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WORKSHOP ON ADVANCED NUCLEAR POWER PLANT SIMULATION

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Overview of the VVER Simulator

Part I
Distinctive features of VVER

Part II
VVER-1000 Reactor Department Simulator

Part III
Overview of ENICAD - Simulator shell and code generators

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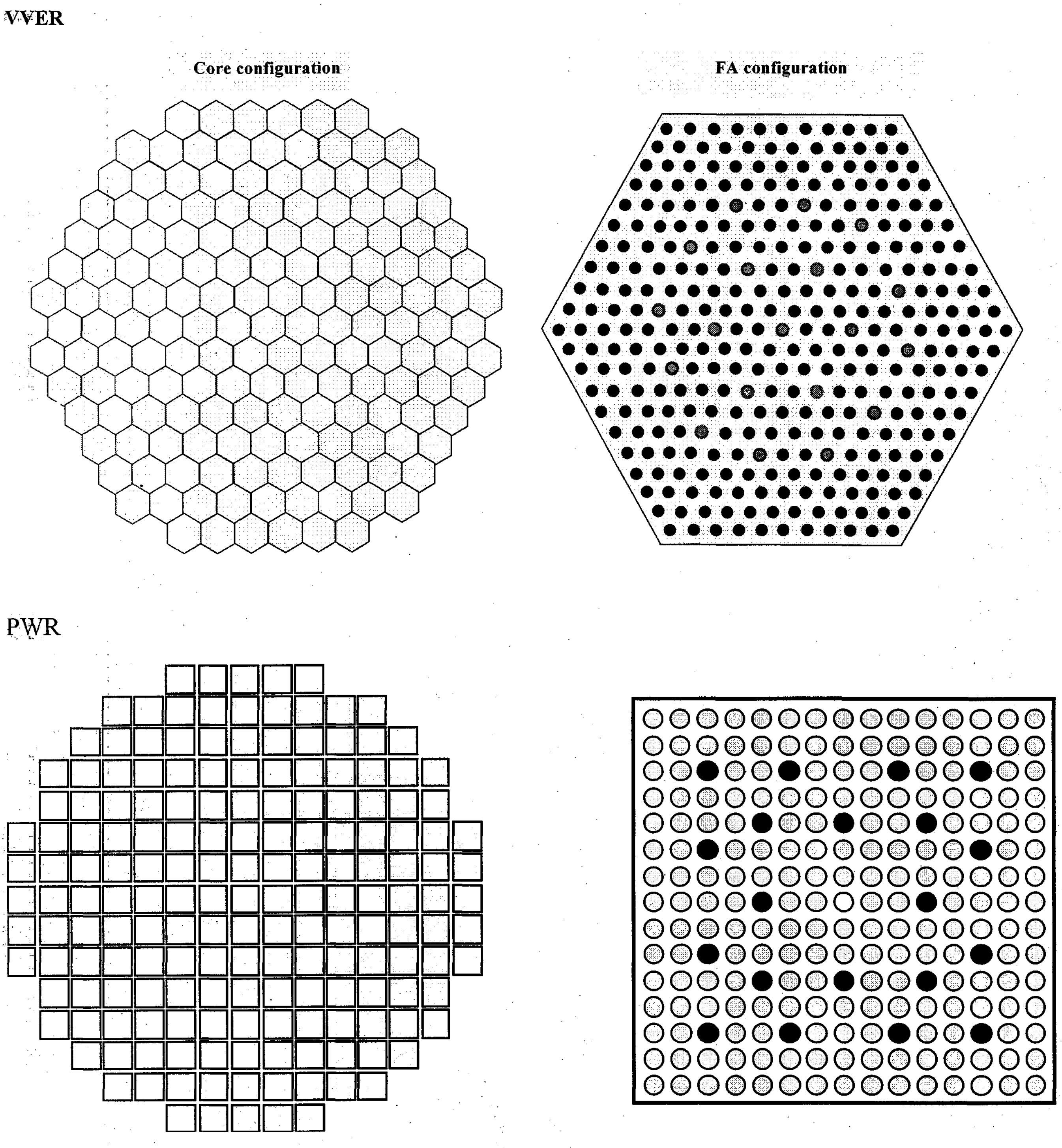
These are preliminary lecture notes, intended only for distribution to participants

Part 1.

Distinctive features of VVER



Core and Fuel Assembly Configuration for VVER and PWR

Construction characteristics of VVER and PWR core

Parameter name	Value	
	PWR	VVER
Total number of FAs in the core,	**177**	**163**
Number of FAs with control rods,	**49**	**61**
Number of FAs with SBA (for 3-years life time and for the first year).	**-**	**54**
Height of heating part (in cold state), m	**3.55**	**3.53**
Step between FAs, m	**0.215**	**0.236**
Pass section of the core in heating part, m^2	**4.41**	**4.17**
Coolant's flow rate through the core, kg/s	**15984**	**17650**
Reactor thermal power.	**3002**	**3000**
Maximum deviation, including measurements error and regulation precision, MW	**-**	**120**

Characteristics of VVER and PWR FA

Parameter name	Value	
	PWR	VVER
Fuel rods step, mm	**215.6**	**234**
Step between fuel rods, mm	**14.30**	**12.75**
Number of fuel rods,	**205**	**312**
Number of tubes for absorber elements,	**20**	**18**
Number of tubes for NMC,	**1**	**1**
Length of FA active part, mm	**3550 (3564)**	**3530 (3550)**
Number of distant grids (support plates),	**6**	**14**
Grid material	**zirconium**	**steel (zirconium)**

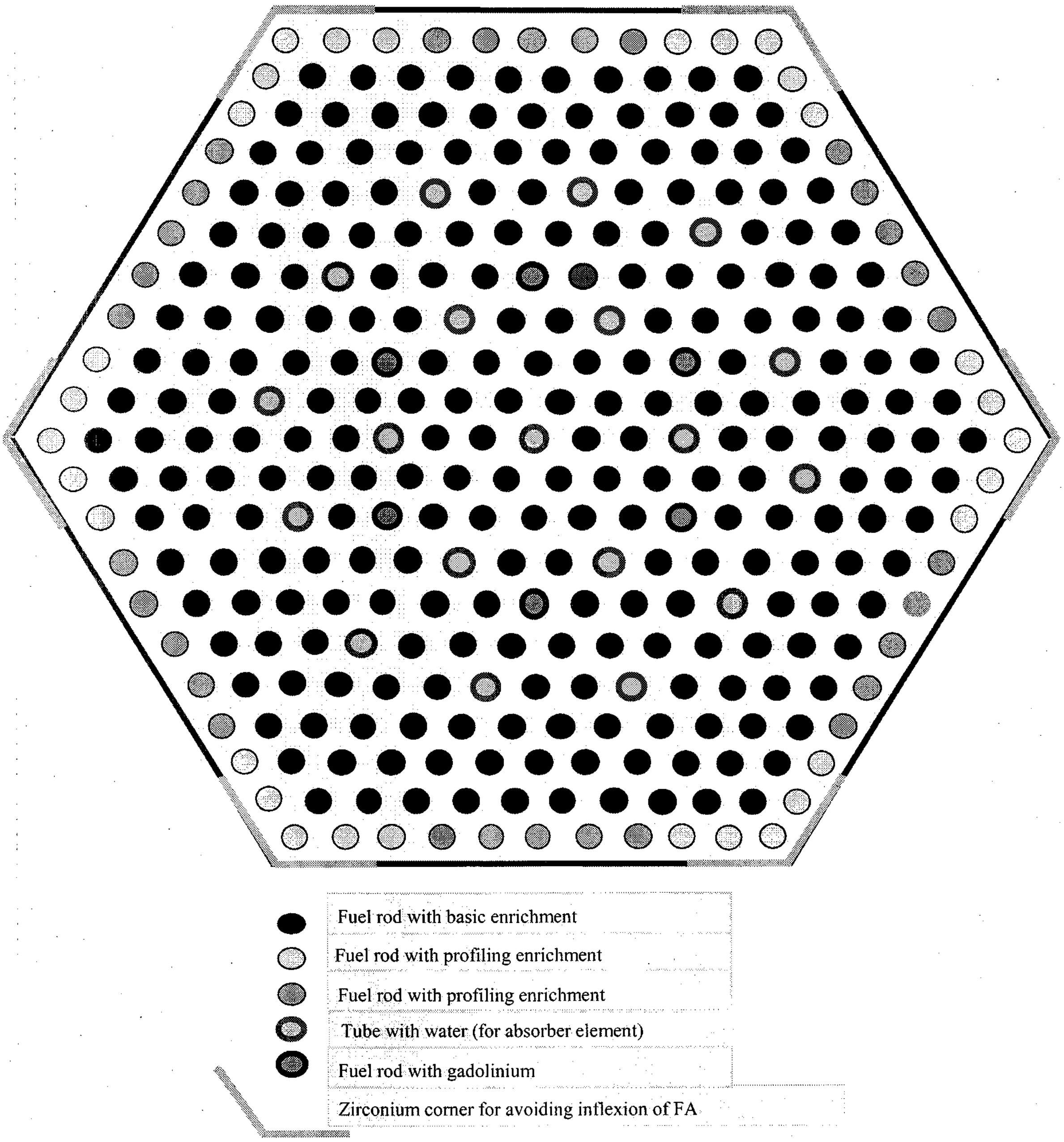
Characteristics of VVER and PWR fuel rod

Parameter name	Value	
	PWR	VVER
Fuel rod diameter, mm	**10.75**	**9,1**
Cladding thickness, mm	**0.725**	**0,69**
Cladding material, mm	**Zircalloy**	**alloy – 110**
Fuel part diameter, mm	**9.11**	**7.53**
Fuel material	**UO_2**	**UO_2**
Diameter of central whole in fuel pellet, mm	**-**	**2.3**
Fuel density, g/cm^3	**10.28**	**10.4**
Enrichment of feeding fuel, %	**3.5, 3.8, 4.0,4.3**	**3.3, 4.4, 3.0,4.0**

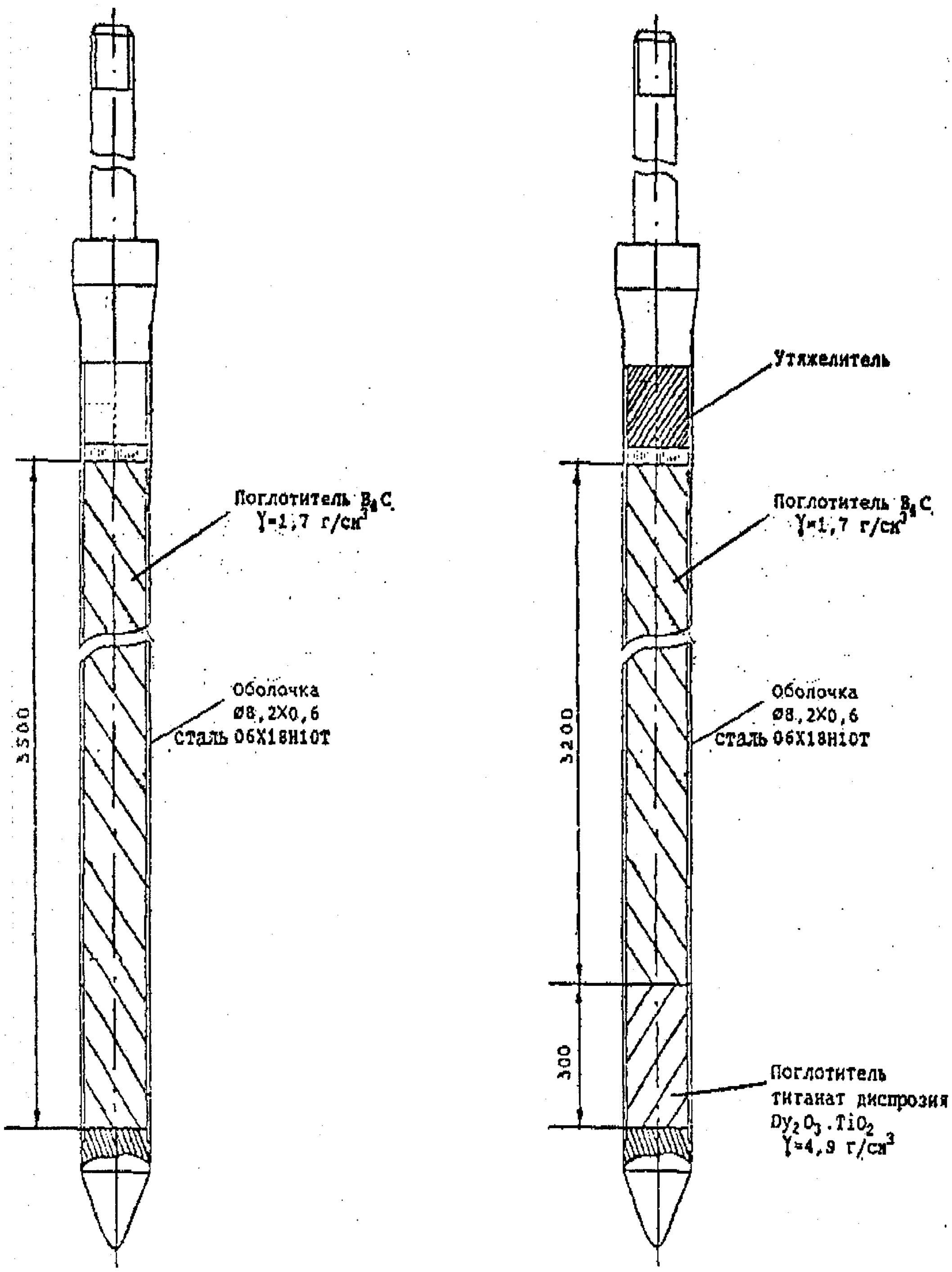
Thermal hydraulic parameters of reactor and core

Reactor type	VVER-1000		PWR-1000	
1- Rated regime 2- Regime with deviations	1	2	1	2
1.Thermal power, MW	3000	3210	3002	3360
2.Primary circuit pressure, MPa	15.7	15.4	15.4	15.4
3.Coolant temperature at reactor inlet, ^{0}C	289.7	292.3	291.5	291.5
4. Coolant temperature at reactor outlet, ^{0}C	320.0	325.5	326.2	332.2
5. Coolant temperature at the outlet of FA with maximal load,. ^{0}C	328.0	334.0	342.9	344.2
6.Design basis coolant flow rate through reactor, kg/s	18300	17620	15984	14990
7.Value of flow aside from the core, %	3	3	6.5	6.5
8.Mass velocity in the core, kg/s/m^2	3850	3670	3338	3180
9. Mass velocity in the core through FA with maximal load	3560	3430	3129	3080
10.Mean coolant velocity in the core, m/s	5.4	5.4	—	—
11.Total surface of heat exchange in the core, m^2	5176	5176	4376	4376
12.Fraction of power released in fuel rods	1.0	1.0	0.95	0.95
13.Power release density in the core, W/cm^3	107	112	105	115
14.Thermal flow from the fuel rod surface (mean), W/cm2	58	61	67.3	71.2
15.Linear thermal flow (mean), W/cm	165.7	172.3	227.7	242.5
16.Radial power release non-uniformity factor	1.5	1.5	1.6	1.6
17.Local nuclear power release non-uniformity factor	2.24	2.24	2.60	2.60
18. Local nuclear power release non-uniformity factor with adjustment on possible calculation errors	2.60	2.60	2.60	2.60
19.Maximal thermal flow from fuel rod surface, W/cm^2	150.7	165.7	155.2	173.8
20. Maximal linear thermal flow, W/cm	430.8	448.0	525.0	588.0
21.Maximal fuel temperature in stationary regime, ^{0}C	1690	1800	2030	2151
22.Maximal temperature of fuel rod cladding external surface, ^{0}C	350	352	346.6	346.7
23.Minimal factor of DNBR	1.73	1.19	1.86*	1.53*

New FA construction



Old and new constructions of VVER-1000 absorber elements



Summary of modifications in core construction

- **Criteria of optimal water-fuel ratio has changed on the basis of multiyear operation from 2.5 - 2.8 to 2.0 - 2.2.**

- **Changing FA construction by replacement of steel with zircalloy and introduction of special zirconium corners for elimination of FA azimuthal twisting.**

- **Utilization in fuel assemblies with uranium-gadolinium fuel instead of burnable absorber. This permits to solve the problem of incomplete burning out of burnable absorber to the end of fuel cycle, decrease initial boron concentration in the coolant and increase number of control rods.**

- **Increasing number of control rods from 61 to 121, weighting and strength characteristics of absorber rods.**

- **Utilization of core refueling scheme «IN-IN-OUT» without worsening of core thermal safety.**

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Part 2.

VVER-1000 Reactor Department Simulator

Purpose:

RDS is a tool for initial and continuous training of Nuclear Safety Department personnel (Station Nuclear Engineers, etc.).

RDS can be effectively used in training of Control Room personnel (Reactor Operator, Unit Operator, Plant Shift Supervisor) and Reactor Department operation personnel.

Capabilities:

RDS simulates the RD equipment steady state and transients in all normal operation and emergency regimes which includes the complete list of transients and all emergency regimes considered in VVER-1000 Safety Analysis Reports.

In RDS the following technological plant systems are simulated:

- Reactor control and shutdown system,
- In-core and ex-core reactor instrumentation,
- Four primary circulation loops,
- Pressurizer and pressure control system,
- Emergency core cooling system,
- Emergency boron injection system,
- Emergency feed water supply system,
- Primary circuit blow-up system,
- Primary circuit organized leakages system,
- Boron concentrate system,
- Emergency steam and gas release system,
- Containment,
- Steam generators and main steam collectors,
- Emergency steam release system,
- Secondary circuit control and shutdown system.

Table 9

List of signals for emergency protection

Signal name	Reason	Possible regimes
Reactor period in source range.	Reactor scram if reactor period is less than 10 sec.	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods. 3. Eject of control rod due to rapture of its cover.
Reactor period in intermediate range.	Reactor scram if reactor period is less than 10 sec.	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods 3. Eject of control rod due to rapture of its cover.
Reactor period in power range.	Reactor scram if reactor period is less than 10 sec.	1. Lowering boric acid concentration. 2. Uncontrolled withdrawal of control rods 3. Eject of control rod due to rapture of its cover.
Neutron flux density in power range. Nset.: setpoint for neutron power – set by operator according regulations. It can vary. Setpoint N_{set}. is set discretely with step 1% in range (3-107)%N_{nom}.	Reactor scram if reactor neutron power is greater than 107%Nset	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%Nset	1.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	1.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.	1.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 aatm)	Reactor scram in case of primary circuit pressure lowering	1.Unintended injection to pressurizer.. 2.Unintended actuation of pressurizer safety valve 3.LOCAs
Margin up to boiling in primary circuit is less than 10^{0}C for each loop	Reactor scram in case of increasing coolant temperature	Various.
Lowering pressure drop at MCP from 0,39 MPa to 0,25 MPa for time less 5c	Reactor scram in case of lowering 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/cm2) 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. Steamline rupture. 2. Feedwater line rupture 3. Non-closing of relief valves
MCP loss of power supply: • 1 of 2 operating MCPs while reactor power greater than: 5% Nnom;: • 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/cm2)	Reactor scram in case of raising pressure in SG	Turbine trip under conditions when relief valves can't provide rated dump of steam
Seism with intensity about 6	Reactor scram in case of seisms	Seism is more than designed basis
Loss of reliable power supply for CPS in 2 sections of 3	Reactor scram in case of CPS loss of power	CPS loss of power
SG level is lower 650mm while appropriate MCP is operating.	Reactor scram in case of loss of rated feedwater flow.	Rupture of feedwater pipelines, Trip of feedwater pumps.
Pressure in primary circuit greater 180 kgs/cm2	Reactor scram in case of raising pressure and temperature in primary circuit.	
Coolant temperature in any of hot legs t_{set}+8^{0}C	Unintended increasing of coolant temperature.	
Level in pressurizer is less than 4600 mm.	Reactor scram in case of lowering pressurizer level	LOCAs,.
Actuating AZ by key	Reactor scram by operator	Regulations requirements

Table 10

List of signals for preventive protection PZ-1

Signal name	Reason
Reactor period: - in source range, less than 20 s - in intermediate range, less than 20s - in power range, less than 20s	Lowering of reactor period
Neutron flux in power range is greater setpoint. Setpoint is variable 104:107 from Nset	Lowering reactor power in case of unintended loss of neutron power
Neutron flux in intermediate range is greater setpoint. Setpoint is variable 10:15 from Nset	Lowering reactor power during reactor startup
Thermal power is greater than permitted for current number of operating MCPs	Lowering reactor power with ROM
Pressure above reactor core greater 16,8 MPa (172 kgs/cm2)	Lowering reactor power in case of pressure growth
Coolant temperature in any loop $t_{nom}+3^0C$	Lowering reactor power in case of unintended growth of coolant temperature
Pressure in main steam collector is greater than 6,85 MPa (70 kgs/cm2)	Lowering reactor power due to increasing of steam pressure
Loss of power supply: - 1 MCP из 4-х работающих - 1-го MCP из 3-х работающих	Reducing reactor power due to decreasing of flow rate through the core
PZ-1 by key from control panel	Lowering reactor power by operator from control panel
Trip of 1 from 2 FWP in secondary circuit	Lowering reactor power to 75% Nnom with ROM

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Table 11

List of signals for preventive protection PZ-2

Signal name	Reason
Neutron flux is in source range. Setpoint variable - 10:15 from set value.	Limitation of reactor power during startup.
Pressure above the core is greater than 16.2 MPa (165 kgs/cm2)	Limitation of reactor power in case of raising pressure in primary circuit
Drop of one control rod	Limitation of power level to provide permitted values of neutron flux non-uniformity.

Table 12

List of signals for Fast Unit Unloading URB

Signal name	Reason
2 MCP trip of 4 MCP in operation	Lowering reactor power to 50% (opposite loops) or 40% (adjacent loops) in case of 2 MCP trip
Feed water pump trip	Lowering reactor power to 50% in case of closing of turbo FWP stop valves or reducing steam pressure after regulating valves to setpoint
Generator trip	Lowering reactor power to 40% in case of disconnection of generator from electric network
MSV closed	Lowering reactor power to 40% in case of disconnection of turbine by steam (closing of 2 main steam valves)

Part 3.

Overview of ENICAD – simulator shell and code generators

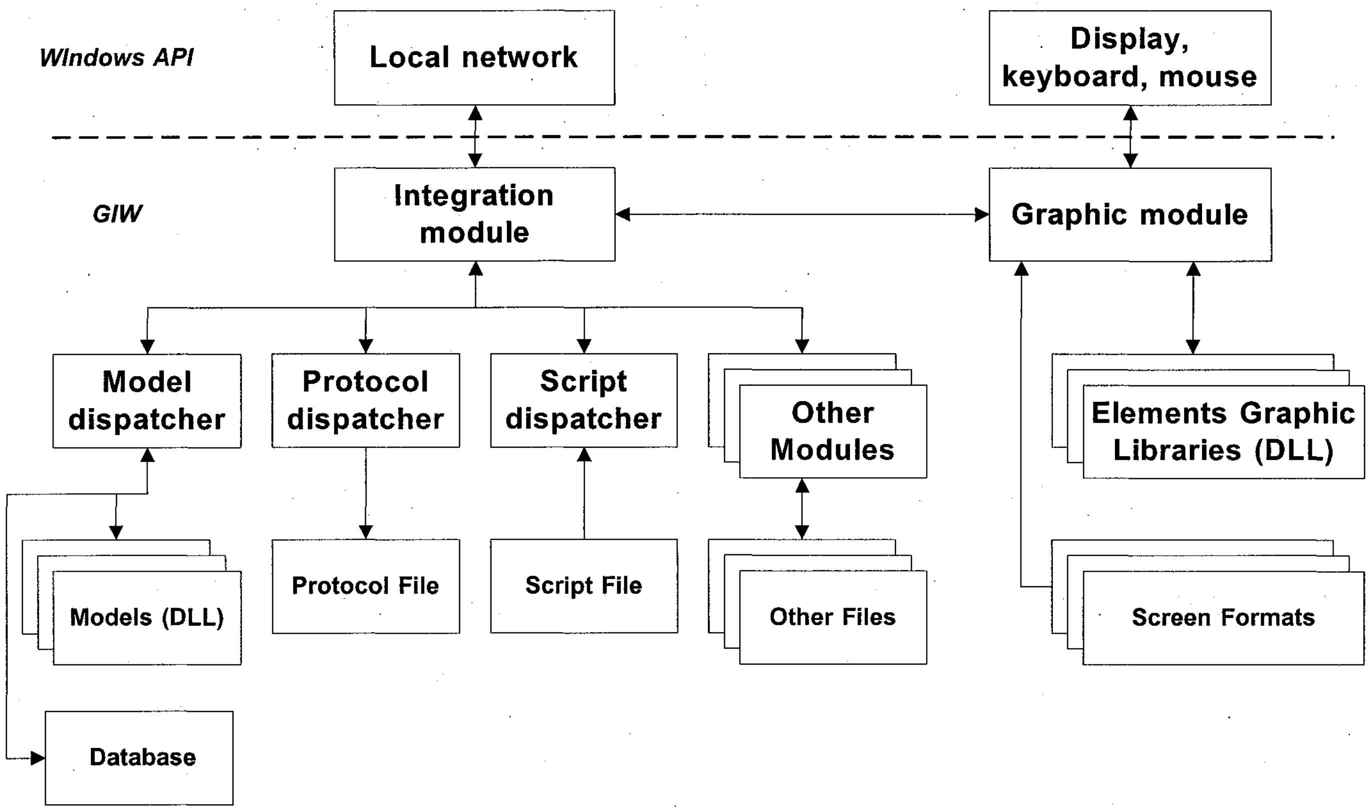

Fig.1 GIW structure

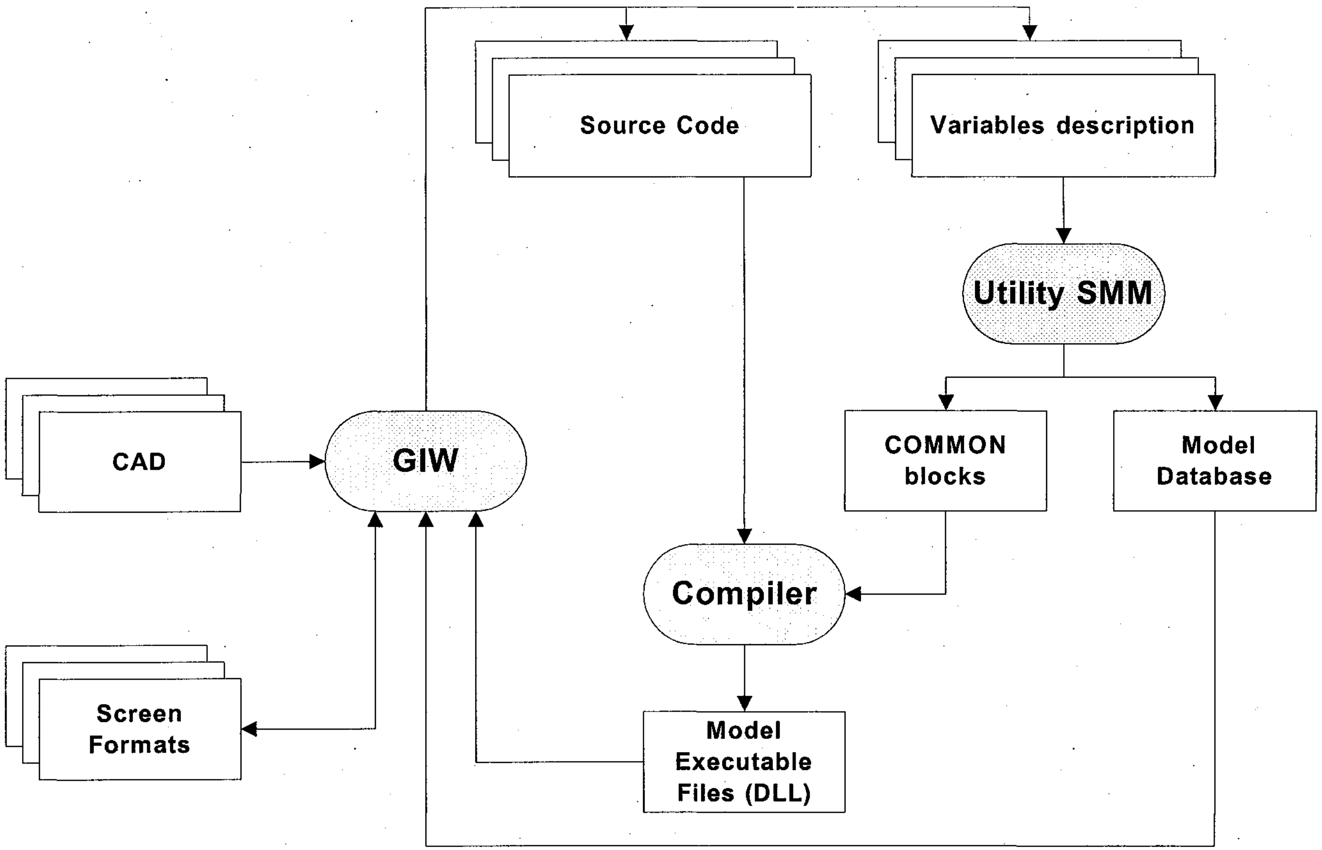

Fig.2 Project Development under GIW Shell

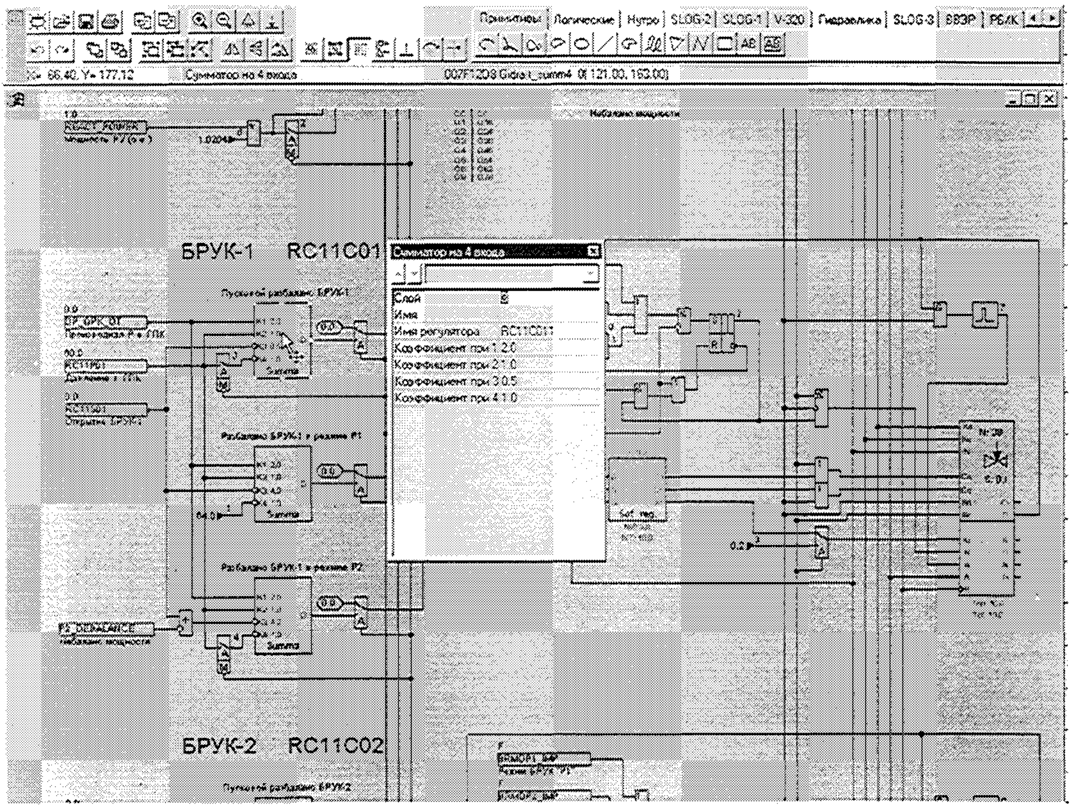

**Fig 3. Plant Automation CAD System.
Development format.**

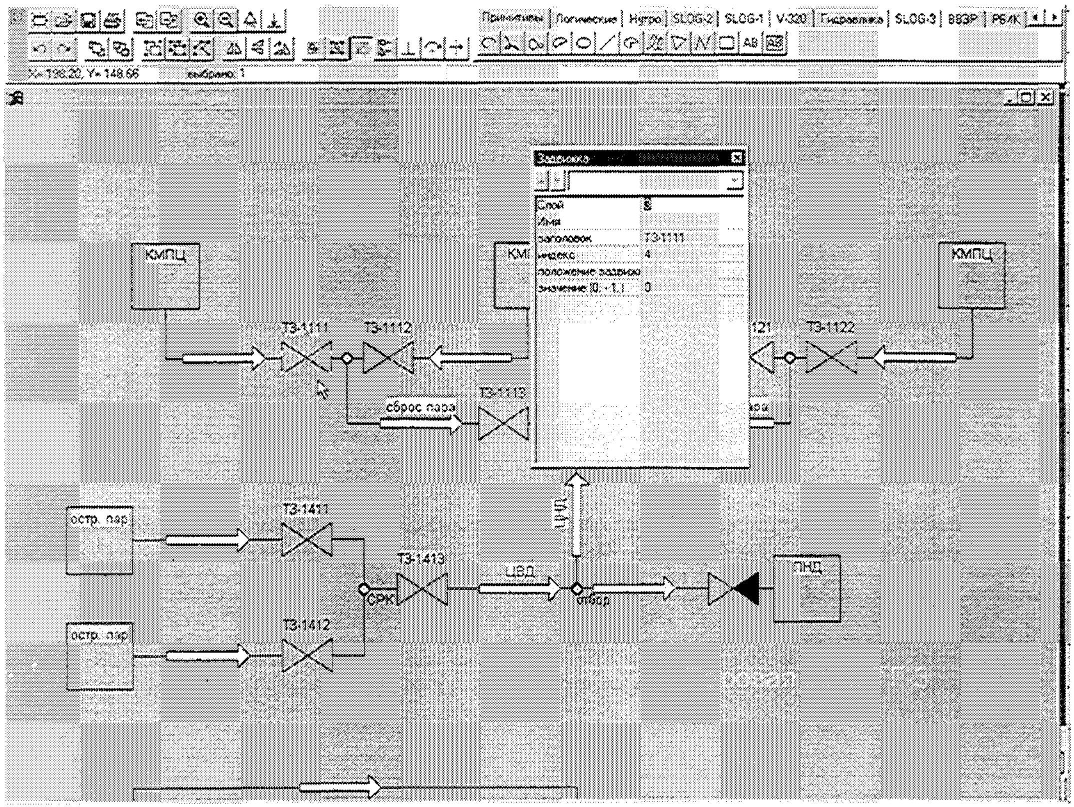

**Fig 4. Thermal Hydraulics CAD System.
Development format.**

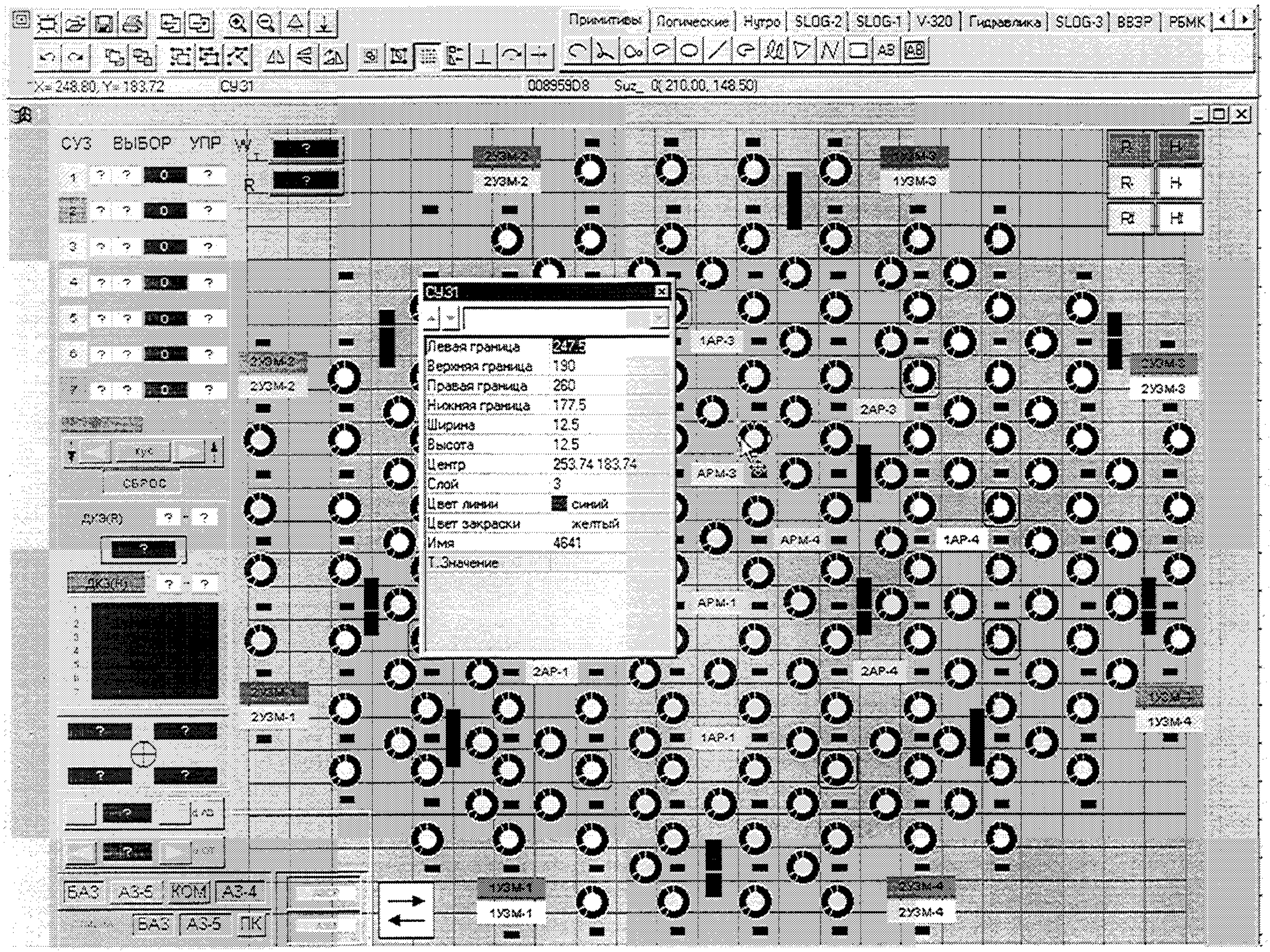

Fig 5. Simulator Screen Format example.
RBMK Control Rod System.