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DESIGN OF NPP OF NEW GENERATION BEING CONSTRUCTED AT THE NOVOVORONEZH NPP SITE

A. AFROV, V. BERKOVICH, V. GENERALOV,
Yu. DRAGUNOV, V. KRUSHELNITSKY
OKB "Gidropress" Experimental and Design Office,
Podolsk, Moscow Region, Russian Federation

Abstract

The design of a new generation NPP is described, underscoring advances in physical attributes and passive safety systems based on experiences with earlier designs at operating NPPs. This paper elaborates on systems for handling and storing radioactive wastes, on refinements in containment measures and on experimental and analytic validation of critical design factors.

1 GENERAL LAYOUT

The general layout of the nuclear power plant (NPP) was developed for a twin-unit station, considering the following requirements:

- maximum independence of each Unit;
- modular construction using the monoblock principle;
- optimal arrangement of buildings and structures of the main production process, as well as support production and auxiliary buildings and structures;
- mitigation of extreme external effects on NPP operability; and
- site zoning versus main production and auxiliary buildings.

The main building complex is in the center of the general layout (cf. Figure 1). It consists of the reactor and turbine plants, including outdoor main and standby transformers, standby power diesel units, spray ponds of the cooling system for reactor plant vital loads and a common sheltered center for management of emergency actions at the NPP.

2 INFORMATION ON THE MAIN PROCESS SOLUTIONS

2.1 Process solutions pertaining to the Main Building complex

2.1.1 General provisions and schematic solutions

The NPP with the ALWR class V-392 is a monoblock with a four-loop reactor and two turbine-driven feedwater pumps. The monoblock layout includes the double containment of the reactor plant, the turbine hall, the safety and auxiliary systems building and ensures a minimum length of engineering infrastructure lines, as well as high reliability in both normal operation and safety functions. The main engineering solutions of the design are aimed at:

- achieving a new qualitative level of safety compared to the V-320, and compliance with recommendations of the International Safety Group (INSAG)and the International Atomic Energy Agency (IAEA), etc.;
- application of mature processes and design solutions only (an evolutionary approach);
- improvement of economical performances compared to theV-320 and fossil fuel sources;

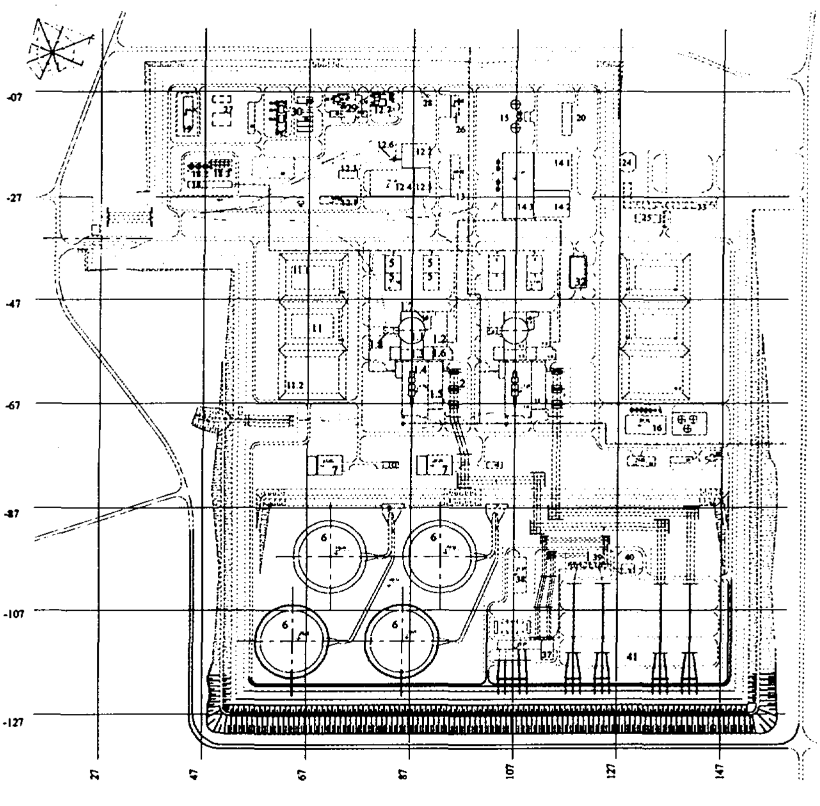
- application of feedback from operating NPPs and results of analysis by domestic and foreign companies; and
- creation of prerequisites and scope of preparatory works for implementation of a large power unit by the year 2020, with mature inherent safety features, intended for large-scale application in Russia. It features obvious advantages in efficiency compared to fossil fuel plants, irrespective of the region in which it is to be located.

The process solutions are essentially based upon:

- an advanced VVER-1000 with an improved reactor trip system, capable of maintaining reactor sub-criticality during cooling down to a temperature of 100 - 120 ^{0}C without boron injection; reactivity feedback is improved by negative coolant temperature coefficients of reactivity throughout the fuel cycle;
- advanced steam generators with a modified primary header structure using austenitic stainless steel in the heat exchanger tubes to extend service life; blow-down is arranged from the section with the highest salt concentration in the steam generator boiler water;
- an advanced reactor coolant pump with a shaft seal that prevents the coolant from leaking in case of a loss of power for 24 hours and loss of sealing water and other cooling media;
- auxiliary systems of reactor and turbine islands. The operating experience of many units is used during design development; new engineering solutions tested at operating units are used as well as an approach aimed at continuous diagnosis of disturbances.
- radioactive waste processing and storage systems using new advanced process solutions alongside traditional ones, with efficiency proven by many years of experience at domestic and foreign NPPs. Selection of new processes and new equipment is justified by research and development (R&D), and orders are placed with manufacturers to develop new equipment.

2.1.2 Schematic solutions

The reactor plant comprises four steam generator loops. The primary coolant temperature is 322^{0}C at the reactor outlet, and the design primary pressure is 17,6 MPa. Each loop has one reactor



Legend for buildings and structures

No. on Drwg.	CCS code.	Designation
1		Main building complex
1.1	UJA	Containment and Process Safety Systems Building
1.2	UKC	Normal Operation Process Bldg. of Active Water Treatment System
1.3	UBP	Control Safety System Bldg.
1.4	UMA	Turbine Generator and Deaerator Bldg., incl. Lub. oil facilities
1.5	UBA	Normal Operation Electric Device Bldg. and Ventilation Centre
1.6	UYB	Unit Sanitary and Welfare Building
1.7	UKH	Ventilation stack
1.8	UJG	Transportation bridge
1.9	UCA	Monitoring and Control Systems Bldg.
2	UBF	Transformers with transportation tracks
5	UKD	Standby Diesel Power Plant Safety Systems
6	URA	Cooling Tower
7	URD	Pump station for cooling tower with back-up tank
8	URJ	
11	URR	Spray pool
12		Set of auxiliary structures of the controlled area
12.1	UFC	Fresh fuel storage with simulator room
12.2	UKS	Solid waste treatment Bldg. with storage Cooling towers' discharge channels and integrated structures
12.3	UFA	Spent fuel container storage
12.4	UKU	Repair shops of controlled area
12.5	UKW	Laboratory and welfare bldg. and active laundry
12.6	UKH	Ventilation stack with air duct for CRS of controlled area
12.7	UCY	Biological and deep treatment station for waste water from controlled area
14		Set of auxiliary structures of the free access area Common auxiliary building
14.1	USY	Control repair shop and control motors store
14.2	UYC	Administration and Laboratory bldg. and Canteen
14.3	UGD	Chemical water treatment with tank facilities
18	UTM	Lub. oil and diesel facilities
25	UVE	Gate house
27		Production and fire water supply structures
28		Domestic waste pump station of the free access zone
29		Sewage piping of drains water contaminated with petroleum products
30		Sewage piping of production and rain drains water
32	O1U2M	Protected central control station
33		Parking lot
41		Open-air switchgear for 500 kV

FIG. 1 General layout of NV NPP-2 [Novovoronezh NPP-2]

coolant pump (RCP) with external motor and necessary inertia performance and one horizontal steam generator with submerged heat exchanging surface. The live steam pressure is 6,37 MPa (design pressure is 8,0 MPa (80 bar)), and the steam capacity of the plant as a whole is ≈ 4 x 1470 t/h.

The reactor plant is equipped with four, first-stage accumulators with a nitrogen cushion pressure of 6,0 MPa (60 bar). The first-stage accumulators are connected in pairs through check valves to the emergency nozzles in the upper and lower plenums of the reactor pressure vessel. The bypass line of each RCP is equipped with systems for high-temperature mechanical cleaning of the coolant. Tanks with high concentration boric acid and quick-acting valves of the boron supply system are connected to the bypass lines, backing up the function of the solid absorbers of the reactor scram system.

The 1000 MW power turbine plant has an optimized steam cycle with regenerative heating of the feedwater. The feedwater plant consists of two non-redundant, turbine-driven feedwater pumps. It is capable of maintaining 70 per cent of the unit steam capacity with one pump operating.

The process part of the unit safety system is a train-type structure to fulfil the criteria of high reliability of crucial safety functions and minimize common-cause failure probability. This approach led to safety systems design with mutually redundant active and passive trains; this diversity covers practically all the main safety functions. Safety systems solutions are described in more detail in the following sections

2.1.3 Nuclear Fuel Management

The nuclear fuel management system facilitates all fuel handling at the NPP and comprises the following systems:

- fresh fuel storage and management system, including fuel transfer to the reactor department;
- the refueling system;
- a spent fuel handling system consisting of,
    - spent fuel storage near the reactor, and
    - spent fuel storage in a special building outside the reactor building;
- a system of nuclear fuel transportation at the NPP site covering operations from reception of a special carriage with fresh fuel, to sending away a special carriage with spent fuel and internal site transportation of nuclear fuel.

Figure 2 depicts the movement of fuel to, within and away from the NPP.

2.1.3.1 Fuel Handling in the Fresh Fuel Storage Facility (FFSF)

The FFSF is common to the whole NPP and is in a separate building at the site. For nuclear safety, the FFSF is assigned to seismic category 1, which means that FFSF structures and the fresh fuel handling equipment are designed to withstand extreme external effects. FFSF is designed to house:

- 170 fuel assemblies necessary to refuel 2 reactors, plus 20 per cent;
- 180 fuel assemblies in casks for complete core loading plus 10 per cent.

FFSF is equipped with racks to store the fuel assemblies and absorbing rods prepared for refueling. The rack is a metal work consisting of three slabs rigidly connected by pillars. The cells housing fuel assemblies are arranged in a 400-mm, triangular array. Before refueling, the prepared fuel assemblies are installed by the FFSF into a site-internal transportation container, which in turn is mounted onto a site internal platform and transported to the reactor department.

2.1.3.2 Fuel Handling in the Reactor Island

The main fuel handling operations conducted within the NPP reactor island comprise:

- fresh fuel delivery into the reactor island and loading into the reactor;
- spent fuel removal from the reactor;
- spent fuel storage in the spent fuel pool for not less than 3 years; and
- cooled fuel removal from the reactor island.

Cooled fuel from the spent fuel pool is removed simultaneously with the preparation for refueling operations on the reactor. Core refueling covers replacement of spent fuel assemblies and spent burnable absorber rods by new ones and fuel assemblies and absorber rods shifting in the core. A refueling machine handles fuel assemblies and absorber rods under a protective water layer. During these operations, the refueling machine can handle only one fuel assembly or one burnable absorber rods bundle at a time; fuel assemblies may be shifted together with absorber rod bundles.

After removal of cooled fuel from the reactor island, withdrawal of spent fuel out of the reactor and rearrangement of fuel assemblies within the core, the fresh fuel prepared in the NFSF and delivered to the reactor island is loaded into the core. The spent fuel unloaded from the reactor is stored in spent fuel pool compact storage racks consisting of borated steel cells. Fuel assemblies in the racks are stored in a 300-mm triangular array. Fuel in the spent fuel pool is protected by water with a boric acid concentration of 16 g/kg.

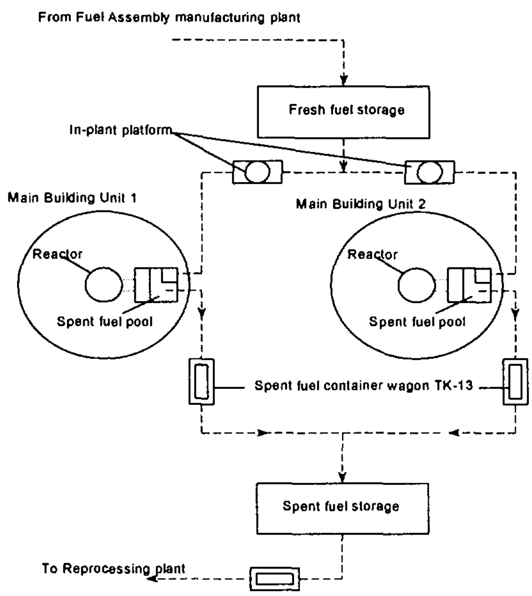


FIG. 2 General fuel movements at NPP

2.1.4 Radioactive Waste Management

2.1.4.1 Radioactive concentrated liquid media processing

The process for treating liquid radioactive effluents includes:

- collection and temporary storage; and
- processing and solidification in concrete.

Liquid radioactive waste is stored separately depending upon composition and activity level. Temporary storage is envisaged for waste accumulation to allow decay of short-lived nuclides prior to processing in the solidification plant. Medium-active solutions are stored for 3 months, while dry residues and low-active solutions are processed as accumulated. The system is designed to solidify liquid waste in proper containers for transportation and safe long-term storage in the processing and storage building. The following technologies are used for liquid waste processing, depending upon composition and activity;

- concentration of low-active dry residues to obtain dry salts; and
- concrete solidification of medium-active ion exchange resins, sludge, and salt concentrates.

The method of waste storage enables extraction of waste from storage cells to check packaging or for removal to regional disposal.

2.1.4.2 System of Solid Radioactive Waste Processing and Storage

The design of solid and solidified radwaste processing and storage is based on process needs, observing safety concerns during normal plant operation and during design basis accidents. With respect to nuclear safety, solid radwaste processing and storage applies a system of barriers to the potential propagation route for radioactive substances to the environment.

2.1.4.3 System of Solid Radwaste Collection, Sorting and Transportation

Solid radwaste is collected and sorted where it is generated with regard to level of radioactivity and processing methods and loaded into appropriate containers or disposable packing material. Containers and disposable packing material are brought to unattended rooms when repair work with waste generation is to be performed; in rooms that are periodically attended and rooms having permanent personnel, containers are installed in allocated positions. The number and types of containers are determined in advance by prediction of waste quantity, composition and radioactivity. Containers filled with waste are combined into lots for removal to the processing and storing building. Special areas are provided in the main building complex for this purpose.

Special vehicles remove the waste from the reactor island through the transportation corridors. Before leaving the transportation corridor, vehicles are subject to dose metering, washed and decontaminated, if necessary. Solid radwaste from the containment is removed through the equipment lock.

2.1.4.4 Solid Radioactive Waste Processing System

To reduce the volume of solid radwaste to be stored, it is processed by the following methods:

- grinding;
- incineration; and
- compacting.

The final product of processing is packed into standard drums. Metal waste (pipes, rolled stock, etc.), ventilation filters, and small items of equipment are ground. Solid radwaste of the 1st and 2nd radioactivity group is compacted. Combustible waste of the 1st and 2nd radioactivity groups is incinerated. Solid radwaste processing and storage facilities are in a separate building.

2.1.4.5 System of Solidified Liquid and Solid Radwaste Storage

A solid and solidified-liquid radioactive waste storage facility is provided at the site. Solid radwaste is stored in a specially equipped above-grade reinforced concrete storage building with walls and ceilings sufficiently thick to ensure mechanical strength and biological shielding. The waste storage method enables withdrawal of waste from storage facility cells for package inspection or for removal and transport to the regional repository. Solid radwaste and solidified products are placed in the storage facility in baskets containing 6 standard drums, each for subsequent waste withdrawal and transport to regional disposal. The filling coefficient of the cells in case of such storage is 0,34.

Special drums with highly radioactive waste are stored along guide lines in the storage cells for 50 years (the service life of the NPP), with the feasibility of their subsequent removal to the regional disposal. The filling factor is in this case 0,59.

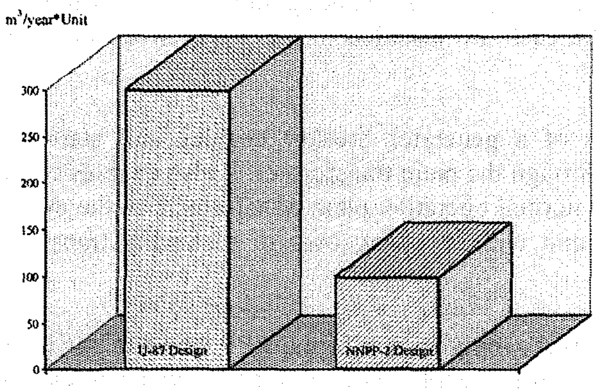
Nowadays, the concept of solid radwaste storage at the site for the whole NPP service life of 50 years is adopted. Nevertheless, the technology and storage structures enable withdrawal of drummed waste for further processing and subsequent transport to the regional disposal, as soon as it has been designed and constructed. The solid radwaste storage facility at the NPP site is built in stages with subsequent extension. The initial storage volume is designed for 10 years and is commissioned together with the NPP pilot unit; subsequent extensions are carried out, if required.

The storage facility is equipped with railroad and motor vehicle access points, systems of inspection and decontamination of transport, a radiation monitoring system, a system of explosive and fire hazard detection, and a heat and humidity detection system.

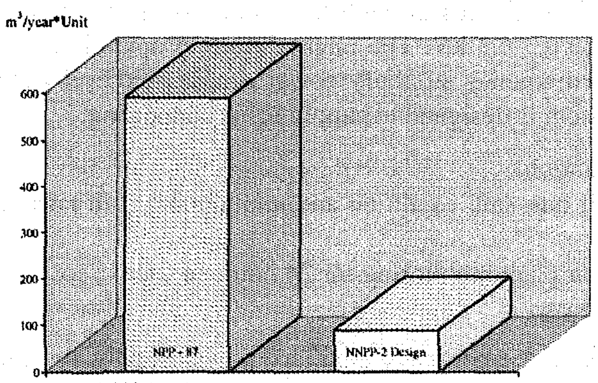
The generation of radioactive waste has been reduced considerably compared with that in existing plants. Figures 3 a and b illustrate the improvement by comparing the annual volumes of evaporator residue and solid radwaste from the NV NPP-2 with those for the U-87 design.

2.2 Main Engineering Solutions Pertaining to the Electrical Part

An overview of the electrical systems is shown in the single-line diagram of Figure 4. High-voltage (HV) switchgears for 500 and 220 kV are provided in the design for NPP power output to the grid.



a) Evaporation residues per year



b) Solid radioactive waste per year

FIG. 3 A comparison of radwaste volumes at NV NPP-2 and U-87

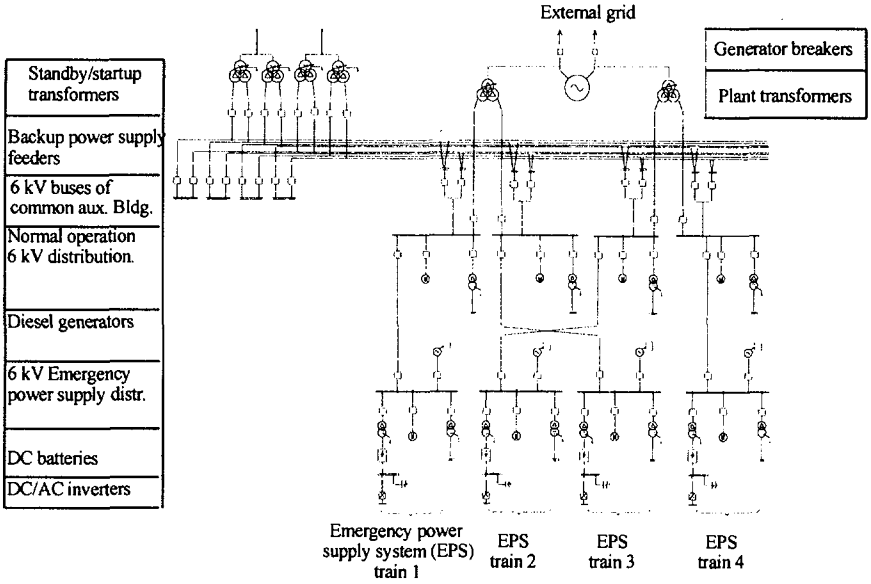


FIG. 4 Basic single-line diagram

An outdoor 500 kV switchgear is constructed in the 500 kV line; a metal-clad SF_6 gas-insulated switchgear (MCS) is provided for the 220 kV line instead of an outdoor switchgear due to the vicinity to the cooling tower and the reduction in land.

According to the design, each unit is equipped with one completely water-cooled turbine generator of 1100 MW. The generator and the main transformer are connected by shielded bus ducts with generator breaker units capable of breaking short-circuit currents. Two normal operation plant transformers of 63 MVA each are installed in the tap between the generator breaker and the main transformer.

The availability of a generator breaker enables unit start-up and shutdown with power supplied from the grid through the main transformer. It also enables continued application of auxiliary power from the grid via normal operation plant transformers in the event of the failure of a generator or process part of the unit without changeover to a standby transformer. This enhances the unit reliability considerably.

The auxiliary power supply system of the NPP contains power sources for normal operation, standby and emergency power supply. The auxiliary power supplies are divided into off-site and internal. The power grid with its power plants represents the off-site power supply. Off-site power may be supplied to NPP auxiliaries through the normal operation plant transformers or through the standby transformers 220/6,3-6,3 kV. The internal normal operation auxiliary power sources are the turbine generators and emergency auxiliary power sources – diesel generators and storage batteries. The auxiliary power system is designed to supply loads supporting:

- NPP normal operation;
- bringing the Unit into safe condition and maintaining it in normal and emergency conditions; and
- state of the reactor plant monitoring for 24 hours in case of loss of power and failure to start all the diesel generators.

Normal operation and emergency auxiliary power supply systems are envisaged at the Unit. Each Unit is equipped with two working transformers of 63 MVA each, feeding normal operation and emergency power supply system loads under normal plant operation. The design envisages four normal operating sections of 6 kV, in accordance with the number of RCPs. Power to each RCP is supplied from individual sections so the Unit operation is more stable in case of the loss of 6 kV sections. Loss of an RCP requires Unit power reduction or shutdown.

The emergency power supply system consists of two independent subsystems; each of the independent subsystems consists in turn of two trains with mutual redundancy. Installation of diesel-generators of 6300 kW and startup time of 1,5 s and three storage batteries is envisaged in each train as power supplies. Emergency power supply system switchgears are connected to normal operation loads ensuring serviceability of the main process equipment, which requires power in case of a loss of normal supply. Turbine oil pumps, shaft turning gear and rotor hydro-jack are examples in this category. Normal operation loads are connected to the switchgear of only one independent subsystem.

Emergency power supply system storage batteries are intended:

- to provide power to control and automation and relay protection devices of emergency power supply system elements and emergency lighting of loads of this channel of the emergency power supply system through one storage battery. Battery discharge time is 2 hours.
- to provide power to I&C hardware through the second battery. Duration of battery discharge is also 2 hours;
- to provide power to reactor control and monitoring devices in case of total loss of AC power through the third battery. Battery discharge time is 24 hours.

The electrical equipment of the emergency power supply system is located in the standby diesel power station (SDPS). The SDPS for each subsystem are in two separate buildings. Each building consists of two physically separated cells and each cell houses the equipment of one train. The trains are separated by structures with a fire resistance of at least 1.5 hours.

2.3 Computer-aided Process Control System (Instrumentation and Control[I&C])

Each NPP unit is envisaged as having independent I&C systems. From the unit point of view, the I&C is intended to maintain design basis limits set by process parameter values and characteristics of state of process elements and systems designed for normal operation conditions, operational occurrences, emergency situations and accidents. In developing the design, consideration has been given to comments by the IAEA on the automation of power plant units with RU-320 and their operational experience. The I&C hardware was developed in parallel with the plant design. The I&C concept is based on the following fundamentals:

- I&C is mainly implemented using modern digital hardware proved by positive operating experience at fossil fuel or nuclear power plants;
- the design adopts a centralized system of unit equipment control, which envisages unit automatic control from the main control room;
- a standby control room is provided at the unit to ensure unit shutdown, cooling down and reactor plant sub-criticality monitoring;
- a protection system train is envisaged for each of the four safety system trains;
- two independent sets of scram (emergency protection) are envisaged for emergency shutdown; and
- a unit top level system is envisaged to combine all the automation sub-systems into a unified system, which implements unit common tasks as well.

The structural I&C diagram (cf. Figure 5) of the unit represents the main components of the system/protection systems, low level automation and unit top level control and monitoring systems, as well as digital trains of data exchange and remote control.

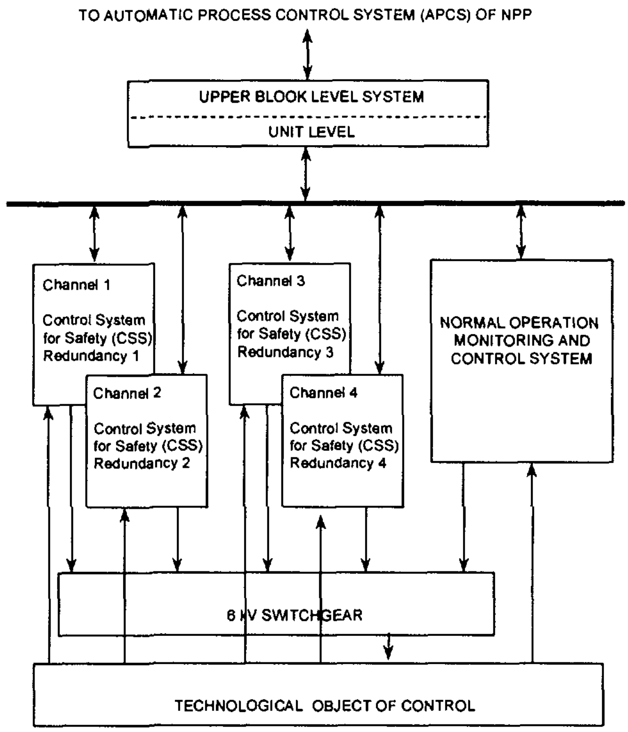


FIG. 5 Structural diagram of the automatic process control system (APCS)

2.4 Civil Solutions, Main Building Complex General Layout

The main building complex layout is of monoblock type; the main and ancillary equipment of the reactor and turbine plants of each unit are located in separate compartments. The main building complex, which is a standardized module, combines the reactor and turbine islands and a sanitary-social building.

2.4.1 Reactor island

The reactor island is a number of adjacent compartments containing the reactor plant and systems supporting normal operation and those ensuring emergency shutdown of the unit. The reactor island consists of the containment building and process safety systems, buildings of the protection safety systems, monitoring and control systems building, normal operation process and special water treatment systems building. All buildings are constructed on separate foundation slabs. To achieve independent response under static and special dynamic effects, gaps between buildings are assumed to be 400 mm.

2.4.2 Containment and process safety systems building

The containment building consists of a cylindrical containment (the leaktight part) and two adjacent buildings on opposite sides (non-leaktight parts), on the same foundation slab. The containment houses safety-related systems, and therefore is designed to withstand external effects; it is in Safety Class 1.

The building plan dimensions are 73,2 x 43,2 m. The building height is 89,4 m. The containment is an accident confinement system and consists of two shells: an inner leaktight containment and an outer one protecting it from external effects. The reactor plant spent fuel pool, ancillary process systems working at primary parameters, ventilation systems and equipment providing for fuel handling and repair operations are located within the containment.

The reactor occupies the central part of the containment. The spent fuel pool and internals inspection wells, two Main Circulation Circuit (MCC) compartments housing steam generators, RCPs, Main Circulation Pipelines, pressurizer, bubbler and quick boron supply system tanks are located on both sides of the reactor pit. The Reactor Pressure Vessel inspection machine and the core catcher for beyond design basis accidents are located on the slab of the leaktight containment under the reactor.

Maintenance rooms for pulse valves actuators, purification systems rooms (SWT-1, SWT-2), control chambers for valves in contaminated pipelines and ventilation plants are located around the reactor pit. The ECCS tanks of the 1st and 2nd stages are located at the maintenance elevation. The contaminated equipment-washing unit is located next to the spent fuel pool from the side of equipment lock. For execution of transportation operations through the confinement boundary, the containment is equipped with air locks designed to ensure tightness under design basis accidents and external design impacts.

Personnel access into the containment is through the main air lock to the maintenance elevation from the controlled access area of the auxiliary process systems building. An emergency air lock is provided at the lower elevation to ensure the emergency exit of personnel from the containment. All fuel and equipment handling operations are carried out through an equipment lock at the maintenance elevation and ledge located outside the containment.

The containment building basement is the space between the leaktight containment slab and the building foundation slab. The structural features of the cooling down pumps define the height of the basement. The containment basement houses primary circuit and spent fuel pool cooling systems, primary circuit I&C and radiation monitoring assemblies (immediately under the core), the intermediate circuit system, the steam generators emergency cooling down and blow-down systems, I&C and radiation monitoring and I&C rooms of the systems located in the basement.

Filters of the containment overpressure release system, emergency inventory tanks for service water and exhaust ventilation center are in the building adjacent to the containment, on the side opposite the turbine building. The building in question is in the controlled access area, together with the basement.

The adjacent building from the side of the turbine hall, except for the floor of the rooms for the steam generators emergency cooling down and blow-down systems, belongs to the free access area. It houses the safety-related part of the main steam and feedwater pipelines, plenum ventilation center, cable corridors, I&C and radiation monitoring hardware premises.

Systems in the non-leaktight part of containment building (except for the containment overpressure release system) are divided into two independent channels with 100 per cent redundancy in each channel. Walls separate rooms of different channels. Areas of one channel (including rooms, corridors, and stairs) are completely isolated from the areas of another channel and no communication lines (pipelines, cable and ventilation ducts) of another channel pass through it. A vestibule separates corridors of the area of one channel from the common corridor to provide fire protection. Common corridor and emergency exits into the outdoor area provide for the evacuation of personnel.

2.4.2.1 Main Solutions Pertaining to Containment Civil Structures

Outer Containment

The outer containment is a cylinder with a spherical dome of monolithic reinforced concrete. The wall thickness of the cylindrical part and the dome is 600 mm and the inner diameter, 50,8 m. The outer containment will absorb loads of external impacts arising from hurricanes, tornadoes, external shock wave, and aircraft crashes. An inter-containment gap of 2,2 m enables maintenance of the inner containment pre-stressing system and accessibility for visual inspection of surfaces. This gap enables controlled collection of gas-air media leaks. The inner surface of the outer containment will be provided with polymeric coating ensuring the required tightness of outer containment. The heat exchangers of the passive heat removal system are located on the outer containment and the installation is arranged such that the heat exchangers of only one steam generator could be damaged in case of an aircraft crash.

Inner Containment

The inner containment is cylindrical pre-stressed monolithic reinforced concrete covered with a semi-spherical dome. The equipment layout in the leaktight space defines the basic dimensions of the containment, which are:

- cylinder and dome inner diameter, 44 m;
- cylindrical part height, 40 m;
- wall and dome thickness is 1,2 m, based on structural requirements and biological shielding.

A steel lining ensures leaktightness of the inner containment; basic ambient design parameters are:

- During design basis accidents:
  - emergency design overpressure 0,4 MPa
  - emergency design temperature + 150 ^{0}C.
- During beyond design basis accidents:
  - emergency overpressure is 0,6 MPa
  - emergency temperature is + 200 ^{0}C

2.4.3 Protection Safety Systems Building and Monitoring and Control Systems Building

The electrical building is located between the reactor and turbine buildings. There are two protection safety system buildings – a standard layout solution – that are in Safety Class 1. They are on opposite sides of the monitoring and control systems building to prevent simultaneous destruction in the event of an aircraft crash. Each of the buildings houses electrical equipment of two independent channels of active safety systems, including independent systems of plenum and exhaust ventilation.

The Safety Class 1 building of the monitoring and control systems electrical equipment is between the protection safety system buildings. It houses the main control room, control and protection system (SUZ) and information computer system panel. These buildings are prefabricated-monolithic reinforced concrete.

2.4.4 Building Normal Operation Process Systems of Special Water Treatment System (SWT)

The dimensions of the process systems (PB) and special water treatment (SWT) building are 45 x 66 m (Safety Class II). It is a process support to the containment building and adjoins the latter by the long wall from the side of the transportation ledge. The building houses auxiliary primary systems (PB) including Unit special water treatment. The building is monolithic reinforced concrete. Special sewage, borated and service water drainage collection systems are at lower elevations. Laboratories are designed at elevation 7.200. I&C, control and protection system premises, plenum ventilation center, exhaust ventilation center and stack are at higher elevations. An access air lock to the containment leaktight space is at elevation 31.800; entrance into the building is through the sanitary-social service building.

2.4.5 Turbine Hall and Deaerator Rack Including Oil Handling Building

The layout of the turbine hall, deaerator rack and oil handling building is determined mainly by the type of turbine plant, with condensers in the basement, three low-pressure cylinders and a water-cooled generator, as well as upon the layout of auxiliary systems and equipment. The turbine hall and deaerator rack, including the oil handling building are in Safety category II. The turbine hall dimensions are 36 x 102 m, with a height of 40,8 m; the turbine hall is a steel structure. Placing the turbine hall with its end facing towards the reactor enables the best use of the layout volume for the arrangement of equipment and to locate turbine steam exhausts as close to the reactor building as possible.

The turbine plant foundation is provided with vibration dampers so that the transfer of dynamic and vibrational effects on civil structures of platforms and ceilings, resting on the turbine plant foundation pillars and the lower support plate, are practically avoided.

2.4.6 Standby Diesel Power Station (SDPS)

The SDPS (Safety Class 1) is intended to supply power to safety system loads under loss of off-site power to the plant. The equipment of each safety system train is located in an isolated SDPS cell. Four cells with dimensions of 30 x 33 m and 16 m high are provided for each unit. The cells are arranged in pairs, with their short sides contacting each other, and they are located in two separate buildings, to prevent their simultaneous destruction in case of an aircraft crash. Each cell houses:

- the standby diesel power plant itself;
- intermediate circuit and vital consumer service water supply pump house.

The building is a monolithic reinforced concrete structure.

2.4.7 Sheltered Center of Emergency Actions Management at NPP (SEAMC)

The SEAMC is designed as a sheltered two-story underground structure belonging to Safety Class 1 and seismic category 1. Its plan dimensions are 24 x 54 m. The building houses:

- standby control rooms of Units 1 and 2 (SCR1 and SCR2);
- NPP central control room (CCR); and
- shelter for 900 persons.

The SCR is intended to shut down the reactor unit in the event of an MCR failure. From the SCR, it is possible to monitor and initiate the safety systems and remove heat from reactor plant. The status of the reactor plant and spent fuel pool can be monitored from the SCR under all operating conditions, including station blackout. SCR availability is ensured for 24 hours in case of loss of power by means of separate storage batteries. A system for recording crucial parameters ("black box"), ensuring information preservation in case of an accident at the plant unit, is located at the SCR. The NPP CCR is intended for control of the power generation process and NPP-common facilities, and radiation monitoring at the site and at buffer area. (ARSMS).

3 SAFETY CONCEPT, SAFETY EVALUATION RESULTS, RESEARCH PERFORMED

3.1 General Philosophy and Safety Concept

Unit 1 of the NVNPP-2 is designed as an NPP with the enhanced safety VVER-1000 reactor of a new generation. The safety concept has been elaborated in an evolutionary approach based on thorough analysis of operating experience and design solutions of NPP units with V-320. The analyses that incorporated an evaluation of advantages and weak points of these operating NPPs, were carried

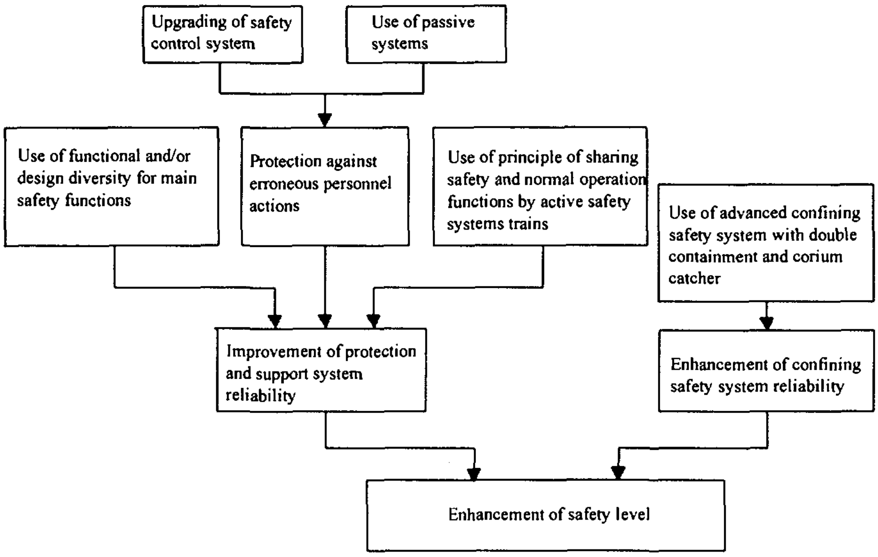


FIG. 6 Enhancement of the safety level in NPP with V-392

out within the framework of an effort to enhance the safety of these units. This effort was implemented in line with domestic and international programs with participation of leading Russian (AEP, OKB Gidropress, RSC "Kurchatov Institute" etc.) and foreign (EdF, GRS, Siemens) companies as well as the IAEA. The safety enhancement concept of the NPP with V-320 was developed using deterministic and probabilistic safety analyses. Based on the analyses, conclusions were drawn that the safety level of the operating NP units with V-320 reactors in major part complies with the safety level of other operating NPP units equipped with PWR reactors. Implementation of measures suggested for the NPP with V-320 safety enhancement concept would ensure compliance of these units with most of the requirements of the regulatory documents. However, the new qualitative level of safety can be achieved by elaboration of new design solutions ensuring resolution or reduction of the weak points revealed at NPP with V-320. It is necessary to point out that recommendations for the NPPs with V-320 were thoroughly considered for NPP with V-392 and were subject to special analysis focused upon two main points.

The safety concept of the NVNPP-2 is based on application to the maximum possible extent of the engineering principles of defense-in-depth concepts described in IAEA documents and on the input of data from operating NPPs to the V-320 safety analysis. The NVNPP-2 concept for safety enhancement that is schematically summarized in Figure 6, covers the following basic principles:

1. Application of functional and/or structural diversity in the systems performing each individual safety function. Mutually redundant active and passive systems are used in the design. Diversity ensures sufficient depth of protection against common cause failures and reduces safety systems unavailability indices by several decades.

2. Use of active safety system trains (emergency cooling down and ECCS) for execution of normal operational functions. At the same time, most of those components are in the states similar to the state they are in during the execution of assigned safety functions in the course of accidents. Such a mode of functioning of those systems makes it possible to enhance availability and ensure additional protection against common cause failures. For continuously functioning components, latent failures, the main cause of unavailability of the systems in the "waiting" mode of operation, are avoided.

3. The following design solutions provide protection against human error:

   •increased automatic system control (prevention of personnel interference) in case of a number of design basis accidents and, in particular, in case of primary to secondary leaks;

   •introduction of passive systems not requiring personnel actions for actuation.

4. Application of a full pressure double containment equipped with a hydrogen removal system, containment air discharge and purification (filter) system and core melt catcher, ensuring that the system does not exceed the established limiting release under beyond design basis accidents with severe core damage.

An overview of safety functions together with a list of systems capable of fulfilling each of them is provided in Table 1.
One should note that most design solutions mentioned above were developed based upon results of PSAs performed for operating NPPs with V-320 reactors and as a part of the design of NVNPP-2 Unit 1.

3.2 Safety Systems Technological Bases

The greatest modifications to the V-320 concern the safety systems. Probabilistic analysis revealed the dominant safety functions and "weak" points of the existing design and concluded that the following solutions are necessary:

- the main critical safety functions shall be fulfilled by diverse systems, both active and passive;

- in terms of functional reliability and taking into consideration peculiarities in the maintenance procedures, the best active safety system part structure is 4 x 100 per cent, at the same time the structure of 2 x 200 per cent yields the best results for support and protection safety systems.

Table 1. Structure and composition of the safety systems

Active part of Safety systems	**Structure**			
	Sub-system 1		Sub-system 2	
	Train 1	Train 2	Train 3	Train 4
Protection and Isolation systems				
- Multi-functional system of emergency heat removal from the reactor core	100 %	100 %	100 %	100 %
- Emergency heat removal system through the secondary circuit	100 %	100 %	100 %	100 %
Power supply systems				
- *Intermediate circuit and service water*				
▪ under closed conditions	100 %	100 %	100 %	100 %
▪ under open conditions	200 %		200 %	
- Ventilation	100 %	100 %	100 %	100 %
- Valves fail-safe supply				
• Redundant valves of the active part	100 %	100 %	100 %	100 %
• Isolation valves and passive part valves	200 %		200 %	
- *Pumps fail-safe power supply*	100 %	100 %	100 %	100 %
- APCS fail-safe power supply				
• Control safety system of the active part	100 %	100 %	100 %	100 %
• Control safety system of the passive part	200 %		200 %	
Control safety systems				
- Scram (emergency protection) system	200 %			
- Sensors	400 %		400 %	
- Logic part	100 %	100 %	100 %	100 %

Passive part of Safety systems	***Structure***
- Fast boron injection system	4x25 %
- Hydraulic accumulators of stage 1	4x50 %
- Hydraulic accumulators of stage 2	4x33 %
- Passive heat removal system	4x33 %
- Medium let-down and purification system from the containment	100 %
- Core catcher	100 %

The greatest reliability in main safety functions execution by active systems is achieved when the so-called "combination" principle is implemented, when active safety mechanisms perform normal operational functions and upon the appearance of accident indications, perform safety functions, either without change-over or with a minimum number of change-overs. Compared to traditional "waiting" safety systems, this solution improves reliability considerably (5-6 times). This is due to low sensitivity to latent failures (failures not evident in the "waiting" state of mechanism, which is out of operation) and reduces significantly the equipment (valves, cables, instruments, automatic devices, etc.). For example, in the NPP-392 design, there are four pumps in the active emergency cooling system through the primary circuit, whereas the same functions in the traditional design are carried out by 12 pump assemblies (The comparison refers to identical structures of 4 x 100 per cent).

The active part of the safety systems includes:

- scheduled and emergency, primary circuit and spent fuel pool cooling down ;
- steam generator emergency cooling down and blow-down;
- intermediate circuit system;
- service water supply system;
- ventilation and air conditioning support systems.

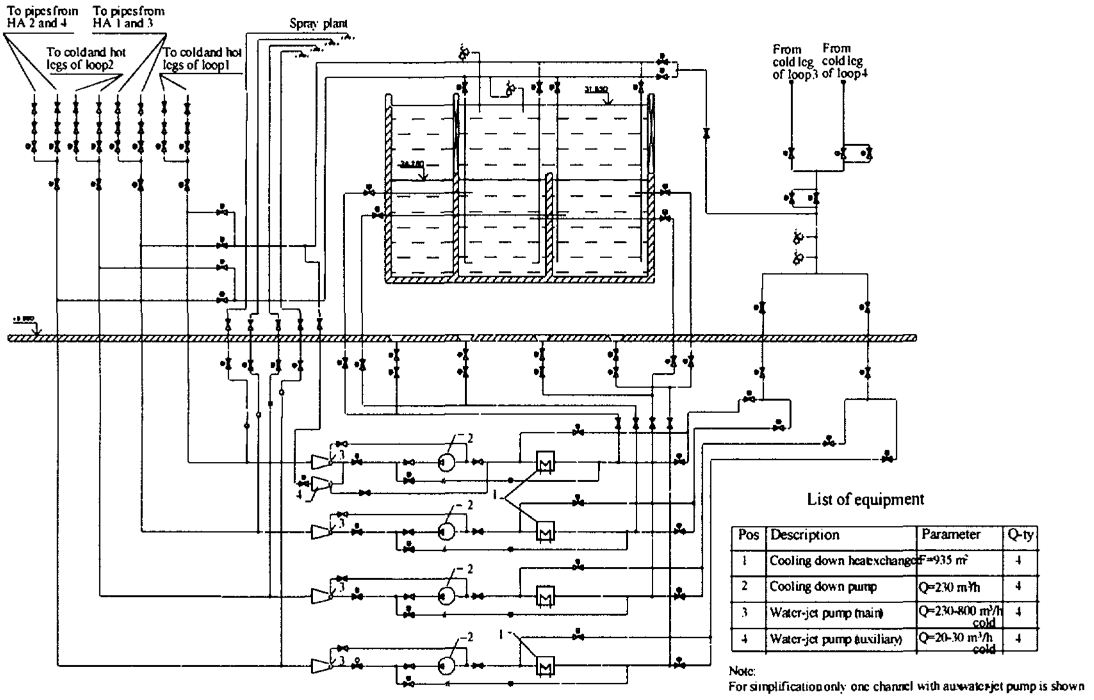
The passive part of safety systems incorporates:

- the passive heat removal system (PHRS);
- the 1st and 2nd stage accumulator system
- the quick boron injection system; and
- a system for maintaining under-pressure in the inter-containment gap.

The active emergency cooling system (cf. Figure 7) through the primary circuit comprises four groups of cool-down circuits with a combination of centrifugal and jet pumps in each. Under normal operation, these circuits are used for spent fuel pool cooling; under accident conditions, the system executes circuit emergency makeup in the pressure range of 8,0-0,1 MPa (80 - 1 bar), as well as a spray function.

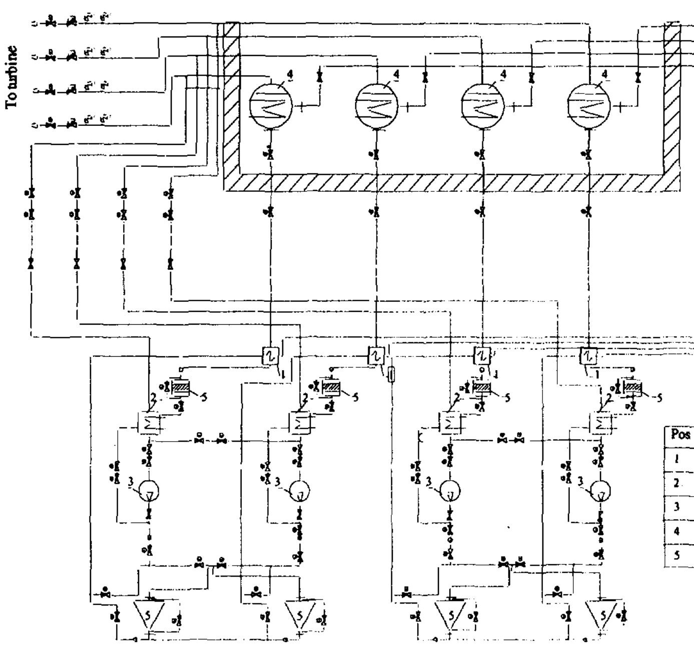
The active emergency cooling system (cf. Figure 8) through the secondary circuit is made up of four closed circuits of secondary coolant cooling - one per each steam generator. Under normal operation these circuits are used for steam generator boiler water blow-down cooling.

The passive quick boron supply system facilitates reactor shutdown in case of control rod system failure (conditions without scram). The system has four subsystems; each subsystem has a tank



Pos	Description	Parameter	Q-ty
1	Cooling down heatexchanger	F=935 m²	4
2	Cooling down pump	Q=230 m³/h	4
3	Water-jet pump (main)	Q=230-800 m³/h cold	4
4	Water-jet pump (auxiliary)	Q=20-30 m³/h cold	4

FIG. 7 Emergency cooling system for primary circuit and spent fuel pool



List of equipment

Pos	Description	Parameter	Q-ty
1	Regenerative heat exchanger	$S=55\ m^2$	4
2	Emergency cooling downheat exch.	N=64 MW	4
3	Emergency SG cooling downpump	$Q=150\ m^3/h$ P=0,8 MPa	4
4	Steam generator		4
5	High temperature filter		4

FIG. 8 SG blow-down and emergency cooling system

with concentrated boron solution connected to an RCP intake, discharged by pipelines, i.e., it represents a RCP bypass. If the solid absorber system fails in an event requiring the reactor to trip, the tank is connected to the loop. In this case, the boron solution goes to the primary circuit via the RCP intake. Inventory and concentration of the boron solution is selected to ensure compliance with safety criteria, in case of design-initiating events and where the reactor trip system fails to actuate. The system also functions where RCP power is lost, since the boron solution from the tanks is forced into the reactor due to RCP coast-down.

The passive system for heat removal from the primary circuit (PHRS) (cf. Figure 9) consists of four groups (corresponding to the number of SGs) of closed natural circulation circuits. In the ribbed tubular heat exchangers of these circuits, steam extracted from the steam generator condenses, and the condensate flows by gravity down the letdown pipelines into the steam generator boiler water volume. The PHRS heat exchangers are cooled by atmospheric air coming to the heat-exchanging surface near the draft air-duct outlet, through special direct action control gates that maintain steam generator pressure at a higher than nominal level. This solution makes it possible to prevent heat losses under normal operation and at the same time, keep the PHRS circuits warm. In case of station blackout, this solution prevents adverse primary circuit dynamics (coolant cooling, pressurizer level walk-away etc.). The cooling circuit is fully integrated and the operational duration of the PHRS is practically unlimited.

The passive system for reactor flooding (cf. Figure 10) during primary leaks comprises two groups of accumulators:

- 1st stage accumulators - 4 tanks of 50 m^3, each with a gaseous nitrogen cushion pressurized to 6,0 MPa (60 bar) and connected in pairs by pipelines equipped with check valves to the upper and lower reactor plena, through special nozzles in the reactor pressure vessel;
- 2nd stage accumulators are 8 tanks of 120 m^3 each, connected to the primary cold leg through check valves and to the primary hot leg through special spring-type valves. These valves are kept closed by primary circuit media pressure; when the pressure drops below 1,5 MPa (15 bar), the spring opens the valves. Such a connection configuration and valve design ensures continuity of hydrostatic pressure irrespective of primary pressure variation. Installation of throttling devices ensures a step-wise limitation of water drainage flow-rate with a decrease in

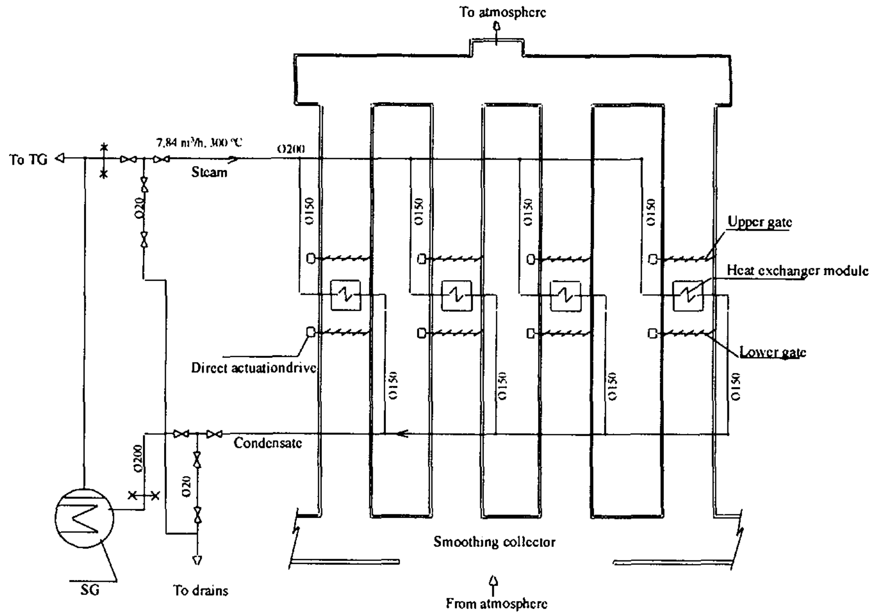
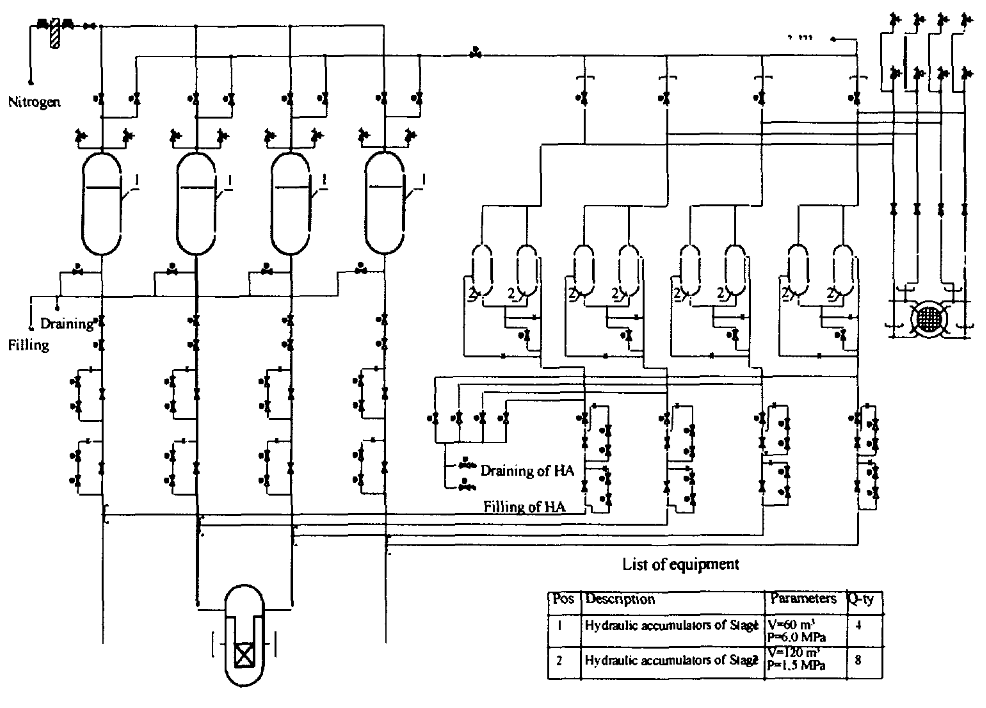


FIG. 9 Passive Heat Removal system



List of equipment

Pos	Description	Parameters	Q-ty
1	Hydraulic accumulators of Stage 1	V=60 m³ P=6.0 MPa	4
2	Hydraulic accumulators of Stage 2	V=120 m³ P=1.5 MPa	8

FIG. 10 The Hydraulic Accumulators system

the 2nd stage accumulator level. Stepwise limitation of the drainage flow-rate follows the law of decay power decrease with the necessary margin. The water inventory in the 2nd stage accumulators enables reactor cooling for 24 hours in case of leaks, even under blackout conditions, with all active mechanisms inoperable.

If AC power is not recovered after 24 hours, further core cooling is possible by PHRS. This system maintains low temperature in the steam generator boiler water and ensures the condensation of primary coolant steam inside the steam generator heat exchanging tubes, under conditions with the coolant level in the reactor pressure vessel below the hot nozzles. The evaporated coolant condenses, and returns, gravity-driven, along the loop pipe back into the reactor pressure vessel to cool the fuel.

3.3 Accident Product Confining System

The design basis sequence of loss of primary coolant accidents is overcome by the actuation of process systems (accumulators, emergency core cooling systems, containment spray system, containment isolation, etc.). Design values for the containment ambient under design basis accidents are:

$P = 0{,}5$ MPa; $T = 150\ ^0C$.

Under the beyond design basis accidents considered in the design when fuel cooling is impaired, the limit accident performance values are established as follows:

$P = 0{,}7$ MPa; $T = 200\ ^0C$.

The containment protection is provided by independent systems in the same way as for other critical safety functions:

- spray system (conventional);
- passive heat removal system; and
- an ultimate over-pressure relief device equipped with a high efficiency filtering plant.

The filtered discharge system is not actuated earlier than 12 hours after an accident initiation. To limit considerably the release of fission products beyond the containment, a permanent under-pressure is maintained in the inter-containment gap. This safety function, one of the most important, is fulfilled by two systems:

- an exhaust ventilation system equipped with a filtering plant - with suction from the inter-containment gap and outlet into the stack;
- a passive system of suction from the inter-containment gap. This system consists of lines connecting the inter-containment gap with the PHRS exhaust ducts, which are in a hot state. This solution enables permanent removal and purification of inner containment leaks irrespective of NPP power availability and operator actions. According to estimations, an under-pressure is maintained at any point in the inter-containment gap with inner containment leaks up to 2,8 per cent of volume per day (the design leak value is 0,3 per cent of the volume per day).

A hydrogen suppression system is designed to prevent hydrogen burning or explosion in the containment. The system comprises passive catalytic hydrogen igniters of efficient high porosity cellular materials. An element of 1,5 x 0,3 x 1,4 m dimensions oxidizes 30 l/h of hydrogen; its volumetric concentration is 4 per cent. Fifty igniters prevent explosive concentrations of hydrogen under beyond design basis accidents, when 100 per cent of Zr reacts with steam and hydrogen generated from other sources.

Table 2. Contributions to core damage frequency by different initiating event categories

Initiating event category	Contribution to core damage frequency (CDF)	
	absolute, 1/year	relative, %
1. Internal event during reactor power operation	2.6 E-8	48
2. Internal event at shutdown conditions	2.2 E-8	40
3. Seismic effects	5.9 E-9	11
4. Fires in the NPP premises	4.0 E-10	1
All event categories	5.4 E-8	100

For a beyond design basis accident leading to core damage and melting through the reactor pressure vessel, a core catcher is designed to make it possible to retain the corium within the compartments with a refractory coating. It is then cooled by passive means, with the help of accumulated water at the initial stage and by return of the condensate generated in PHRS heat exchangers.

3.4 Probabilistic Safety Assessment (PSA)

NVNPP-2 Unit 1 design includes performance of probabilistic safety analyses comprising:

- PSA-1 for internal initiating events;
- PSA-2 for internal initiating events;
- PSA for fires within the NPP premises; and
- PSA for seismic effects.

The main goals of PSA are to develop probabilistic models, assess probabilistic safety indices using those models and evaluate the safety level achieved based on the results. Core damage frequency and limit frequencies for large release of radioactive material to the environment were used as indices. Qualitative and quantitative evaluations of safety were performed. For the quantitative evaluation, the values established in OPB-88 (items 1.2.17 and 4.2.2) of 1.0 E-5 per reactor year for core damage frequency and 1.0 E-7 per reactor year for limit release frequency were used as goals.

The qualitative evaluation based on the PSA results was performed by assessing NVNPP-2 design compliance with the main engineering or deterministic safety principles, established in domestic regulatory documents (OPB-88 etc.) and IAEA materials (INSAG-3), which ensure the required level of defense-in-depth when met. Functional and structural diversity of safety systems provide deep protection against common-cause failures, and application of passive systems and active systems actuation without personnel interference yield deep protection against human error. Table 2 presents the estimated contributions to the core damage frequency (CDF) of different initiating events, including internal events probable during reactor power operation and shutdown, internal fires and seismic effects. The calculated total limit environmental release frequency over all the internal initiating events is about 4.77×10^{-8} 1/year.

Internal initiating events are the main contributors to the CDF (~88 per cent). The next contributor (~11 per cent) is seismic impact. Contribution from fires within the NPP rooms is relatively small (~ 1 per cent). So, the engineering solutions used in NVNPP-2 Unit 1 design enable attainment of a qualitatively new safety level, compared to its predecessors. The design meets all requirements of the defense-in-depth concept and target probabilistic safety indices established in regulatory documents.

Table 3 provides more detail with respect to different internal initiating events and Table 4 presents a comparison between the results for the NV NPP-2 and those for Unit 4 of Balakovo NPP.

4 EXPERIMENTAL AND ANALYTIC VALIDATION OF PASSIVE HEAT REMOVAL SYSTEM

Experiments and analyses were conducted to define PHRS operating parameters over the whole range of specified conditions with the aim of justifying design solutions for the passive heat removal system.

4.1 PHRS Experimental Validation

4.1.1 PHRS Experimental Study at Full-Scale Test Rig

Full-scale sections of the PHRS air heat exchanger-condenser with a design power of 5 MW were subject to experimental tests at OKB Gidropress test rig. The experiments were conducted with natural circulation steam condensate and air paths modeled; the environment air temperature was from - 19 ^{0}C up to + 30 ^{0}C and at steam condensing duct pressure from 0,5 MPa to 6,4 MPa. In the course of these experiments, the heat exchanger section thermal power was defined within the range of parameters specified during both operations with open gates and "waiting" conditions.

Table 3 Contribution to core damage frequency from different internal initiating events

Initiating event (IE)	Frequency, 1/year	Contribution to core damage frequency	
		absolute, 1/year	relative, %
1. LOCA into containment volume			
1.1. SLOCA	3,20 E-03	1,26 E-09	~2,6
1.2. MLOCA	1,00 E-03	3,64 E-10	<1
1.3. LLOCA	3,20 E-04	6,79 E-10	~1,4
2. LOCA to second circuit	1,00 E-03	1,26 E-09	~2,6
3.Reactor shutdown	1,00 E+00	7,38 E-09	~15
4. Loss of normal heat removal through secondary circuit	1,00 E-01	7,38 E-09	~15
5. Loss of offsite power	1,00 E-01	7,91 E-09	~16
6. Steam line break in part isolated from SG	1,00 E-03	2,67 E-11	<1
7. Steam line break in part not isolated from SG	4,00 E-04	1,29 E-10	<1
8. Loss of heat removal during shut-down and open pressure vessel	3,50 E-05	1,07 E-08	~22
9. Loss of offsite power during shutdown and open pressure vessel	3,70 E-03	1,12 E-08	~23
Total IE		4,77 E-08	100

4.1.2 PHRS Experimental Investigation at TDU-1 at IPE of Academy of Science of Belorussia

The experiments were performed using a double-loop and three-circuit installation of 1,0 MW. The installation had two identical loops with simulators of steam generators and air heat exchangers-condensers of 400 kW each. The availability of two loops made it possible to simulate the operation of two PHRS-SG circuits in parallel, under the conditions of heat exchangers non-equilibrium loading.

The experiments were performed with air temperature variations from + 5 ^{0}C to + 31 ^{0}C and steam pressure variation in the range from 0,6 MPa to 5,4 Mpa. The experiments demonstrated the stable operation of heat exchangers within the range of the parameters specified. No heat exchanging tube walls and condensate temperature variation was detected.

4.1.3 Experiments at "SPOT-2" Facility at IPE of Academy of Science of Belorussia

This facility simulates the main circulation circuit with VVER-1000 reactor and PHRS circuit in the power scale of 1:5500, maintaining the hydraulic similarity of full-scale and model circuits and actual differences in equipment location elevations. The experiments confirm the possibility of long-term

cooling down by the passive heat removal system for the case of an accident with main circulation pipeline rupture and loss of power. Under the conditions of experiment, the PHRS heat removal capacity was 97 per cent of the decay power.

4.1.4 Experiments for determination of wind effect upon RHRS capacity

When wind flows around the NPP main building, depending upon wind direction and velocity, a non-uniform pressure field is created along the containment circumference. It may cause air flow reversal in one or in a group of PHRS exhaust shafts. Atomenergoproject has suggested using common circular collector at the exhaust shaft inlets and one common collector with a deflector at the outlet to protect PHRS operation against wind effects.

An NPP main building simulator was developed and made in the scale of 1:80 to investigate the wind effect upon PHRS operation. This simulator was used to conduct experiments at Ts-22 NITs

Table 4. Comparison of CDF contributors for NV NPP-2 and operating unit 4 of Balakovo NPP

Initiating event (IE)	Frequency of IE	Contribution to core damage frequency			
		Unit 1 of NV NPP-2		Unit 4 of Balakovo	
		absolute, 1/y	relative, %	absolute, 1/y	relative, %
1. LOCA into containment volume					
1.1. SLOCA	3,20 E-03	1,26 E-09	~4,9	3,40 E-07	~0,8
1.2. MLOCA	1,00 E-03	3,64 E-10	~1,4	8,30 E-08	~0,2
1.3. LLOCA	3,20 E-04	6,79 E-10	~2,6	5,40 E-08	~0,1
2. Interfacing LOCA	1,00 E-03	1,26 E-09	~4,9	1,10 E-06	~2,6
3. Reactor shutdown	1,00 E+0	7,38 E-09	~28,6	1,65 E-06	~3,9
4. Loss of normal heat removal through the secondary circuit	1,00 E-01	7,38 E-09	~28,6	6,50 E-07	~1,5
5. Loss of offsite power	1,00 E-01	7,91 E-09	~30,6	3,54 E-05	~82,9
6. Steam line break in part isolated from SG	1,00 E-03	2,67 E-11	~0,1	3,40 E-06	~8,0
7. Steam line break in part not isolated from SG	4,00 E-04	1,29 E-10	~0,5	1,00 E-10	~0
Total IE		2,58 E-08	100	4,27 E-05	100

TsIAM in TsAGI. Aerodynamic tests on the main building simulator were carried out for the wind velocity range of 0 to 90 m/s (from calm to hurricane) with wind direction from 0 up to 360 degrees, in respect to the main building axis. The experiments performed on the NPP 1:80 simulator proved the correctness of design decisions to join the PHRS channels by a common inlet collector and common outlet collector, equipped with deflectors. No circulation reversal in the PHRS exhaust shafts was observed.

4.2 PHRS Parameters Analyses

Atomenergoproject carried out calculations to justify PHRS design parameters over the whole range of set conditions and under all PHRS modes of operation, including those under extreme temperature and wind effects. A number of computer codes were developed at Atomenergoproject, which facilitate calculations of PHRS steady-state and dynamic parameters, in particular, "RADUGA", "SPOT-KT", "GAMBIT", "STVORKA" codes. Using the above codes, calculations substantiating PHRS-related design solutions were performed and the effect of PHRS upon reactor plant operation was calculated for different accident conditions. Analyses of dynamic processes during PHRS operation with passive governor of air heat exchangers heat removing capacity were carried out. Analysis proving the optimum control device diagram was performed.

4.3 Passive Filtering System

4.3.1 Purpose of passive filtering system

The passive filtering system (PFS) is intended to control removal of the steam-gas mixture from the inter-containment gap under beyond design basis accidents with total loss of power. Prior to release of the steam-gas mixture into the atmosphere, it must be purified at the filters from radioactive substances entrained through the containment systems and elements entering through untightness into the inter-containment gap. The passive filtering system shall be operable under beyond the design basis accidents with loss of all AC power, both with tight primary circuit and with primary or secondary leaks.

4.3.1.1 System design

The passive filtering system can be divided into the following functional sections:

- inter-containment gap;
- stack;
- filtering device;
- air heater; and
- valves.

In a loss of coolant accident, the steam-gas mixture under high pressure and temperature, would appear in the containment. Through containment untightness (steel lining micro-cracks,. penetration untightness, cracks in concrete) the steam-gas mixture enters the inter-containment gap. Air heating in the stack at the expense of the pipe contact with hot air coming into the outlet collector from the PHRS heat exchangers, results in leveling the pressure differential in the passive filtering system. Due to this pressure differential the steam-gas mixture passes through the stack and is discharged into the atmosphere. Passing through the stack the mixture is heated, dehumidified and purified before release.

4.3.1.2 Mode of Operation

There are two modes of operation of the passive filtering system:

- waiting mode; and
- working mode.

Under normal Unit operation and design basis accidents the system is in the waiting mode. In the working mode, the passive filtering system should ensure rarefaction over the whole height of the inter-containment gap, as compared to atmospheric pressure.

4.3.1.3 System Operating Parameters

The greatest capacity of the passive filtering system in terms of clean air, provided rarefaction is maintained over the height of the inter-containment gap, is 0,066 kg/s for the worst external conditions. This value is equivalent to the total untightness of the containment, or 1,5 per cent of the containment volume per day.The containment pressure is 0,5 MPa(abs.), i.e., it exceeds design leak value by 5 times.

4.3.2 Experimental Study of Steam-Gas Media Flow through Concrete Wall Cracks

This experimental installation is designed and built with the purpose of conducting research into steam-gas propagation through cracks in the concrete wall. The results demonstrate that within the concrete block temperature range of 20,0 – 100 ^{0}C, there are no moisture drops at the concrete crack outlet. The absence of drops indicates that the filtering plant will not be subject to moistening. No drops were detected at the crack outlet during an increase in flow rate of pure steam as well as steam-air mixture at concrete block temperature of 20 ^{0}C. Block heating over 100 ^{0}C resulted in drops appearing from the concrete crack.

Additional design illustrations and safety data are provided in Figures 11 to 17.

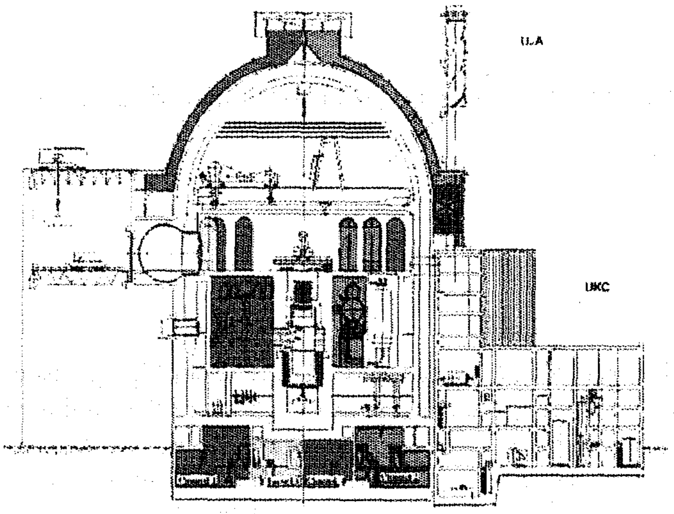


FIG. 11 Section through Reactor building and Process building

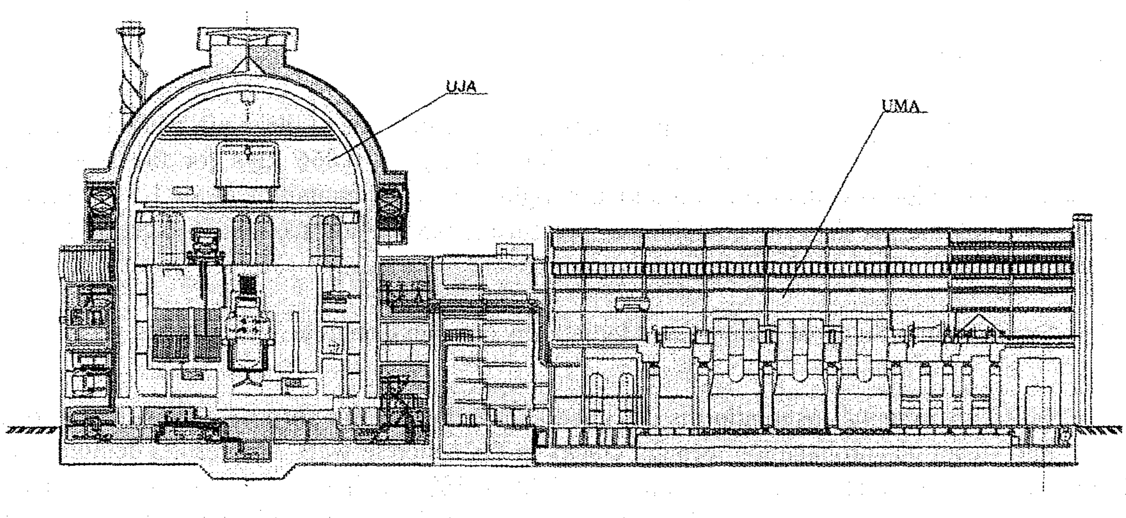


FIG. 12 Section through Reactor building and Turbine building

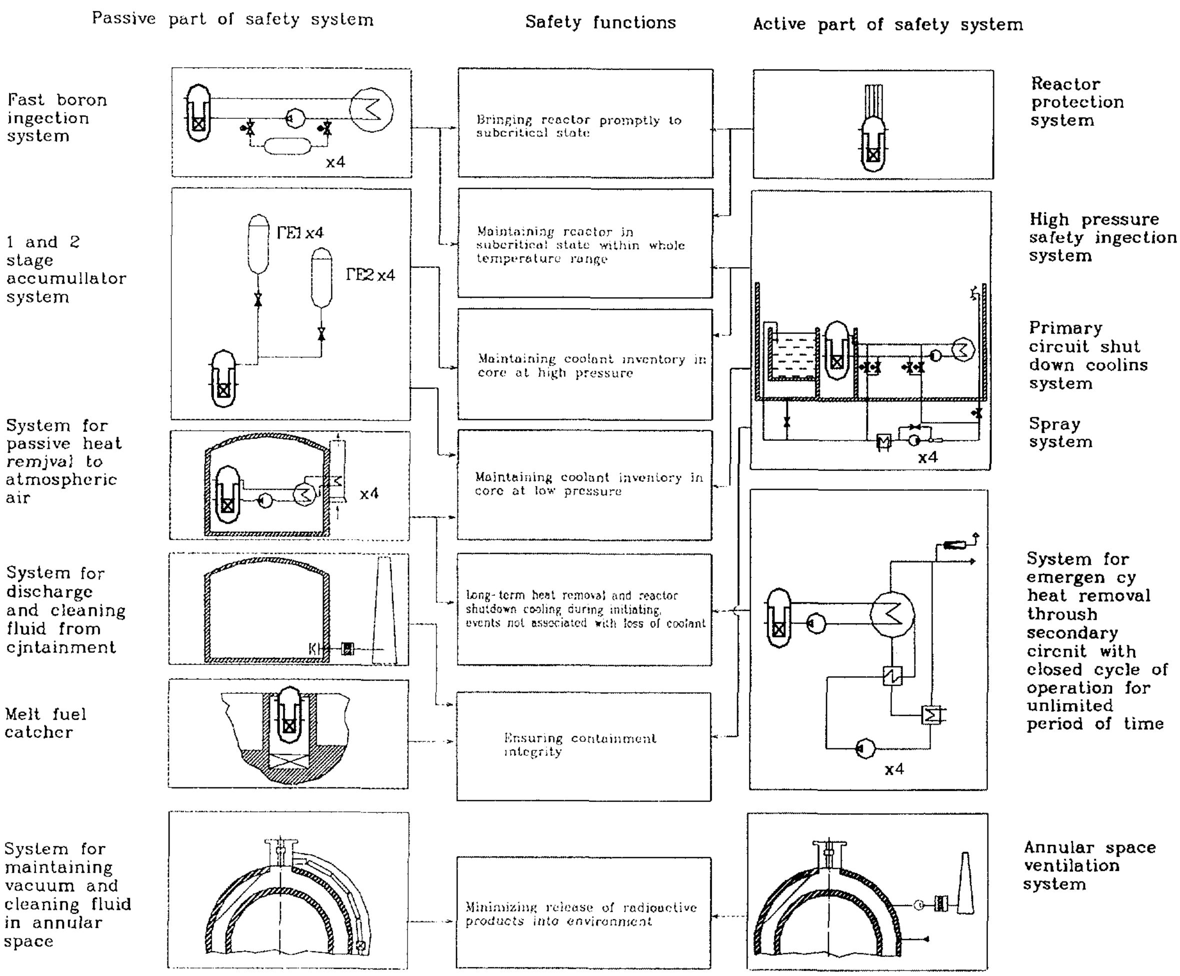


FIG. 13 Overview of principal approaches to ensure safety function in V-392

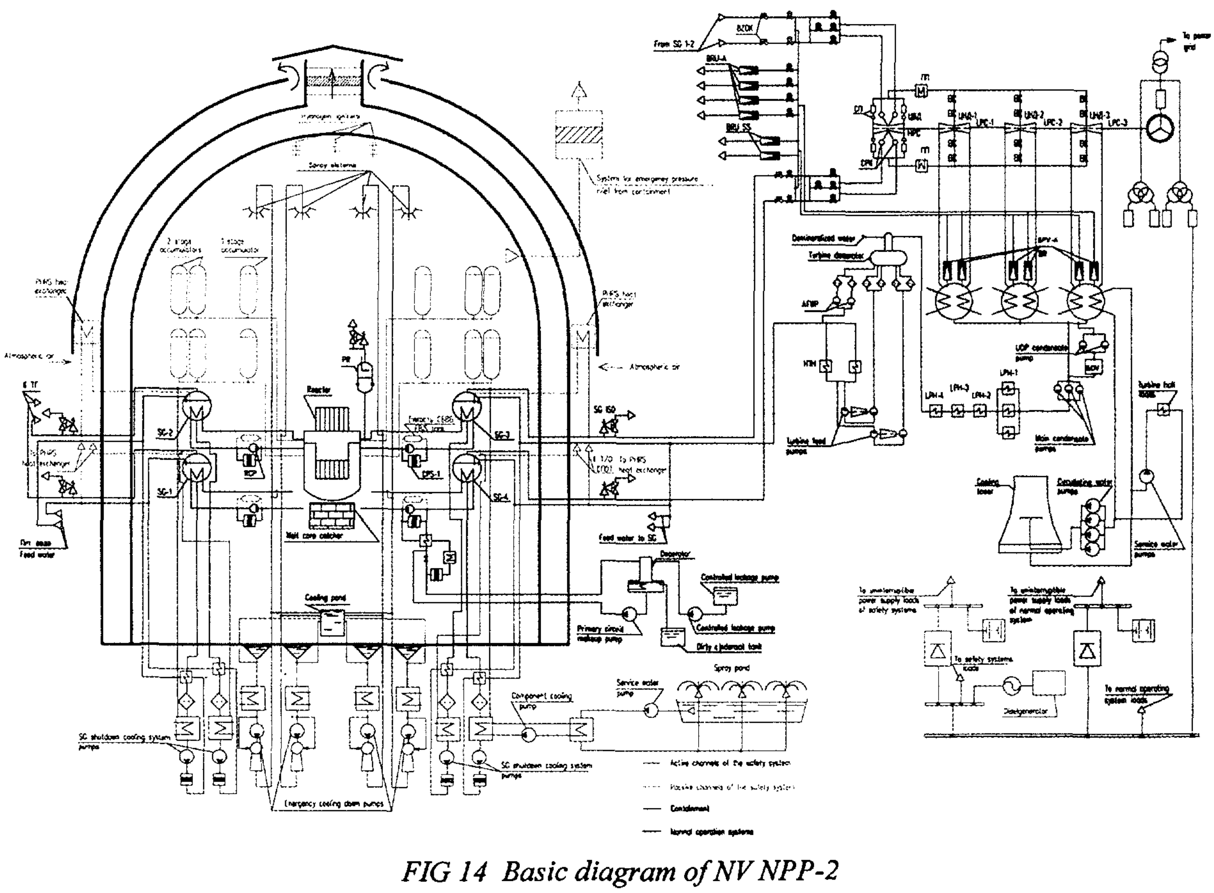


FIG 14 Basic diagram of NV NPP-2

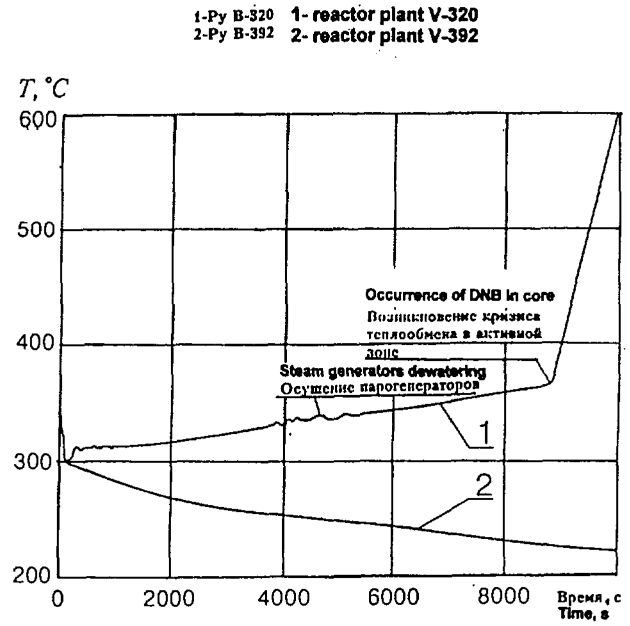


FIG 15 Max. fuel cladding temp. upon total loss of AC power

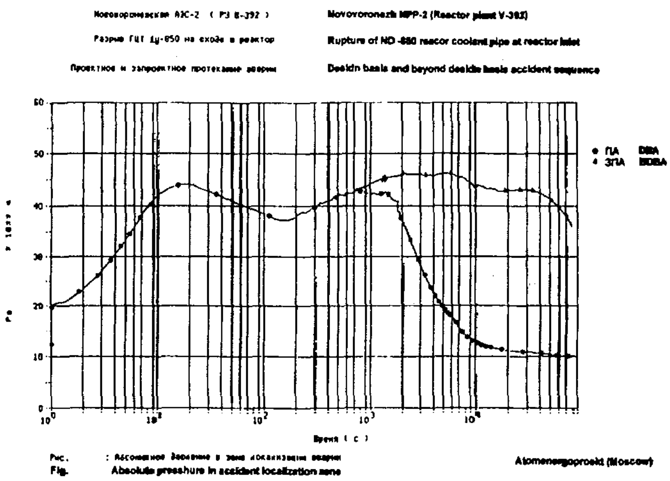


FIG 16 Rupture of cold leg pipe (DBA and BDBA)

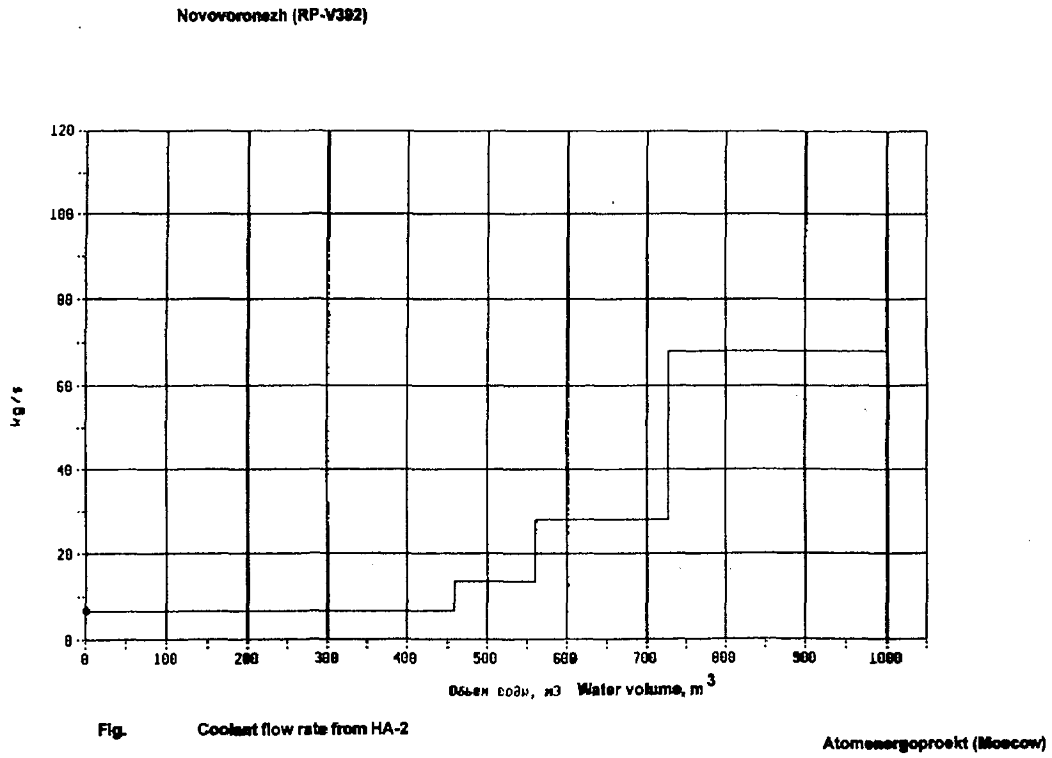


FIG. 17 Coolant flow rate from 2nd stage accumulators