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Development and validation of natural circulation based systems for new WWER designs

Y.A. Kurakov
Minatom RF, Moscow, Russian Federation

Y.G. Dragunov, A.K. Podshibiakin, N.S. Fil, S.A. Logvinov, Y.K. Sitnik
OKB Gidropress, Podolsk, Russian Federation

V.M. Berkovich, G.S. Taranov
Atomenergoproject, Moscow, Russian Federation

Abstract. Elaboration and introduction of NPP designs with improved technical and economic parameters are defined as an important element of the National Program of nuclear power development approved by the Russian Federation Government in 1998. This Program considers the designs of WWER-1000/V-392 and WWER-640/V-407 power units as the priority projects of the new generation NPPs with increased safety. A number of passive systems based on natural circulation phenomena are used in V-392 and V-407 designs to prevent or mitigate severe accidents. Design basis, configuration and effect of some naturally driven systems of V-392 design sited at Novovoronezh are mainly reflected in the present paper. One of the most important mean for severe accident prevention in V-392 design is so called SPOT - passive heat removal system designed to remove core decay heat in case of station blackout (including failure of all diesel generators). This system extracts the steam from the steam generator, condenses it and returns water to steam generator by natural circulation. The SPOT heat exchangers are cooled by atmospheric air coming by natural circulation through a special direct action control gates which operate passively as well. Extensive experimental investigation of the different aspects of this system operation has been carried out to validate its functioning under real plant conditions. In particular, full-scale section of air heat exchanger-condenser has been tested with natural circulation steam, condensate and air paths modeled. The environment air temperature and steam pressure condensing were varied in the wide range, and the relevant experimental results are being discussed in this paper. The effect of wind velocity and direction to the containment is also checked by the experiments.

1. INTRODUCTION

The Program of Russian Federation nuclear power development for 1998-2005 years and for the period till 2010 (approved by Russian Federation Government Resolution No. 815 dated July 21, 1998) has defined that the elaboration and implementation of new generation NPP designs with enhanced safety is the necessary factor of nuclear power extension in Russia. The new generation NPP projects shall meet up-to-date national and international requirements and envisage: (1) the probability of limiting release and serious core damage at beyond-design accidents less than 10^{-7} and 10^{-5} per reactor-year, respectively; (2) reduction of urgent evacuation area to 300-500 meters and emergency planning area to protect the population in case of beyond-design accidents to 700-3000 meters.

The safety of new NPP is provided by consistently implementing the defense-in-depth principle, based on the application of a system of barriers in the way of ionizing radiation and radioactive substance release into the environment, and also by realizing the engineering and organizational activities to protect these barriers. The National Program of nuclear power development considers the design of 1000 MW power unit with WWER-1000/V-392 reactor to be the priority project of new generation plants. This unit is so designed that radiation effect on the population and the environment is considerably below the allowable values established by the up-to-date regulatory documentation. The permit of Russian Federal Nuclear and Radiation Safety Authority (Gosatomnadzor) was granted to construct the power units with V-407 reactor plant on the Sosnovy Bor and Kola sites and two units with V-392 reactor plant on the Novovoronezh site.

Extensive application of passive safety means, using natural physical processes, along with the traditional active systems is a specific feature of both these designs. The IAEA Conference on "The Safety of Nuclear Power: Strategies for the Future" [1] has noted that the use of passive safety features is a desirable method of achieving simplification and increasing the reliability of the performance of essential safety functions, and should be used wherever appropriate. However, the application of passive means is connected with some problems, which have to be solved by each plant designer. The passive systems have their own advantages and drawbacks in comparison with the active systems both in the area of plant safety and economics. Therefore a reasonable balance of active systems and new passive means is adopted in V-392 design to improve safety and public acceptability of nuclear energy.

One important problem related to the implementation of the passive means is that, in the most cases, sufficient operating experience of the passive systems/components under real plant conditions does not exist. Besides, the existing computer codes for transient and accident analysis are not sufficiently validated for the conditions and phenomena which are relevant to the passive system functioning (low pressure, low driving pressure and temperature heads, increased effect of non-condensable, boron transport at low velocities, and the like). As a result, the time- and money-consuming research and development works may be needed individually for each reactor concept to validate the operability of the passive safety means proposed in the design. Therefore, the extensive experimental investigations and tests have been already performed and are being planned to substantiate the design of the safety features proposed for new units with WWER-1000/V-392 and WWER-640/V-407 reactor plants.

2. PASSIVE SYSTEMS IN NEW WWER DESIGNS

Safety features desired in future plants have been summarized by INSAG-5 in "The Safety of Nuclear Power" [2]. It notes that the Basic Safety Principles of INSAG-3 [3] remain valid and should become mandatory, and the extension of INSAG-3 principles gives further opportunities for improvement of safety on which new plant designs should begin to draw. These opportunities include several design approaches such as avoiding complexity, reducing dependence on early operator actions, among others, and include specifically giving considerations in the design process to passive safety features. Such functions as the containment heat removal, hydrogen management, core debris cool down and prevention of the containment floor melt-through are probably among the most appropriate areas for passive systems usage. Both novel and more or less proven passive means are proposed in many new water-cooled reactor designs [4] to fulfill these and other functions.

Some future reactor designs have only added a few passive components to the traditional systems, some others make wide use of the passive systems and components. New WWER concepts V-392 and V-407 are of this last category since a number of relatively innovative passive safety means are implemented in these designs to ensure or to back up the fundamental safety functions: reactivity control, fuel cooling and confinement of radioactivity.

2.1. Reactivity control

Traditional gravity-driven control rods are the main system to ensure reactor scram both in currently operating and new WWERs. For existing pressurized water reactors, this system is not sufficient to bring the reactor to a cold shutdown state; therefore the control rod system of existing WWERs is supported by pumped emergency supply of the borated water to the primary circuit. New WWER designs V-407 and V-392 have an increased number of gravity-driven scram rods to maintain shutdown margin even in the absence of boron supply during the reactor cooling down.

Although very good reliability records exist for scram excitation, some failures of the gravity-driven control rod insertion have been recognized. The failures occurred for the different reasons; in particular, the cases of insertion speed reduction and incomplete insertion due to fuel assembly deformation have been reported during last ten years (see for example [5]). Besides, some failure modes may be considered which could prevent all the control rods to insert, and it was the basis for designers to analyze Anticipated Transient Without Scram events.

Keeping that in mind, for WWER-1000/V-392 a special quick boron supply system has been designed as a diverse system to the gravity-driven scram system. A concentrated boron solution tank is connected to the suction and discharge pipes of each main coolant pump. The valves in the connecting pipes will automatically open if there is a demand for reactor trip but the reactor power after some time is higher than its value after scram should be. The concentrated boron solution is supplied to the reactor due to pressure difference between discharge and suction of the main coolant pump (pump head). Inventory and concentration of the boron solution is selected to ensure compliance with safety criteria in the design events accompanied by control rod system failure to trip the reactor. Even in case of loss of power, the pump head during coastdown is sufficient to push all the boron solution from the tank (i.e. for this case the boron supply function is ensured passively). The operability of the quick boron supply system has been confirmed by extensive experimental investigation using a scaled model.

2.2. Fuel cooling

The safety function “fuel cooling during transients and accidents” is ensured by provision of sufficient coolant inventory, by coolant injection, sufficient heat transfer, by circulation of the coolant, and by provision of an ultimate heat sink. Depending on the type of transient/accident, a subset of these function or all of them may be required. Various passive systems and components are proposed for WWER-1000/V-392 and WWER-640/V-407 reactor concepts to fulfill these functions.

For WWER-640/V-407 reactor, steam generator passive heat removal system (SG-PHRS) which does not require the electricity supply is designed to remove the decay heat in case of non-LOCA events and to support the emergency core cooling function in case of LOCAs. Reactor coolant system and passive heat removal equipment layouts provide heat removal from the core following reactor shutdown via steam generator to the tanks of chemically demineralized water outside the containment and further to the atmosphere by natural circulation as it is shown in Figure 1. Reactor power that can be removed from the core by coolant natural circulation is about 10% of the nominal value, which guarantees a reliable residual heat removal. Thus, in case of non-LOCAs the decay heat is removed by coolant natural circulation to steam generator boiler water. The steam generated comes into the passive heat removal system where steam is condensed on the internal surface of the tubes that are cooled on the outside surface by the water stored in the demineralized water tank outside the containment. The water inventory in this tank is sufficient for the long-term heat removal (at least 24 hours) and can be replenished if necessary from an external source.

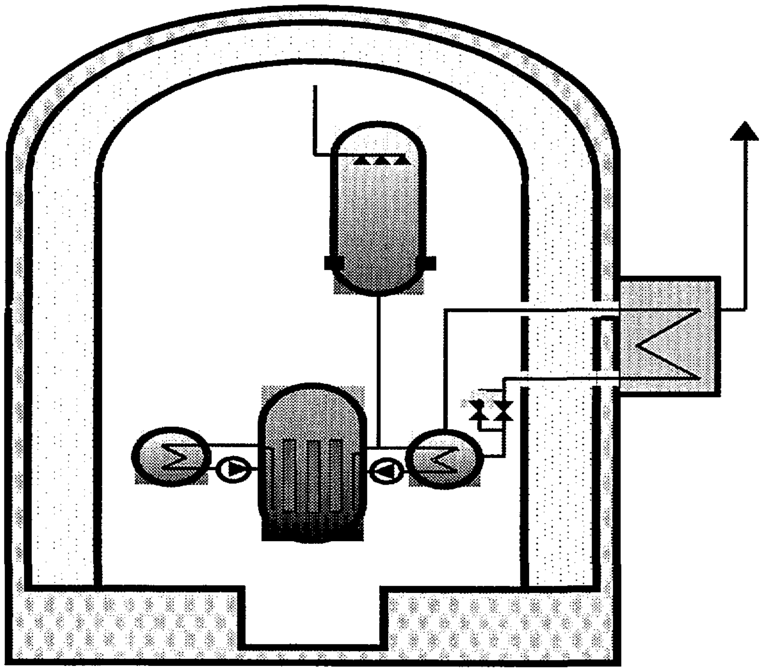

FIG. 1. V-407. Passive heat removal for non-LOCAs

Containment passive heat removal system (C-PHRS) of V-407 reactor removes heat from the containment in case of a LOCA and is designed to fulfill the following functions: (1) emergency isolation of service lines penetrating the containment and not pertaining to systems intended to cope with the accident; (2) condensation of the steam from the containment atmosphere; (3) retention of radioactive products released into the containment; (4) fixing of the iodine released into the containment atmosphere. The steam from the containment atmosphere condenses on the internal steel wall of the double-containment being cooled from the outside surface by the water stored in the tank. So, the system operates due to natural circulation of the containment atmosphere and water storage tank. The design basis of this system is to condense the amount of steam equivalent to decay heat release during 24 hours after reactor trip without the water storage tank replenishment.

The emergency core cooling system of V-407 reactor comprises three automatically initiated subsystems: (1) hydroaccumulators with nitrogen under pressure, which are the traditional ECCS accumulators being used at operating WWER-1000 reactors, (2) elevated hydrotanks open to the containment, and (3) equipment for deliberate emergency depressurization of the primary circuit. All these subsystems are based on the principle of passive operation providing for long-term residual heat removal in case of a loss-of-coolant accident accompanied by the plant blackout (i.e. AC power supply is not needed for ECCS operation). In the first stage of the accident, primary pressure is decreased due to loss of coolant and operation of the passive heat removal system. The further cooling down and pressure decrease are realized via steam generator PHRS and containment PHRS. When the pressure difference between primary circuit and containment has decreased to 0.6 MPa, the passive valves of the emergency depressurization system open connecting reactor inlet and outlet with the fuel pond space. When the reactor and containment pressure difference has decreased below 0.3 MPa, the ECCS hydrotanks begin to flood the reactor. This sequence results in creating of so called emergency pool where the reactor coolant system is submerged to and in connection of this emergency pool with the spent fuel pond. The natural circulation along the flow path shown in Figure 2 (reactor inlet plenum - core - reactor outlet plenum - "hot" depressurization pipe - fuel pond - "cold" depressurization pipe - reactor inlet plenum) provides the long-term heat removal from the core in case of a LOCA combined with loss of all electric power. The water in the emergency pool and spent fuel pool reaches the saturation point in about 10 hours. The steam generated will condense on the internal surface of the steel inner containment wall, and condensate flows back into the emergency pool. This configuration ensures also the heat removal from reactor vessel bottom to keep the corium inside the reactor in case of postulated core melt event.

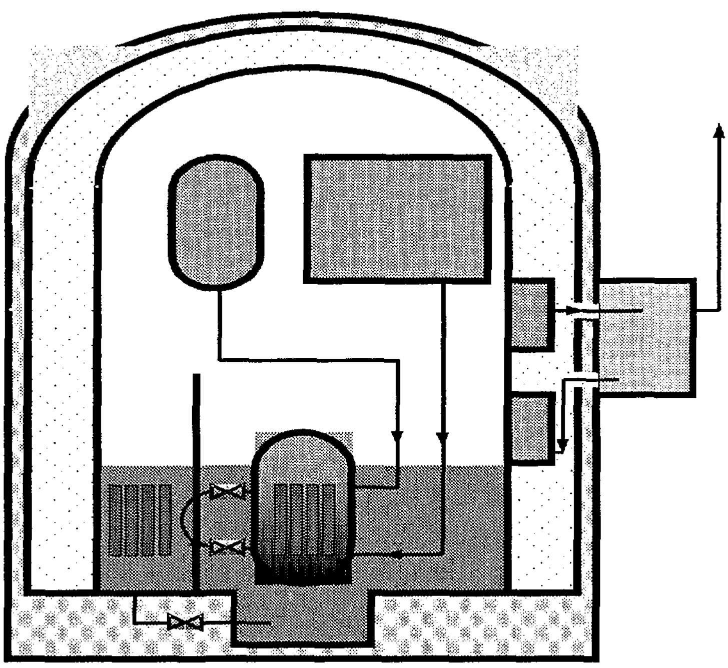

FIG. 2. V-407. Passive heat removal for LOCAs

The passive residual heat removal system (PHRS) is included in the V-392 design to remove heat from the reactor plant. The design basis of this system is that in case of station blackout, including loss of emergency power supply, the removal of residual heat should be provided without damage of the fuel and of the reactor coolant system boundary for a long time period. The PHRS consist of four independent trains; each of them is connected to the respective loop of the reactor plant via the secondary side of the steam generator. Each train has pipes for steam and condensate, valves and modular air-cooled heat exchanger installed outside of the containment as it is shown in Figure 3. The steam that is generated in the steam generator due to the heat released in the core condenses in the air-cooled heat exchanger, and condensate is returned back to the steam generator. The motion of the cooling media (steam, condensate and air) takes place in natural circulation.

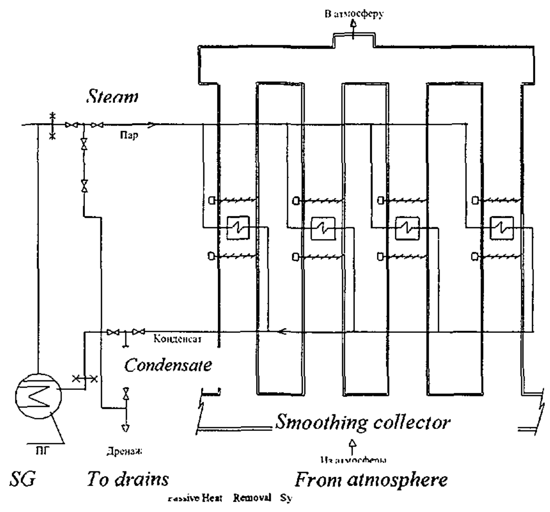

FIG. 3. V-392. Passive heat removal system

The passive system for reactor flooding during LOCA in V-392 design comprises two groups of hydroaccumulators as it is shown in Figure 4. First group (so called first stage accumulators) consists of four traditional ECCS accumulators being used at operating WWER-1000 reactors; these accumulators are pressurized by nitrogen to 6 MPa and connected in pairs to the upper and lower plenums through special nozzles in the reactor pressure vessel. Second stage accumulators are 8 tanks connected to the reactor coolant system through the check valves and special spring-type valves. These valves are kept closed by the primary pressure; when the primary pressure drops below 1,5 MPa, the spring open the valve. Such a connection configuration and valve design ensures continuity of hydrostatic head irrespective of the primary pressure change during an accident. Installation of hydraulic profiling of the outlet route ensures a step-wise limitation of the water flow rate from the tank when the water level in the tank is decreasing. The water inventory in the second stage accumulators (about 1000 t) ensures the core cooling for 24 hours during a LOCA even if all active ECCS mechanisms are inoperable. Joint operation of the second stage accumulators and SPOT gives a possibility to increase the period indicated.

2.3. Confinement of radioactivity

This safety function is ensured by protecting and maintaining the integrity of the potential radioactivity release barriers (fuel, reactor system boundary and containment). These barriers are passive components as themselves; in addition, several passive means are proposed in V-407 and V-392 concepts for the protection of these barriers (some of them are reflected above). As the containment is the last and most important barrier, both these designs imply substantial improvement of the containment protection against different loads related to design basis and severe accidents, and various passive systems are important part of this protection in V-407 and V-392 designs. This design decision is derived from the assumption that the active systems are more vulnerable to failures under conditions inside the containment during an accident.

In V-407 design, containment over-pressurization is avoided by passive containment cooling system (C-PHRS) as described above. To limit considerably the release of fission products beyond the containment, a permanent under-pressure is maintained in the inter-containment gap of the V-392 design. This safety function, one of the most important, is fulfilled by two systems: (1) an exhaust ventilation system equipped with a filtering plant with suction from the inter-containment gap and outlet into the stack; (2) a passive system of suction from the inter-containment gap. The first system is intended to control removal of steam-gas mixture from the inter-containment gap under accidents with total loss of power. The system is capable to remove at least 240 kg per hour that is equivalent to the inner containment leaks of 1.5% containment volume per 24 hours. The second system consists of lines connecting the inter-containment gap with the PHRS exhaust ducts, which are always in the hot state. This solution enables permanent removal and purification of inner containment leaks irrespective of the electricity supply and operator actions. According to estimations, the under-pressure is maintained at any point of the inter-containment gap with inner containment leaks up to 2.8% of containment volume per day (the design basis for the containment is 0.3%). The technical solution described above in combination with the systems for the containment pressure decrease (traditional spray system and new passive heat removal system) allows to give up the filtered venting system designed for V-392 in spite of this system follows the current requirements that filtered venting should not increase the risk of loosing the containment function and filtered venting is not required in the short term of a core melt accident.

Special systems and components are implemented in both new WWER designs to prevent hydrogen burning or explosion. For example, in V-392 design the hydrogen suppression system comprises passive catalytic hydrogen igniters based on an efficient high porosity cellular material. Each of 50 elements of this system is capable to oxidize about 30 grams of hydrogen per hour at its volumetric concentration 4%. This system prevents the explosive concentration of hydrogen even if 100% of the core Zr will be oxidized during an accident.

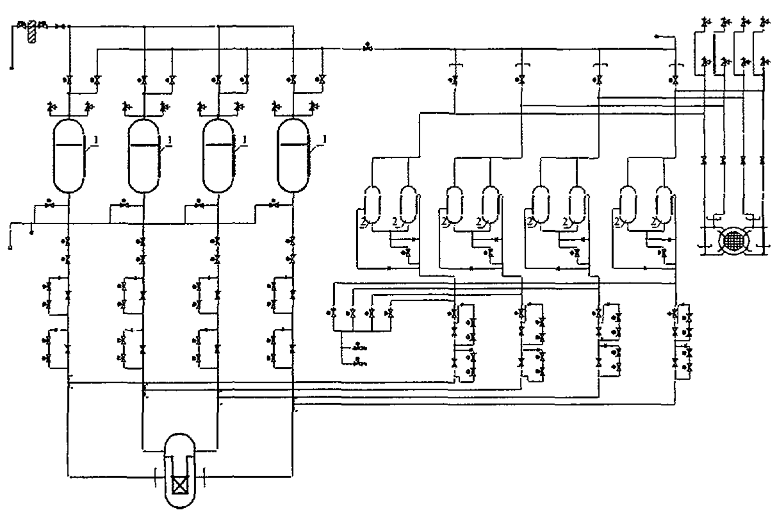

FIG. 4. V-392. Passive core flooding system

3. RESIDUAL HEAT REMOVAL SYSTEM FOR NVAES-2

Functional and structural diversity of WWER-1000/V-392 safety systems provide deep protection against common-course failures, and application of passive systems and active systems actuation without personnel interference yield deep protection against human errors. These engineering solutions used in NVAES-2 design enable the attainment of increased safety level, compared to the existing WWER-1000 plants.

For existing plants where the most of safety functions are ensured by active systems, the electric power supply is an important precondition for the successful operation of the safety systems. In spite of the very reliable emergency power supply from diesel-generators, loss of offsite power remains to be very essential contributor to the estimated core melt frequency from the internal initiating events (for example, more than 80% for unit 4 of Balakovo NPP [6]). For NVAES-2 with V-392 reactor plant, this figure is reduced to about 30% at absolute value 7.91×10^{-9}. As a whole, total core melt frequency for NVAES-2 is about three orders of magnitude less than for Balakovo-4 which is the latest WWER-1000/V-320 unit commissioned in Russia.

Passive residual heat removal system from the core via steam generators to the atmosphere as the ultimate heat sink (so called SPOT) plays an important role in the core melt frequency reduction mentioned above. The design basis for this system is that in case of station blackout during the most unfavorable atmosphere conditions the heat removal capacity with account for the failure of one channel shall amount to not less than 2% of the reactor rated power. The heat removal at the initial stage of the accident is performed due to partial water evaporation from the secondary side via steam generator relief valves to the atmosphere.

The SPOT system consists of four groups (corresponding to the number of reactor coolant system loops) of closed natural circulation circuits. In the ribbed tubular air-cooled heat exchanger (four heat exchangers for each of these circuits), steam extracted from the steam generator condenses, and the condensate flows by gravity to the steam generator boiler water volume. Under normal reactor plant operation, the SPOT system is under standby when all the SPOT circuits are in the warmed-up state. In case of plant blackout, the SPOT state changes from the standby to the operating condition. In addition to its main purpose (core decay heat removal in case of complete loss of a.c. power), the SPOT system can maintain the hot standby parameters of the reactor plant; for this purpose the SPOT has a special controller. The system is thermally insulated, so the heat losses in standby conditions are less than 0.1% of reactor rated power. Natural circulation in the SPOT system is provided by the corresponding layout of the steam generator, heat exchanger and draught air duct.
The steam circuit pipeline runs from fresh steam line to collector which distributes the steam by smaller tubes to four heat exchanger. The condensate from each heat exchanger is supplied by tubes to the collecting receiver and then by pipeline to the steam generator. Two isolation valves are installed at heat exchanging module inlet and outlet to isolate it in case of damage or maintenance. Small diameter pipelines with valves installed on them are provided for removal of air from heat exchanger when filling them with water during hydrotest and for periodical removal of non-condensables under standby conditions. Cooling air is taken from the atmosphere outside the reactor building. Air goes through the protective net and enters the annular corridor located around the reactor building and then to the heat exchanging modules. The air takes the heat from the steam and goes to the draught air ducts, which have the common outlet collector-deflector. Inlet and outlet gates and controller are installed on the airside of each heat-exchanging module. The gates open to switch on the heat exchanging

module to operation. The controller can be used to change the airflow rate to ensure additional SPOT system functions (for example, to maintain the reactor plant in the hot standby conditions).

Under standby conditions, the I&C system ensures for each heat exchanging module the measurements of the air inlet and outlet temperature, outlet air humidity, water level and temperature in the condensate lines. Information on the air humidity is especially important to detect a leak in due time. For this aim, humidity measurement is installed close to the upper part of the air space of the heat exchanger. Under accident conditions, power supply to the instrumentation is ensured by the sources of category 1. Under these conditions the number of measured parameters is reduced to condensate and air temperature and air humidity at the heat exchanger outlet.

4. EXPERIMENTAL INVESTIGATION OF SPOT SYSTEM

The SPOT is 16 air cooled heat exchangers-condensers (four ones for each steam generator), located outside around containment. The heat exchanger is canned in the box connected with exhaust stack above to ensure the air circulation. The design basis of the SPOT is the removal of 60 MW heat at the maximum design temperature of the external air (plus 50°C) taking into account failure of one train, so the power of one heat exchanger is 5MW. The air flowrate is controlled by partial opening of the air gates to ensure that the power of the heat exchanger should not be excessive in the minimum design temperature of external air (minus 40°C). Under standby conditions the air gates are closed, but the heat exchangers are in the hot state due to air leakage through air gates and heat transfer through the building constructions. These thermal losses are estimated in the design by value less than 0,1% of reactor rated power. The levers with load open the gates under de-energization of the electromagnets keeping the air gates in the closed position, and the air natural circulation through heat exchangers begins.

To confirm the design solutions mentioned above and the system characteristics, the extensive experimental investigations have been performed in OKB Gidropress. The investigations were carried out at the external air temperature from minus 19°C to plus 30°C and the pressure in the steam-condensate circuit from 0,5 MPa to 6,4 MPa. The start-up of the system from hot standby condition is performed, the heat losses are determined at the closed air gates, the system power against external air temperature and steam pressure is determined, the heat removal control is checked by the partial opening of the air gates. The investigations have given the convincing substantiation that the SPOT design functions are performed for WWER-1000/V-392 in the real conditions.

The test rig constructed in OKB Gidropress in 1991, includes the full scale heat exchanger of the SPOT system with representation of air and steam-condensate circuits as shown in Figure 5. The heat exchanger consists of about 200 flat spirals having small slope in relation to horizontal. The tube bank about 2 m high and total surface more than 300 m^2 has two inlet and two outlet collectors. On the sections with cross-sectional air flow, the tubes have ribs for intensification of the heat transfer to the air.

The tube bank is canned in heat-transfer box of square cross-section, which forms the heat exchanger air flow circuit. The lower and upper part of this box are connected with horizontal boxes for air inlet and outlet accordingly. The box upper outlet is connected to exhaust stack. The air gates are installed before of the heat exchanger and after one in pairs in one plane, each of these overlaps half of the cross-section of the circuit. The gate is the flat plate rotated about one's own axis. The lever with load which is retained by the electromagnet, when the gate is closed, is attached to the axis. The horizontal inlet box, the box of the heat-transfer bank, the horizontal outlet box and exhaust stack together create the circuit for air natural circulation. In this circuit (see Figure 6) the elevations of its components (including the height from inlet box axis up to top of the exhaust stack) and cross sections of all air circuit elements per one real heat exchanger are represented in the full scale. Whole air circuit, starting from the heat exchanger, has been covered by thermal insulation and sheathed by aluminium sheets to create adequate conditions concerning to the heat losses of the real SPOT system.

During the tests, the parameters necessary both for the confirmation of the SPOT design operation and for the development and validation of calculation models for the whole system and its separate elements have been measured. The set of the measured parameters includes:

- flow rate, pressure and temperature of the superheated steam,
- feedwater flow rate,
- flow rate and temperature of the condensate,
- flow rate, pressure and temperature of the saturated steam supplied to the heat exchanger,
- pressure and temperature of the atmospheric air,
- pressure difference along the heat exchanger in the steam-condensate circuit,
- air temperature in the air circuit elements,
- temperature of the heat-transfer tube surface,
- temperature of the thermal insulation.

These measurements have been shown in the control room of the test rig and the registration of the parameters have been realized by the data acquisition system IMPACT 3590 with subsequent treatment by a personal computer.

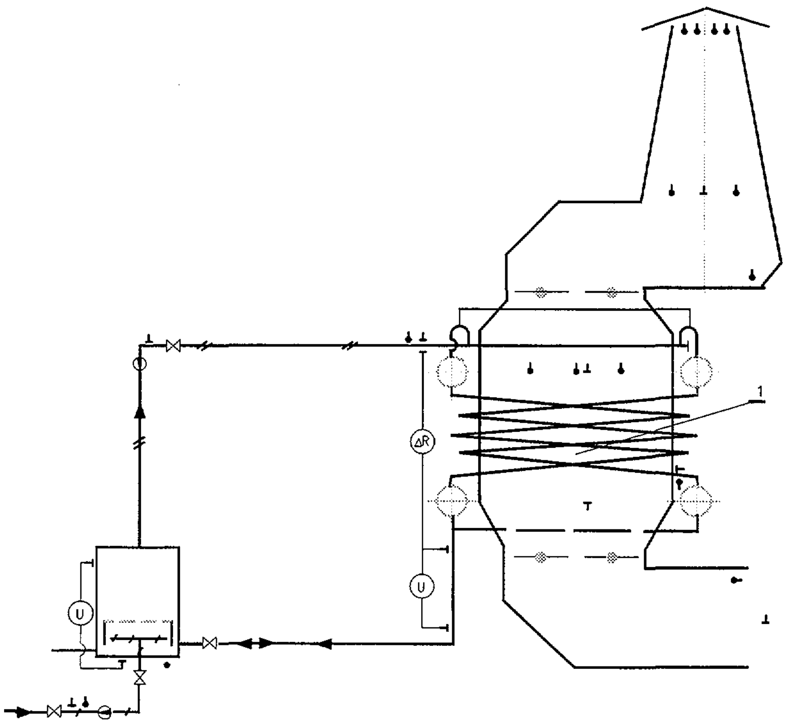

FIG. 5. Test rig flow diagram.

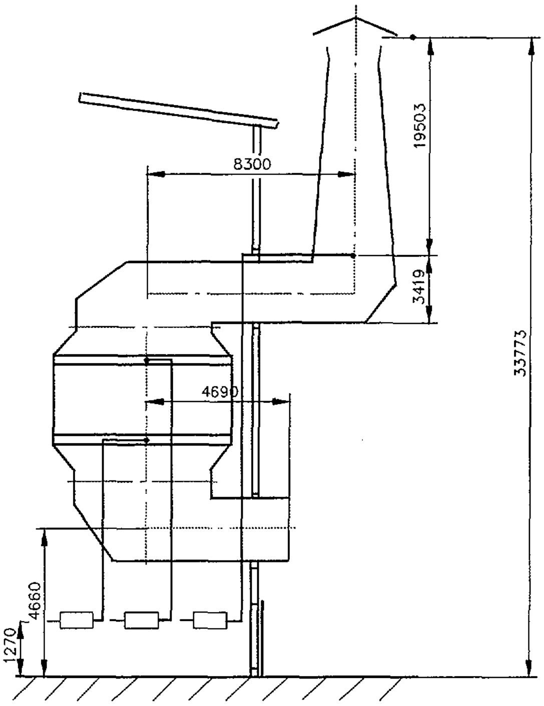

FIG. 6. Air circuit.

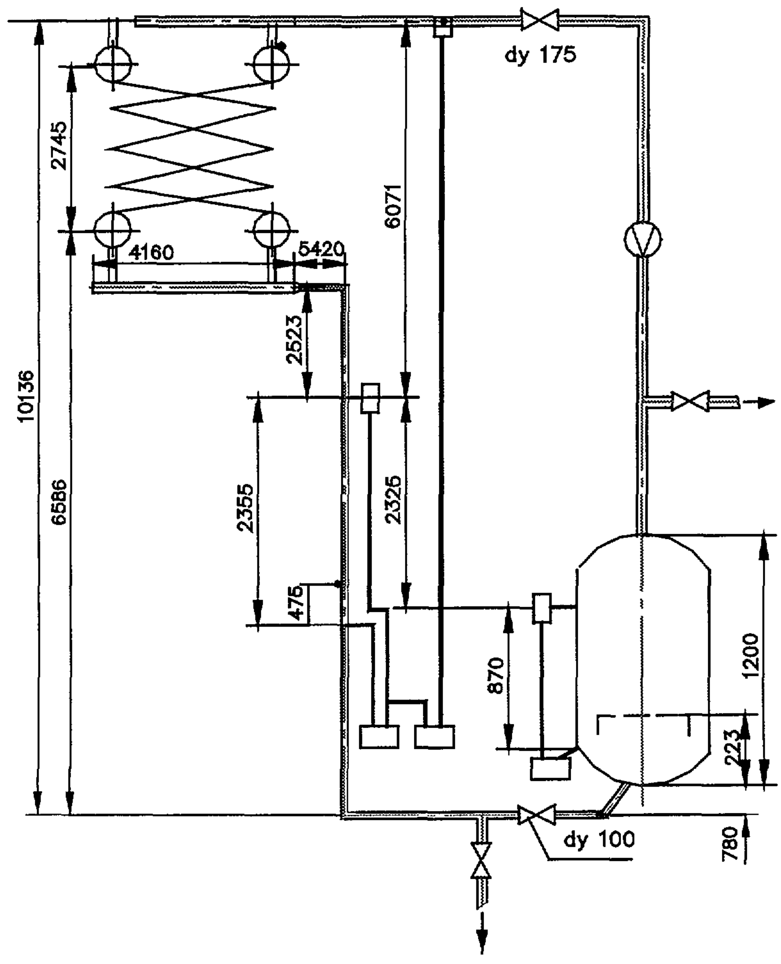

FIG. 7. Steam-condensate circuit.

The investigations to define the heat losses in hot standby condition with the closed air gates have been performed from 07.04.92 to 16.02.93 (totally 30 experiments). In experiments, the steam pressure before the heat exchanger accounted from 0,6 MPa to 6,39 MPa, the steam temperature before the heat exchanger accounted from 159°C-280°C and external air temperature accounted from minus 8°C to plus 18°C. It was found that the direct recalculation of the test results to define the system heat losses results in a larger losses than it is anticipated by the SPOT design (less than 0,1% of the reactor rated power). It was also determined that temperature of the external surface of the heat exchanger thermal insulation exceeds significantly the design value 45°C. The reasons of these differences for the test rig have been found and the recommendations have been developed for the design modification of the SPOT system thermal insulation for WWER-1000/V-392.

To define the dependence of the thermal power removed by the heat exchanger against the external air temperature and steam pressure, the tests have been performed under stationary conditions in the range of external air temperature from –19°C to +30°C and steam pressure at the heat exchanger inlet from 0,54 MPa to 6,39 MPa. During the period from 27.03.92 to 15.02.93, thirty experiments have been performed under positive external air temperature and ten experiments under negative one; the air gates were open partially in eleven experiments to evaluate possibility of the SPOT power control. It was determined that the heat exchanger removes from 5 MW to 7.4 MW at the pressure 6,3 MPa and the external air temperature from plus 30°C to minus 19°C. It was confirmed that the heat exchanger design geometrical characteristics ensure necessary power of the heat exchanger at the maximum design temperature 50°C and fully open air gates. For evaluation of the removed power control by means of partial closing of the lower air gates, two series of the experiments have been performed at the external air temperature plus 18-19°C and minus 11-16°C and steam pressure 6,3 MPa. The gates have been deviated on 10, 20 and 30 degrees in relation to horizontal. It was determined that the design configuration of the air gates ensures the power control of the heat exchanger under turn angles 0-40 degrees, after that the regulating capacity is lost. For example, the heat exchanger power at the turn angle of the gates 10 degrees is approximately 2,5 times less than under fully open gates, and at the angle 30 degrees the power is only approximately 15% less than the power at fully open gates.

To investigate the influence of the non-condesable gas on the heat exchanger operation, two experiments have been performed with supply of the different nitrogen quantity to the steam generator model. The gas quantity in these experiments was approximately 5,5 and 1,3 times more than quantity which corresponds to the real conditions of the reactor plant operation. A number of effects related to the nitrogen supply have been noted. In particular, the condensate temperature is reduced concerning saturation temperature due to decreasing of the partial steam pressure. The nitrogen is distributed on the collectors unequally, according to their location concerning steam pipeline. Discharge of a part of nitrogen with condensate flow was observed. The experiments have been performed with putting the heat exchanger into operation at the fresh steam supply and at the different quantity of nitrogen injected. It was determined that in all cases the heat exchanger power is stabilized in 40-50 seconds, and the value of the stationary power does not depend on quantity of the nitrogen injected and is only determined by the external air temperature. The tests performed have shown that the nitrogen is discharged gradually with the condensate. So, at the injection of the nitrogen quantity, which exceeds possible injection in real conditions about 5,5 times, the nitrogen has practically discharged completely from the heat exchanger in 10 minutes after the air gates opening.

The investigation of the stability of the SPOT heat exchangers parallel operation for the different loops has been performed on TDU-1 facility in one of the institutes of the Science Academy of Belorussia. The experiments have been performed on the two-loop three-circuit test rig by 1 MW power. The steam generator and the SPOT air heat exchanger-condenser were modelled in each loop. It has allowed to model operation of two train of the SPOT in parallel at the different loading of the heat exchangers. The experiments have been performed at the air temperature from plus 5°C to plus 31°C and at the steam pressure from 0,6 MPa to 5,4 MPa. The experiments have demonstrated steady operation of the heat exchangers, the variations of the condensate and heat transfer tubes temperatures were not noted. In the same institute the experiments have been performed on the SPOT-2 facility, which models the circulation circuit WWER-1000 and the SPOT circuit in scale 1:5500 in relation to the power and ensures the hydraulic similarity and real difference of the equipment elevations. These experiments have confirmed the possibility of the long-time passive heat removal in case of the main circulation pipeline rupture and station blackout.

The containment model have been developed and constructed in scale 1:80 for the experimental investigation of the possible influence of the wind on the SPOT effectiveness. The investigations have been performed for the wind speed from 0 to 90 m/s (from the calm atmosphere to hurricane) at the wind direction from 0 to 360 degrees in relation to the reactor building axes. These experiments have shown the absence of the circulation reversal in the exhaust air stacks and have confirmed the design solution correctness, at which the SPOT trains are connected by the common inlet collector and the common outlet collector with the deflectors.

5. CONCLUSION

Broad objectives for advanced nuclear power plants have been documented [7] by the International Atomic Energy Agency. With regard to the safety enhancements, this document states that the plant design should seek to take the maximum, feasible advantage of inherent safety features, and efforts should be made to utilize passive safety systems to the extent that they can be shown as reliable and cost effective as active systems for the same function. Following these recommendations, a reasonable balance of active and passive systems based on the weighing of their advantages and disadvantages with regard to the designated functions, overall plant safety, and construction and operation costs has been found in WWER-1000/V-392 and WWER-640/V-407 designs.

A number of the relatively innovative passive safety means are used in new Russian plant designs with V-392 and V-407 reactors to fulfil the fundamental safety functions, such as reactivity control, fuel cooling and radioactivity confinement. Their implementation allowed to significantly increase the power plant safety in terms of the expected severe core damage and excessive radioactivity release frequencies. For example, the estimated core melt frequency of WWER-1000/V-392 is three orders of magnitude less than for the operating power units with WWER-1000/V-320 reactor.

As the sufficient operational experience for some passive systems and components is absent, the extensive experimental investigation and tests have been carried out or planned to prove the functioning of these systems under plant conditions. In particular, the experiments are already performed for residual heat removal system (SPOT), quick boron supply system, the system to keep the rarefied atmosphere in the containment wall's gap, emergency core cooling system, and some others.

The experimental investigations and tests performed have confirmed the design functioning of the passive safety means proposed and allowed to optimize their general configuration and initiating signals. These investigations have also created the necessary experimental data base for the modeling of the passive safety means by the system thermohydraulic codes. Further investigations are being planned for additional verification of the passive safety systems and for the optimization of their design.

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