressurizer, mm
YP10P01	pressure in pressurizer, kg/cm ²
YP20P05	pressure in bubbler-condenser, kg/cm ²
YSAASBB	position of control rods .. AA- y-coordinate, BB -x-coordinate

Appendix B

List of items at the alarm panel for WWER-1000 Reactor Department Simulator

EP	Emergency protection – reactor scram (AZ)
EP from CR	EP from Control Room key by operator
N > Setpoint	Neutron power is greater than setpoint
dT s1 < 10	Margin up to boiling (saturation) in primary circuit is less than 10°C
P1 < 150 N > 75	Pressure in reactor is less than 150 atm, while neutron power is greater than 75%
L SG – 650	Level in SG is less than 650 mm
P mss > 80	Pressure in main steam collector is greater than 80 atm
T < 10	Reactor period is less than 10 seconds
dT s1-2 > 75	Difference of saturation temperatures between primary and secondary circuits is greater than 75°C
P1 < 140	Pressure in reactor is less than 140 atm
F < 46	Frequency in external electric network is less than 46 Hz
P1 > 180	Pressure in reactor is greater than 180 atm
< 2 MCP	Less than 2 MCPs are in operation
T h.l. > 330	Temperature in hot leg is greater than 330°C
dP MCP < 2.5	Pressure drop at MCP is less than 2.5 atm
L Press < 400	Level in Pressurizer is less than 400 mm
PP-1	Preventive protection 1 (PZ-1)
PP from CR	PP from Control Room key by operator
N > Setpoint	Neutron power is greater than setpoint
T < 20	Reactor period is less than 20 seconds
P sg > 70	Pressure in SG is greater than 70 atm
P1 > 172	Pressure in reactor is greater than 172 atm
F < 49	Frequency in external electric network is less than 49 Hz
T1 > 325	Temperature in hot leg is greater than 325°C
MCP trip	MCP trip
TDFWP trip	Feed water pump trip
Generator trip	Generator trip
PCR	Power level limiting regulator (ROM)
N heat > Setpoint	Thermal power is greater than setpoint2
MSV closed	Main steam valves are closed
PP-2	Preventive protection 2 (PZ-2)
N > Setpoint	Neutron power is greater than setpoint
P1 > 165	Pressure in reactor is greater than 165 atm
CR drop	Dropping of control rod (failure)
AUU	Fast reducing of reactor power (URB)
AUU from CR	AUU from Control Room key by operator
2 MCP trip	Two MCP trip
TDFWP trip	Feed water pump trip

Generator trip	Generator trip
MSV closed	Main steam valves are closed
SG	Preventive protection 1 (PZ-1)
L SG1 > 39.5	Level in Steam Generator 1 is greater than 39.5 mm
L SG2 > 39.5	Level in Steam Generator 2 is greater than 39.5 mm
L SG3 > 39.5	Level in Steam Generator 3 is greater than 39.5 mm
L SG4 > 39.5	Level in Steam Generator 4 is greater than 39.5 mm
L SG1 > 22	Level in Steam Generator 1 is greater than 22 mm
L SG2 > 22	Level in Steam Generator 2 is greater than 22 mm
L SG3 > 22	Level in Steam Generator 3 is greater than 22 mm
L SG4 > 22	Level in Steam Generator 4 is greater than 22 mm
MSIV1 closed	Main steam isolating valve 1 is closed
MSIV2 closed	Main steam isolating valve 2 is closed
MSIV3 closed	Main steam isolating valve 3 is closed
MSIV4 closed	Main steam isolating valve 4 is closed
MSSV SG1 opened	Main steam safety valve for SG1 is opened
MSSV SG2 opened	Main steam safety valve for SG2 is opened
MSSV SG3 opened	Main steam safety valve for SG3 is opened
MSSV SG4 opened	Main steam safety valve for SG4 is opened
P SG1 > 84	Pressure in Steam Generator 1 is greater than 84 atm
P SG2 > 84	Pressure in Steam Generator 2 is greater than 84 atm
P SG3 > 84	Pressure in Steam Generator 3 is greater than 84 atm
P SG4 > 84	Pressure in Steam Generator 4 is greater than 84 atm
L SG1 > 62	Level in Steam Generator 1 is greater than 62 mm
↑A II C	Raising of radioactivity in secondary circuit
MSC	Main Steam Collector
P mss < 52	Pressure in MSC is less than 52 atm
P mss < 56	Pressure in MSC is less than 56 atm
P mss < 62	Pressure in MSC is less than 62 atm
1C	Primary Circuit
L Press < 500	Level in pressurizer less than 500 mm
L TY20B01 > 1500	Level in the tank of organized leakages is greater than 1500 mm
TK misbalance	Misbalance of feed water flow and purging water flow
MSSV Press opened	Main safety steam valve of pressurizer is opened