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NEW GENERATION NUCLEAR POWER UNITS OF PWR TYPE INTEGRAL REACTORS

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Abstract

Design bases of new generation nuclear power units (nuclear power plants - NPP, nuclear co-generation plants - NCP), nuclear district heating plants - NDHP), using integral type PWRs, developed in OKBM, Nizhny Novgorod and trends of design decisions optimization are considered in this report.

The problems of diagnostics, servicing and repair of the integral reactor components in course of operation are discussed. The results of safety analysis, including the problems of severe accident localization with postulated core melting and keeping corium in the reactor vessel and guard vessel are presented. Information on experimental substantiation of the suggested plant design decisions is presented.

INTRODUCTION

The integral lay-out realized in boiling water reactors and BN-type reactors is a result of the search for optimum technical and economically substantiated decisions.

An analogous search process is also characteristic of the reactors of PWR type.

Investigations and developments allow the conclusion to be made that certain conditions integral reactors have considerable advantages as for mass and size in comparison with loop-type and unit-type plants.

Besides the integral lay-out, the reactor has advantages as for safety, quality of fabrication, mounting, building time and removal from operation. But the integral lay-out objectively complicates the reactor design and the problems of operational service, it causes the necessity to use highly reliable in-reactor equipment.

In the development of integral reactors especially important are specific characteristics of the heat exchanger (steam generator) built into the reactor, because the reactor vessel dimensions depend largely on heat exchange surface dimensions.

The lifetime reliability of the reactor components should be confirmed by operational experience as a part of operating reactor plants and their prototypes or by broadened complex representative tests at testing facilities in the conditions corresponding to operation conditions in the plant.

For some decades, OKBM specialists developed ship nuclear power plants and experience has been accumulated on the development of some equipment and the NPP as a whole, fabrication and experimental development of some equipment, designer supervision of the fabrication at the factories and in course of operation.

The afore-mentioned allowed the development of new generation nuclear power plants with integral reactors.

First of all is the reactor plant AST-500 for NDHP, which may be located in the vicinity of large cities.

The AST-500 reactor plant is the first in the group of the plants with integral PWRs. Its characteristics are widely known. Its main peculiarities are following: natural coolant circulation in the reactor, high safety level provided by passive means.

The high safety level of the RP AST-500 was recognized by national technical and ecological expert examination, supervision bodies and a special commission PRE-OSART IAEA.

The main fundamental decisions of the NPP, such as integral reactor design, use of guard vessel, use of passive safety systems of various principles of operation with deep redundancy and self-actuation became the basis of the whole group of the developed plants of ATETS-200, VPBER-600 type and the others.

The main advantages of the integral design in comparison with traditional loop-type designs:

- localization of radioactive coolant in one vessel (excluding purification system);
- absence of large diameter pipelines and nozzles in the primary circuit;
- keeping the core under water level at any loss-of-tightness due to the proper choice of guard vessel volume;
- decrease of neutron fluence to the reactor vessel to the level, excluding any noticeable change of the vessel material properties, radiation embrittlement (fluence $<10^{17} n/cm^2$);
- higher completeness of the reactor plant important equipment, of the guard vessel at the delivery to the site and as a result increase in the quality of mounting the power units as a whole;
- reduction of NPP building time to the reducing of the installation work and simplification of construction work;
- considerable simplification of the technology and operations at NPP decommissioning and RP change for repeated use of NPP structures.

Possible negative consequences of the integral reactor design are the following:

- delivery of off-gauge heavy cargo from the factories;
- the necessity to increase considerably the rated load of mounting cranes at the site.

Corresponding analysis and the experience of delivery of AST-500 reactors and guard vessel to the sites of Nizhny Novgorod and Voronezh confirm the feasibility of such delivery by the existing engineering means.

REACTORS PLANT FOR NUCLEAR COGENERATION SYSTEMS OF ATETS-200 TYPE

Reactor plants of ATETS-200 type are a group (ATETS-80, ATETS-150, ATETS-200) of plants of the same kind, developed on the base of an integral reactor with natural coolant circulation, they are autonomous sources of electric energy and heat.

Compactness of the integral reactor, simplicity of the primary circuit and use of a highly efficient steam generator allows natural circulation in all conditions and excludes the use of pumps when providing electric power of up to 200 MW. The ATETS-200 reactor vessel dimensions are not greater, than the dimensions of AST-500 reactor vessel, mastered by the industry.

The investigations confirm the possibility of increasing power up to 250-280 MW.

The plant is notable for the wide variety of passive channels for residual heat removal:

- to each of two heat exchange loops (Fig.1) a channel is connected, which provides

ATETS-200 reactor plant flow diagram

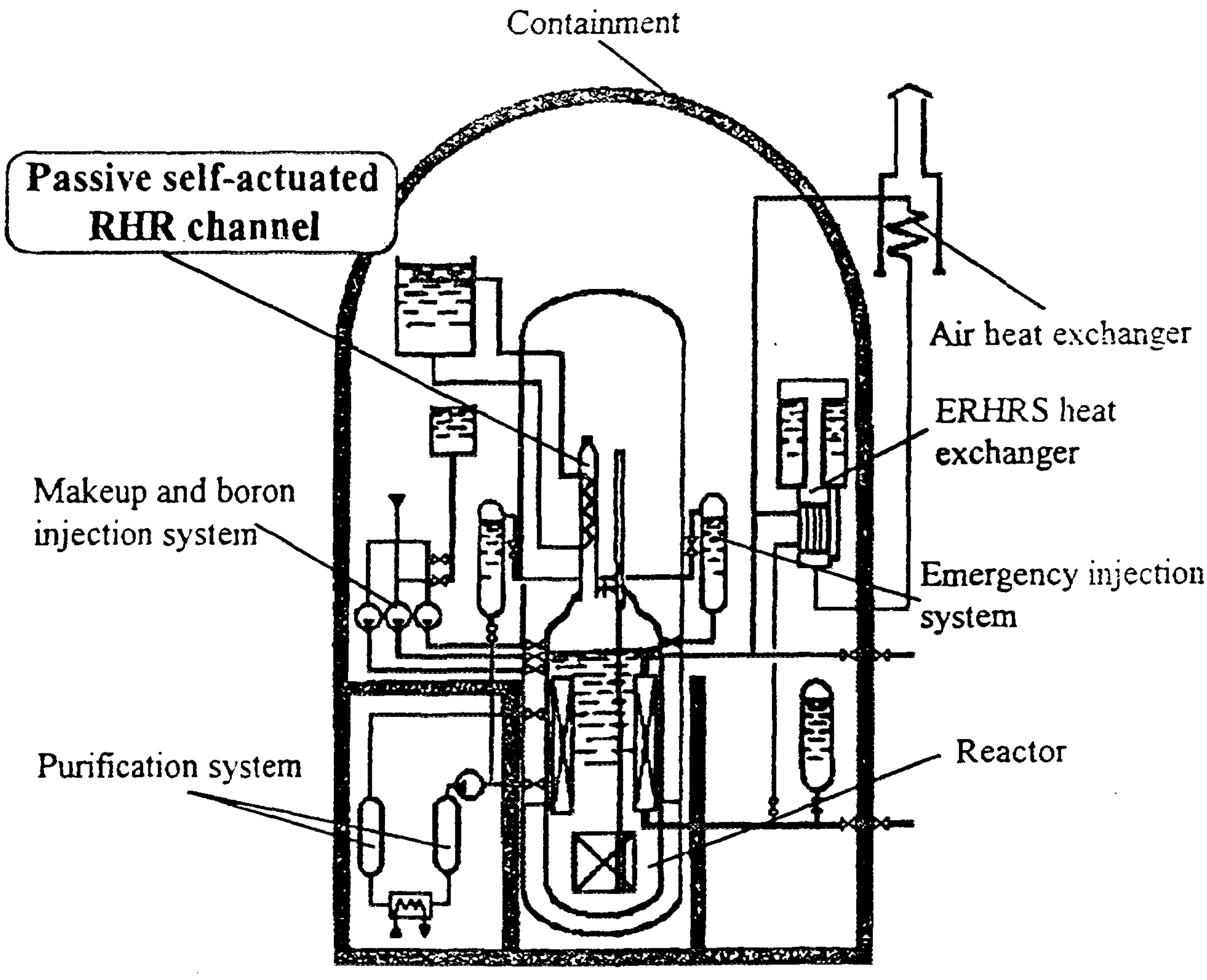
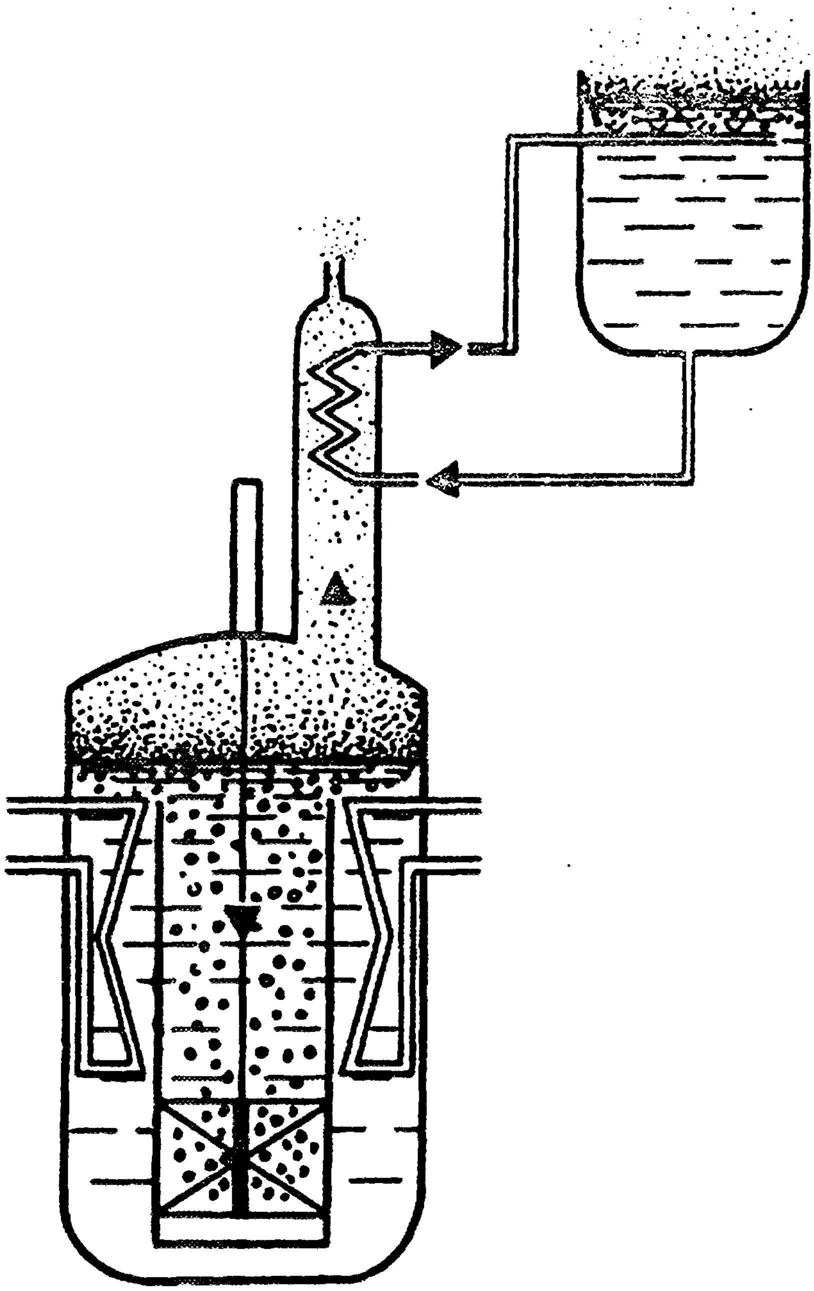


Fig.1



ERHR channel
on the Reactor Upper head

Water inventory in ERHR tank - 150 m^3

Cooldown diration (grace period) - 24 hrs

The channel self actuates at the reactor pressure increase up to 21^{+1} MPa

Fig.2

residual heat removal through SG with natural circulation with heat removal to water tanks, from where water is evaporated to atmosphere;

- independent passive channel of heat removal (Fig.2) is located on the reactor. With its help primary circuit heat is transferred through the wall of the condenser-heat exchanger by natural circulation to a water tank and then it is removed to atmosphere.

Self-actuation of ERHRS channels with emergency protection actuation and, if necessary, of actuations of location system with use of self-actuation devices is provided.

VPBER-600 REACTOR PLANT

An integral reactor with forced coolant circulation at emergency power level operation and natural circulation for residual heat removal is used in the design of the VPBER-600 reactor plant for the power unit of a new generation NPP of 640 MW(el) power.

Forced coolant circulation is provided with the help of six leak-tight circulation electric pumps, located on the bottom of the reactor vessel.

In the design of some equipment and systems the decisions have been mode which have been verified by long experience of operation of the existing nuclear power plants.

Calculation analysis of the wide range of accidents, performed on the base of both deterministic and probabilistic approaches demonstrated the high safety of the plant. Safety in the course of three days is provided by passive means without power supply from outside nor personnel intervention.

Form the point of view of a deterministic approach for severe core damage, multiple failures of safety systems elements and systems as a whole are necessary. The probability of severe damage to the core, evaluated deliberately conservatively is $< 10^{-8}$ per reactor/year.

Nevertheless the search for design decisions for optimization of the reactor design and improvement of characteristics including safety provision for severe accident-accidents with postulated melting of the core continues.

As a result circulation pumps were moved from the bottom to the cylindrical part of the reactor vessel, reactor internal heat exchangers of the system for emergency heat removal were excluded, engineering decisions for the limitation of the consequences of severe accident were proposed and the possibility of corium confinement in the reactor vessel or guard vessel was shown.

Moving the circulation pumps to the cylindrical part of the vessel simplifies the operational servicing of the reactor, excludes the possibility of coolant leakage below the core and improves the conditions for corium confinement in the core and in the reactor vessel and for creation of an in-reactor corium catcher.

The schematic diagram of the system of severe accident localization, presented in Fig.3, includes:

- heat exchangers-condensers, of the system of purification and boric reactivity compensation, total power 10 MW are located in the guard vessel;
- two safety complexes (DN 100), each consisting of a membrane-rupture device and a safety valve in series;
- two temperature-actuated devices for the reactor pressure relief;
- bubbler of approximately 300 m^3 volume;
- receivers of 600-700 m^3 volume.

The bubbler and receivers have the same strength as the guard vessel. In a severe accident with core melting when the temperature in the reactor reaches 500-700°C, safe

SYSTEM OF SEVERE ACCIDENTS LOCALIZTION

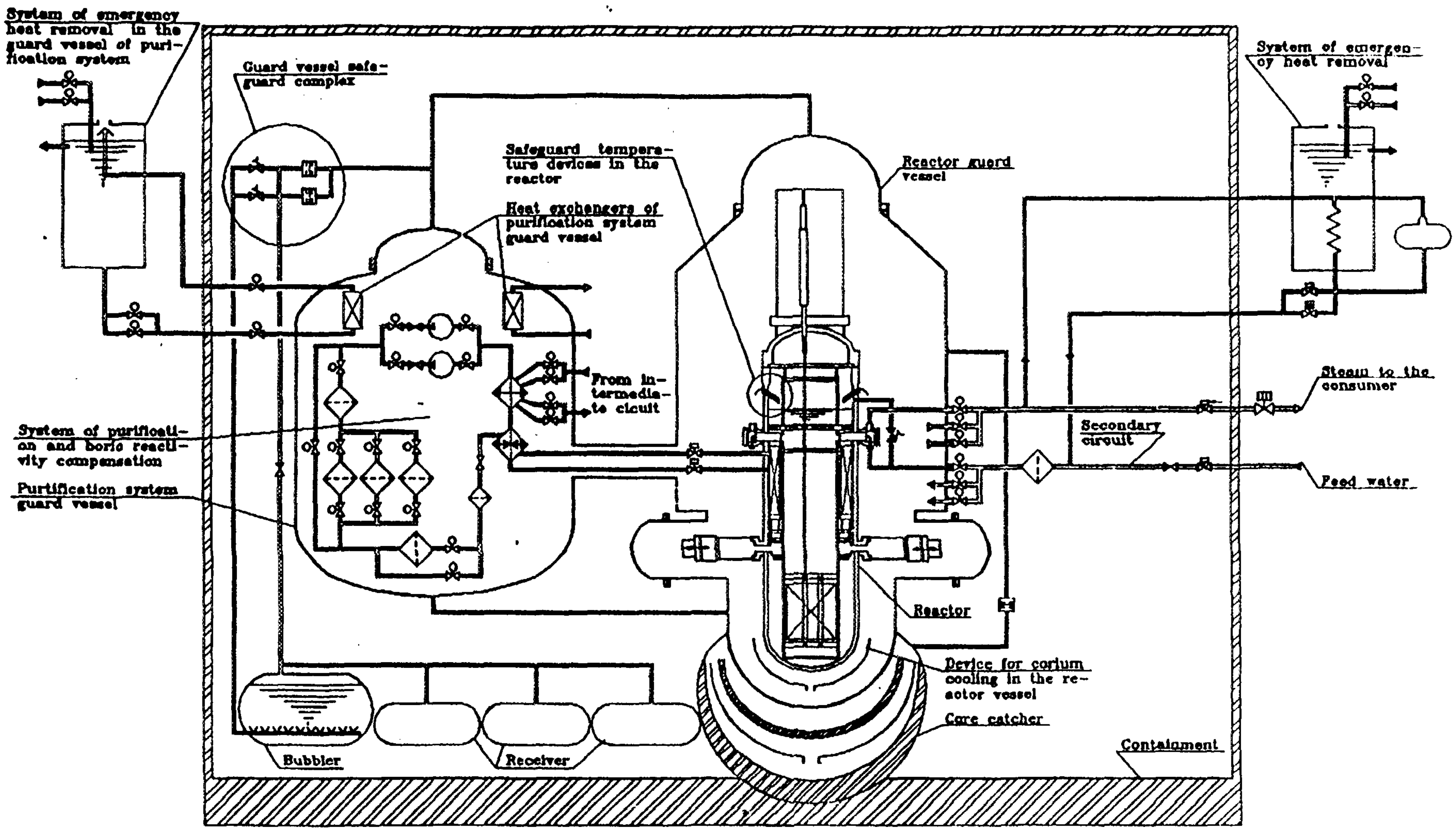


Fig. 3

Complex System of VPBER-600 Thermo-physics and Safety Investigations

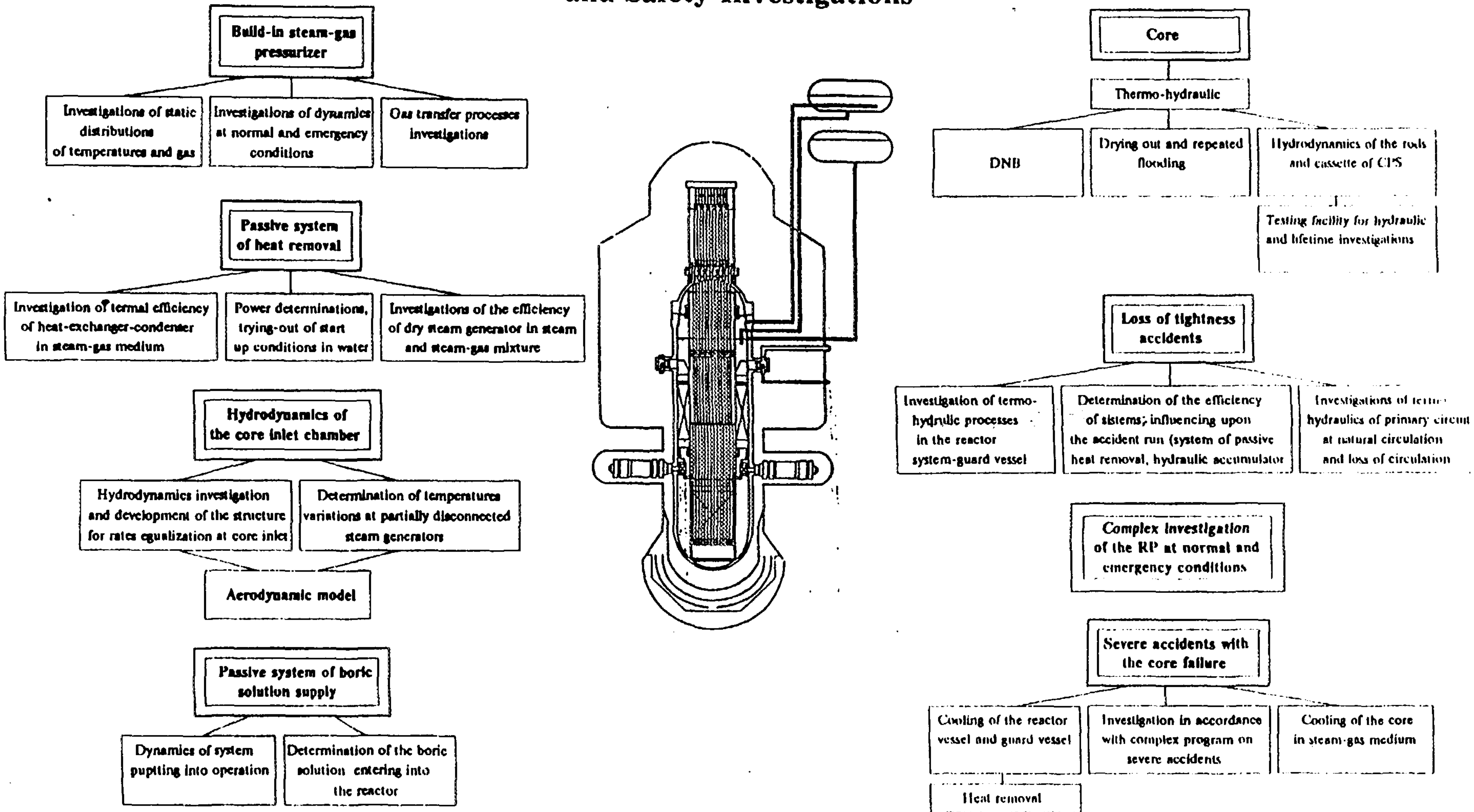


Fig.4

temperature devices of the reactor are opened and steam-gas mixture is discharged from the reactor to the guard vessel, untilpressure is equalized in the guard vessel and the reactor. When pressure in the guard vessel is 5.0 MPa a membrane-rupture device on the guard vessel is broken, a safety valve is opened and steam-gas mixture is discharged from the guard vessel to the bubbler. Gases liberated in core melting are pressed out to receivers, this passively solves the problem of provision for hydrogen safety.

Heat removal from the reactor vessel when corium is confined in the reactor or from the guard vessel false bottom when corium is confined in the guard vessel if it leaves the reactor vessel is performed with the help of heat exchangers-condensers in the guard vessel of the purification system.

In normal operation heat exchangers in the guard vessel of the purification system are disconnected from heat exchangers unit by the pipeline for water supply and are connected remotely by the operator in the event of severe accident.

CALCULATION AND EXPERIMENTAL JUSTIFICATION OF DESIGN DECISIONS OF INTEGRAL REACTORS

Integral reactors, developed in OKBM, being one of the varieties of PWR, are based on the common research and development work and on the experience in the creation, operation and development of such reactors.

But the novelty of the design decisions, connected with the integral lay-out of the reactor, the presence of a steam-gas pressurizer and guard vessel, the absence of circulation loops in the circuit and some others, demands special research work to be performed.

A lot of research work, connected with the experimental study of thermo-hydraulic processes in integral PWRs with a built-in steam-gas pressurizer have been performed in the existing experimental base.

The experimental investigations performed confirmed the main design decisions for equipment and systems and allowed substantion of the correctness of the chosen regime parameters, reliability and safety of the plant.

Together with the problems of the study of thermo-hydraulic processes, occuring in the plant in emergency conditions and of substantiation of the operability and efficiency of the provided safety systems, the most important problem for the experiments is to collect representative information for computer code verification.

The main investigations, which are being performed at present are the following:

- investigations of DNB in fuel assemblies and temperature state of fuel elements at partial and complete dry-out of the core;
- investigation of the conditions of steam condensation from steam-gas mixture in the heat exchangers-condensers of the emergency residual heat removal systems and in the built-in steam generators;
- investigations of steam-gas mixture distribution inside the pressurizer;
- investigations of water-gas and chemical conditions in the primary circuit, including gas transfer in the circuit;
- investigations at integral facilities, including a wide range of emergency conditions with primary circuit loss of tightness and heat removal disruption;
- investigations, verifying thermo-hydraulic and lifetime characteristics of steam generators.

Fig.4 shows the complex of facilities for thermo-physical investigations and safety of VPBER-600.

To verify the results on the problem of corium confinement, it is necessary to perform additional investigations into thermo-physical, physico-chemical and thermo-mechanical processes, to improve calculation modes and computer programs. Besides, the conservativeness of assumptions made considerably compensates for the lack of information and gives every reason to obtain a positive solution of the problem of corium confinement in the reactor vessel or guard vessel.

Now testing facility for an integral PWR of 200 MW power is being made ready for putting into operation.

MAINTENANCE OF INTEGRAL REACTORS (EXAMINATION, REPAIR, DIAGNOSTICS)

The scope and contents of the procedures for maintenance of integral reactors, developed in OKBM meet the requirements of national regulatory documentation. Thereby the following peculiarities of the integral reactor are taken into account:

- presence of the guard vessel;
- location of steam-generators (heat exchangers) in the reactor vessel.

A complex of special devices for scheduled servicing and, if necessary, for repair and reconditioning work which account for the peculiarities of integral reactor lay-out has been developed and tested in AST-500 reactor conditions.

As for inspection of metal and welded joints the following measures are provided in the design:

- periodic visual inspection with video recording of the part of the reactor vessel visible in the zones between heat exchangers with the help of a periscope and of the whole surface when the heat exchangers (SG) are removed;
- periodic eddy current and ultrasonic inspection of the welding and main metal of the reactor vessel in the core zone;
- periodic outside visual and ultrasonic inspction of the reactor vessel with the help of a rotational device and a universal self-propelled device;
- periodic radiographic inspection of the welds of nozzles and penetrations in the upper part of the reactor;
- periodic inspection of the main metal and welds of the reactor vessel using test sample;
- periodic visual inspection of the state of in-vessel devices on removal from the reactor.

The strength and leak-tightness of the structures is confirmed by:

- periodic hydraulic tests of the reactor and heat exchangers of the primary and secondary circuits (steam generators);
- periodic pneumatic tests of the guard vessel.

In RP power operation, constant control monitoring of the reactor and guard vessel leak-tightness is provided by measuring the GV environment parameters (pressure, activity, gas content), also acoustic-emission methods of inspection are used.

Constant monitoring of primary-secondary circuit heat exchanger (SG) leak-tightness at RP power operation is performed by measuring the activity and gas content of the secondary circuit medium.

In the event of heat exchanger (SG) loss of tightness the leaking section is looked for, the leaking part is plugged or (if necessary) the whole section is substituted with the help of special devices.

Analysis and discussion of the decisions made by operating staff experts have shown their acceptability during operation.

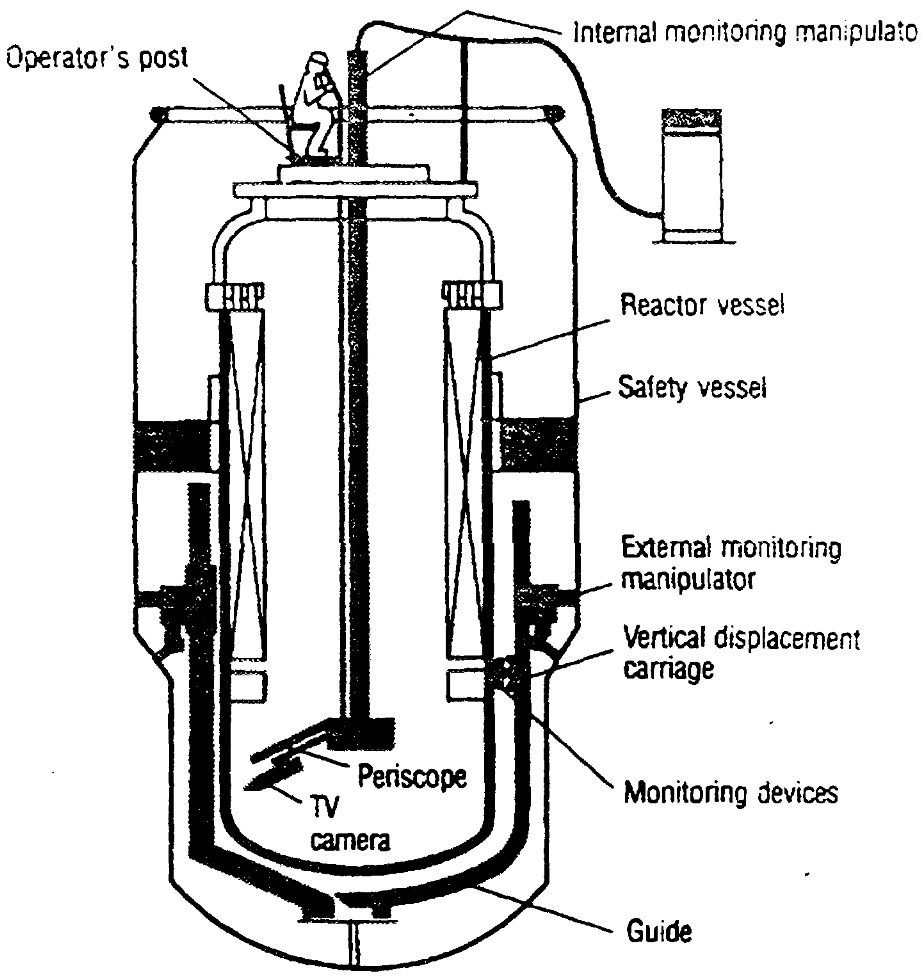


Fig.5

Arrangements for monitoring the AST-500 reactor vessel

When the problems of scheduled servicing and repair are considered, the problem of the cleating to operate the plant before the end of its lifetime should also be discussed.

One of the advantages of the integral reactor is that it is simple from the point of view of technology and radiation to remove it from operation. The presence of a thick water layer between the core and reactor provides low radiation levels from the structures. It allows performance of dismantling work in the reactor cavity using standard equipment without using special means of protection and unique mechanical arms, very soon after reactor shut down and core unloading. The main part of the equipment is low radioactive or even not radioactive and may be dismantled in the same way as at industrial plants. The mass of in-reactor equipment with high radioactivity, which is dismantled by standard means is 2% of the mass of the reactor unit.

In conclusion it should be mentioned, that the results of OKBM work on the reactor plants NDHP, NCP, VPBER show, that an integral lay-out of the reactor gives additional, new possibilities for NPP increase of safety in comparison with loop-type plants. The difficulties of operational servicing, caused by compactness are overcome when highly reliable in-reactor equipment is used.