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BASIC DESIGN DECISIONS FOR ADVANCED AST-TYPE NHRs

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Abstract

On the basis of the AST-500 reference design decisions and of the experience gained in the RF during the pilot NDHPs development and construction, the advanced NHR AST-500M has been developed recently by OKB Mechanical Engineering, as well as a whole series of heating and co-generation reactor plants of various unit power. All the designs represent enhanced safety reactor plants meeting the contemporary national requirements and international recommendations for nuclear plants of the new generation. The main objectives for the advanced NHR development are considered. New design decisions and engineering improvements are described briefly.

1. INTRODUCTION

To the end of the 1980s solid experience had been gained in the ex-USSR resulting from the extensive activity on the development, fabrication, construction and installation of several nuclear heating reactors of the AST-500 type on two twin-unit pilot stations near the large cities of Nizhny Novgorod (former Gorky) and Voronezh.

In-depth analyses of AST-500 NHR safety performed after the Chernobyl accident, activities on design upgrading in compliance with the revised safety codes and new licensing procedures and, at last, the joint work together with the IAEA safety review team during the Pre-OSART mission at the Gorky NDHP (1989) not only proved undoubtedly a sound basis of the design and of its safety concept, but also revealed the potential for further improvement of the design engineering decisions and characteristics.

Taking the AST-500 as a reference NHR design with in-depth-developed inherent safety features, a series of nuclear heating reactors of a unit power ranging from 30 to 600 MW has been developed recently at OKB Mechanical Engineering [1,2]. These designs rely upon the OKBM experience in the development of various types of nuclear reactors, including, besides NHRs, propulsion reactors for nuclear ice-breakers and marine ships. They accumulate the most promising engineering novelties and know how. They also take into account the contemporary world trends in development of the new generation of nuclear power reactors and the generally recognized international requirements for the safety of advanced NPPs.

2. MAIN OBJECTIVES FOR AST-500 IMPROVEMENTS

The following principal objectives have been adopted for the AST-500 NHR upgrading and for the development of its advanced modifications:

- simplification and standardization of engineering decisions in the framework of the whole series of AST-type reactor plants;

- gaining better economics;
- improving the basic equipment reliability and life-time extension;
- enhancement of safety by improving the reactor immunity to plant personnel errors, as well as to external and internal impacts.

The following indicators were accepted as the design goals (compared to the reference design characteristics):

Goal	Indicator
Extension of the reactor pressure vessel and main equipment service life	1.5 - 2 times
Reactor power uprating at the same dimensions	1.2 times
Fuel life-time extension	1.3 times
Fuel burn-up increase	1.7 times
Cut of equipment items in engineered safety systems	1.4-1.5 times
Decrease of auxiliary power, including that for responsible consumers	1.5 times
Cut of electrical controlling safety systems (CSS)	transition from 3-channel to 2-channel structure of CSS
Enhancement of plant immunity to common cause failures (fire, flooding, etc.)	use of self-actuated passive systems
Increase of design margins for strength	with account of hydraulic shocks and emergency increase of pressure in confinement systems
Standardization of engineering decisions for all NHRs in series	90%

The major portion of the advanced design decisions, particularly those regarding the reactor plant components service life extension, reliability and safety enhancement has been already implemented in the upgraded design of the Voronezh NDHP.

Besides, the results of the activity on the improvement of AST-500 reactor characteristics were used in the new designs of small and medium power reactors for electricity and heat co-generation power plants being under development now.

3. IMPROVING AST-500 NHR RELIABILITY AND ECONOMICS

To illustrate the progress in the reference design (AST-500) advancement the following decisions may be highlighted.

3.1. Increase of heat output to the heating grid

The reactor uprating potential is validated for the pilot plant AST-500 with no need for changing the composition of the plant equipment. This goal can be attained at the expense of working parameter variation in periods when the grid water temperature required for the consumer is below the design value. The reactor thermal power can be increased up to 600 MW at the grid water temperature decrease below 128/32°C (in summer, spring and autumn).

Increase in grid water flow by 25% G_{nom}, in grid heat exchanger surface by 25%, as well as in the primary heat exchanger surface allow to provide the delivery of 600 MW into the grid for the entire heating period. Investigations are underway of the capability to further increase the total heat output without deterioration of the reactor plant safety level (Fig.1). A considerable increase in heat-exchanger surface built into the reactor without changing the reactor dimensions has been achieved by the use of smaller tubes with a close pitch (tubes of 10x1.2 mm, square array with 14 mm pitch).

3.2. Reactor plant design simplification

The lower main joint of the reactor pressure vessel is eliminated in the advanced NHR along with the related complex of in-service control means.

The reactor plant systems simplification may be illustrated with the example of the primary coolant purification system (Fig.2). In the pilot plants the reliability of this system operation determined the reliability of the heat delivery from the NDHP to the consumers, which can be explained by the following reasons:

- in the AST-500 the VVER-440 reactor CRDM of the ARK-type were used with the need of assured filling with water of all internal cavities on the primary circuit side;
- in the integral reactor, the aforementioned requirement is met by continuous feeding of sealing water into the drives by pumps of the reactor coolant purification system;
- an emergency protection signal is generated at loss of sealing water flow.

The reactor coolant purification system is the only system in the pilot AST-500 having a continuous circulation of primary water beyond the reactor boundary. The requirements for the flowrate in the system (and hence, the diameter of related tubes and penetrations in the reactor vessel), and the continuous mode of operation are dictated by the CRDM operation and not by maintaining the reactor coolant chemistry.

The operational experience gained from the OKBM-designed marine reactors testifies a sufficiency of intermittent operation of the primary coolant purification system and a possibility to reduce flow in the system.

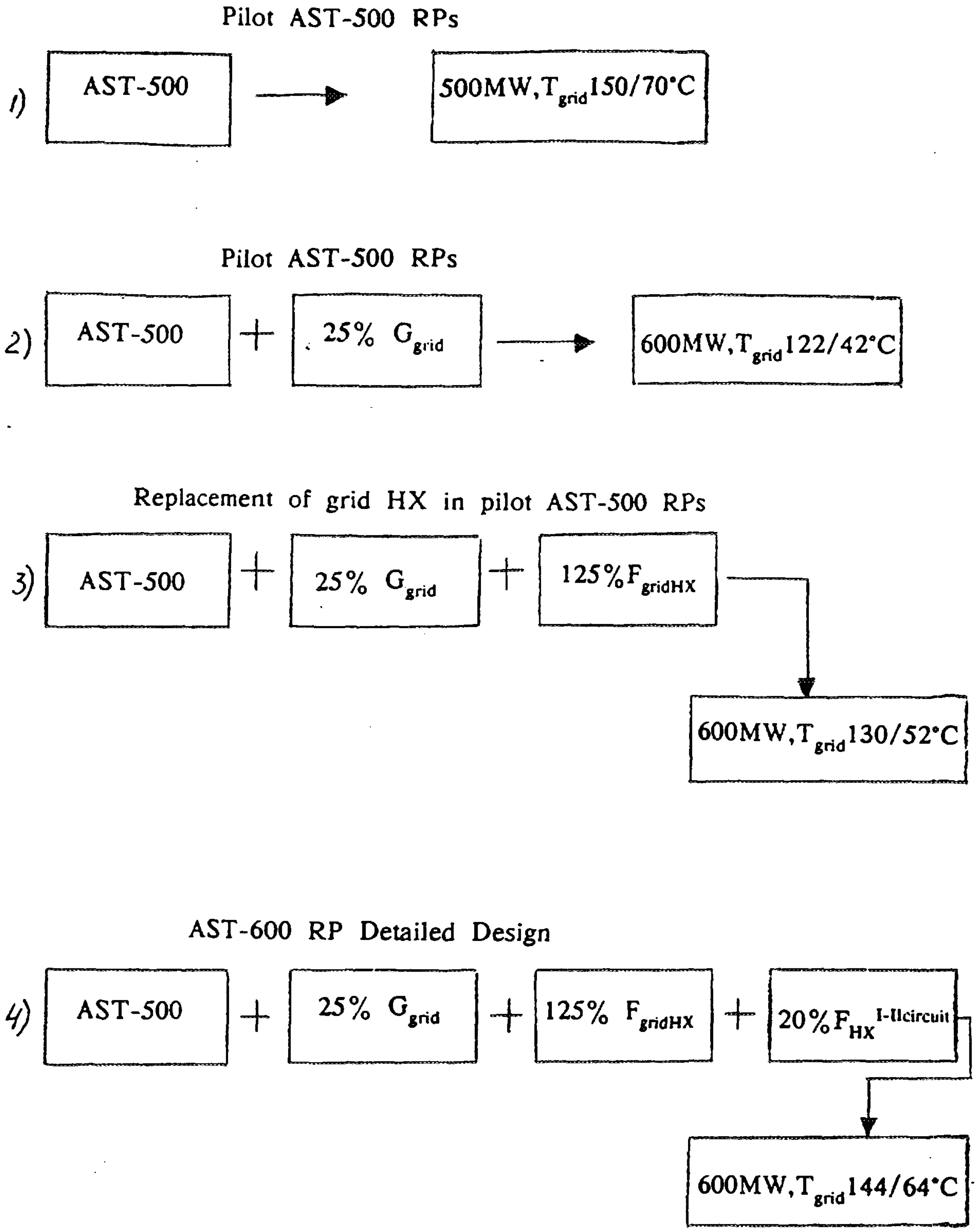


Fig.1. AST-NHR Heat Output Uprating Potential

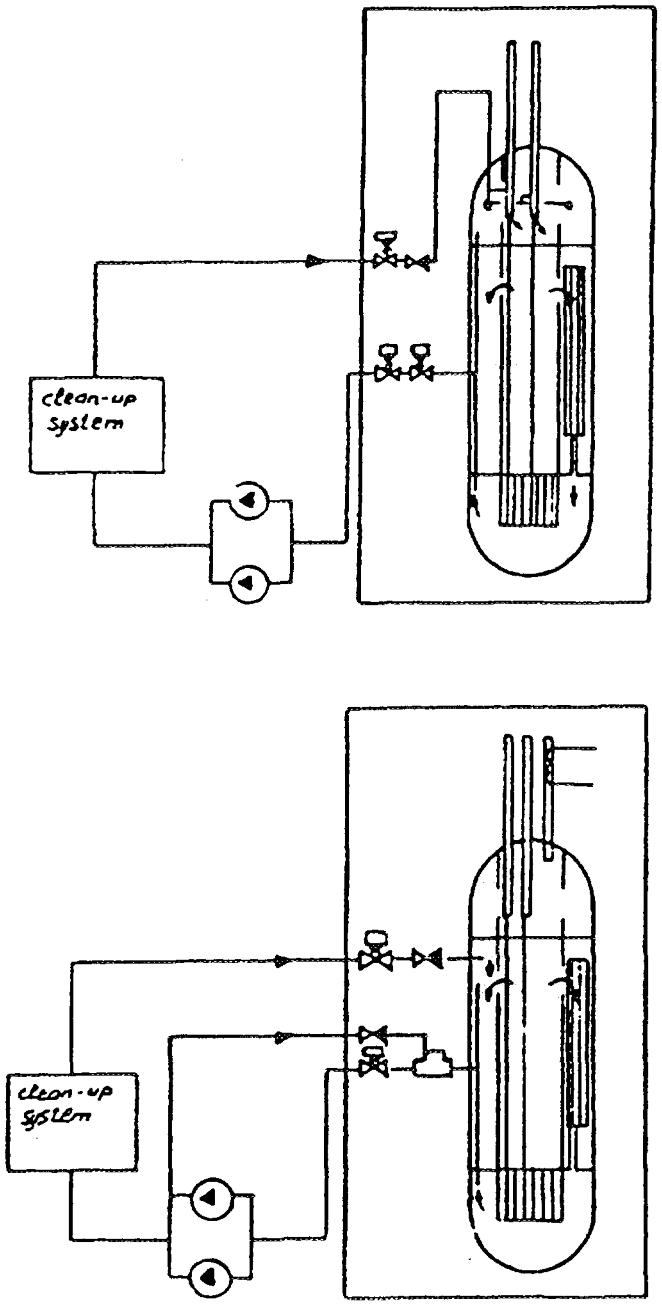


Fig.2. AST-500 Reactor Coolant Purification System Upgrading

The following decisions were adopted to solve the problem:

- upgraded CRDM were developed and tested successfully; they work in a steam-gas medium; besides, their service life was extended;
- the diameters of penetrations in the reactor pressure vessel and of pipelines in the system are decreased (to 40 mm for the penetration);
- in-reactor header and sealing water distributing pipes are eliminated;
- the periodicity of the system connection to the reactor is not more than once per three months;
- the system's pump delivery flow and consumed power are reduced 2 times. A modernized pump of the "impeller upward" type is used without a special gas removal system;
- a self-actuated hydro-controlled valve is used in the intake pipeline of the system within the guard vessel boundaries.

4. SAFETY ENHANCEMENT AND USE OF SELF-ACTUATED SYSTEMS AND SAFETY DEVICES

4.1. Self-actuated ERHR channel on the reactor

The followings factors gave the impetus for development of the additional ERHR channel:

- strive for independence and diversity of ERHR trains, that is an ERHR channel was needed which would be completely arranged within the guard vessel and which would not be connected to secondary loops. It is a structural accessory of the reactor and uses a self-actuation principle;
- evidence of testing the CRDM in a steam-gas medium that showed practically zero heat transfer into the CRDM cooling circuit.

So the idea emerged, and was then realized, to install immediately on the reactor closure head the condensers whose size is similar to that for the CRDM and whose inner cavity is connected like the CRDM with the steam-gas volume of the in-reactor pressurizer.

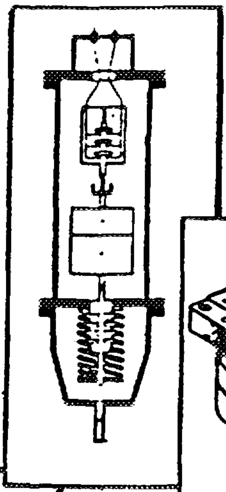
In the stand-by mode the partial pressure of the gas in the condenser's cavity exceeds 0.9 P_1, thermal losses into the cooling circuit are practically absent. The condenser self-actuation is provided by gas blown off through the self-actuated device in response to signals of a primary pressure build-up or a coolant level lowering in the reactor.

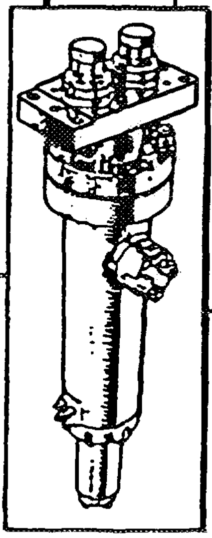
4.2. Self-actuated devices for safety systems

The premises for the development and implementation of these devices were the following (Fig. 3):

- preservation of their operability at external and internal impacts, including fires, flooding, seismic effects, etc. Impossibility to disable the fulfillment of protective functions by the plant personnel;
- design simplicity and relatively low cost;

Direct Action Safety Devices





Functions:

- deenergization of CPS drives;
- connection of emerg. heat removal channels;
- closure of isolation valves;
- connection of tanks with water and boron solution;
- equipmen overpressure protection

Operation principle:
Self-actuated devices actuate by direct influence of determining parameter, independently of electric control systems and actions of operating personnel in case of flooding, fires and other extremal impacts; blocking the devices by operator during reactor operation is excluded

OKBM Priority Development

Fig.3. Direct Action Safety Devices

7. PRIMARY CIRCUIT OVERPRESSURE PROTECTION

The heat removal principle for the primary system overpressure protection without dumping of radioactive medium is used in the design of advanced AST, as well as in the pilot AST-500. This became possible due to the reliable means used for reactor shut down and for residual heat removal, and due to the properties intrinsic for the given reactors.

This design approach meets the modern requirements of regulations for safety of NPPs.

8. IN-REACTOR CONTROL

The following parameters are monitored continuously by the control means provided for the reactor:

- neutron flux density (neutron power, reactor period and reactivity); there are 8 ionization chamber suspended in the reactor;
- coolant level in the reactor; there are 6 level indicators;
- pressure in the reactor; pressure gauges are located in the guard vessel;
- temperature in the reactor; thermocouples are arranged at the heat exchanger inlet (core outlet);
- energy release and temperature at the core inlet/outlet; instrumentation probes are located immediately in the core.

Besides, the radioactivity of the primary circuit medium (in the steam-gas plenum, coolant), the primary coolant chemistry and gas regime indicators (hydrogen, oxygen and the others), leak-tightness of the main reactor joint and primary circuit valves are monitored on-line.

The metal of the reactor vessel is checked when the reactor is shut down.

9. FUEL HANDLING AND STORAGE

Fuel is reloaded by means of the universal refuelling machine ("dry" method), the same as in the pilot units.

The refuelling machine moves on a rail track laid on the reactor hall floor and attends the area between rails, where the reactor pit, the reactor internals pits and the spent fuel storage pool are arranged.

The refuelling machine is intended to perform the following operations:

- transport of the reactor removable internals (control rod guide tubes connecting device unit, in-vessel barrel) from the reactor to the corresponding pits and backwards;
- transport of fuel assemblies inside the reactor core, between the core and the storage pool, inside the storage pool, (with fuel cladding integrity control), between the storage pool and universal seat (shipping cask, etc.).

Spent fuel is held up in the storage pool for radioactivity and residual power decreases, then it is removed in special container-cars from the site to a reprocessing plant.

10. PRIMARY CIRCUIT WATER-GAS REGIME

The steam-gas pressurizer provided for the primary circuit pressure control uses a explosion-proof helium-hydrogen mixture (96% He + 4% H_2). The hydrogen content in the primary coolant is specified (not less than 2 g/kg at core inlet) and monitored. A special system is provided for the primary circuit make-up with a He-H_2 mixture. The given hydrogen content in the coolant ensures water radiolysis suppression in the reactor.

In LOCAs the gas mixture is blown off from the pressurizer through the bubbler.

11. INSPECTIONS AND REPAIRS

The reactor unit design allows to perform location and plugging failed tubes in the primary heat exchanger, as well as a tubing section replacement after their lifetime is expired.

The reactor vessel metal and the weld joints are tested (using a particular ultrasonic technique) when the reactor is shut down.

12. DECOMMISSIONING

The design service life of the AST power units was initially determined to be 30 years. However, the design peculiarities of this reactor plant (thick water layer around the core) provide low irradiation and induced radioactivity of the reactor internal structures and construction concrete which gives ground for an extension of the reactor unit lifetime up to 50-60 years. It also facilitates its decommissioning.

Because the service life of the reactor building structures is determined to be as much as 100 years, a possibility is provided for dismantling the reactor units and installation of new ones.

Taking into consideration the aforementioned peculiarities of the reactor unit, equipment dismantling work could be started immediately following its operation life completion and reactor unit shut down (without delay for a plant conservation and observation period).

The quantity of radioactive materials during decommissioning of the AST unit subjected to disposal amounts to 50 m^3.

13. CONCLUSION

The AST-500 reactor plant, being the first in a family of integral PWRs developed by OKBM, represents a new generation reactor plant of passive safety. It is mastered in serial production and is used as a part of Voronezh NDHP under construction.

- positive operation experience for prototypes and similar devices used in the marine reactors;
- possibility to solve on a system level the problem of reactor emergency protection, emergency cooling and radioactivity confining systems actuation.

So, the main goal of implementing such devices is to improve the controlling safety systems reliability, to simultaneously reduce a number of electrical controlling safety systems' trains and to reduce their price.

5. DESIGN DECISIONS CONTINUITY AND STANDARDIZATION

For the development of a series of AST-type reactors the principle of design decision standardization was adopted. Among the advanced decisions which are to be implemented in the pilot NHRs, the following ones are developed:

- upgraded core with four fuel cycles of two-year duration;
- advanced protective system's control and power supply subsystems;
- use of self-actuated devices;
- equipment life extension;
- heat output increase, etc.

All AST-type designs contain a set of proven engineering decisions which are implemented into small power nuclear plants that are being developed now, including ones for electricity and heat co-generation. Particularly, decisions ensuring enhanced safety were implemented which allows to realize a single concept of the new generation reactors.

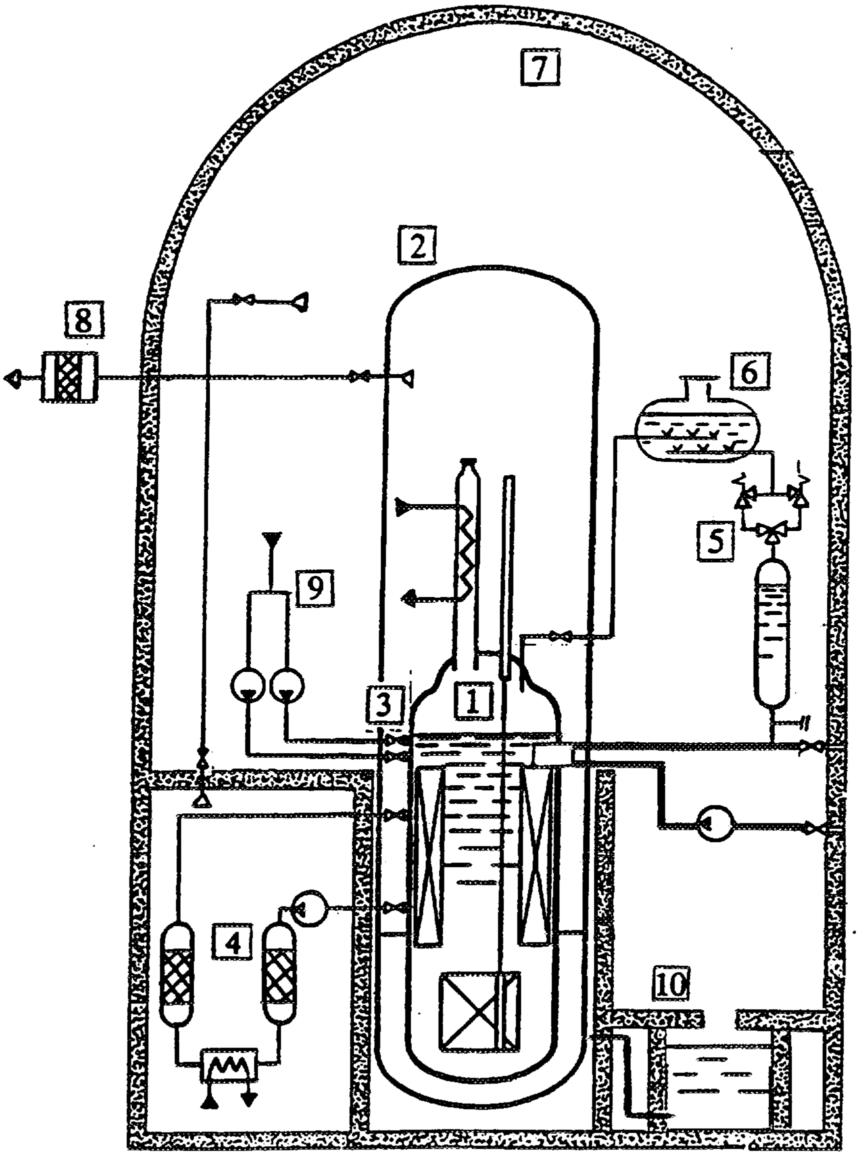
This regards the design of the reactor unit and the basic equipment, structures and equipment of passive safety systems, spectrum of self-actuated devices used, emergency control algorithms and the usage of back-up systems and devices.

6. CONFINEMENT AND MITIGATION OF RADIOACTIVE DISCHARGES, CORE PROTECTION AGAINST UNCOVERING

The following protective barriers (Fig.4) are used effectively to confine radioactive products, and to prevent and mitigate their release at accidents [3]:

- "cold" UO_2 fuel;
- fuel element claddings;
- leak-tight reactor with large margins for strength;
- guard vessel;
- intermediate circuit, designed for primary coolant pressure;
- leak-tight compartments, concrete reactor pit;
- bubbling of emergency discharges before their release into the containment;
- containment with aerosol filters;
- decreased radioactivity of emergency discharges due to keeping the core covered with water.

The high safety level of this reactor plant is recognized by national review and regulatory bodies, and is confirmed by the international safety review team (IAEA Pre-OSART mission).



1. Reactor
2. Guard vessel
3. Quick-acting licalization valves
4. Leak-tight rooms
5. PORV
6. Bubbler
7. Containment
8. Filter
9. 1-ry circuit makeup system
10. Water storage tank

Fig.4. Radioactivity Release Mitigation and Core Uncovery Prevention Means

The basic engineering decisions used in the AST-500 were then borrowed and further developed in a series of AST-type advanced NHRs ranging from 30 to 600 MW. Therewith, problems are being solved for assuring AST plant competitiveness compared with fossil-fuelled heat sources, for simplification of plant systems, equipment and operation, along with the provision of a safety level exceeding that already reached and substantiated for the pilot AST-500 NHR.

A significant portion of the decisions that have been developed for the advanced AST is planned to be used in the pilot NHRs.

Simultaneously, the whole set of engineering decisions providing reliability, economic efficiency and safety of the AST reactor is realized on a system level in reactor plant designs for nuclear power, particularly for the electricity and heat co-generation plants of the ATETS-150 type.

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