Experimental Investigation of Operation of VVER Steam Generator in Condensation Mode in the Event of the Severe Accident

Andrey Morozov

Institute for Physics and Power Engineering by A.I. Leypunsky, 1 Bondarenko sq. Obninsk, 249033, Russia

sas@ippe.ru

ABSTRACT

For new Russian nuclear power plants with VVER-1200 reactor in the event of a beyond design basis accident, provision is made for the use of passive safety systems for necessary core cooling. These safety systems include the passive heat removal system (PHRS). In the case of leakage in the primary circuit this system assures the transition of steam generators (SG) to operation in the mode of condensation of the primary circuit steam. As a result, the condensate from SG arrives at the core providing its additional cooling.

To investigate the condensation mode of VVER SG operation, a large scale HA2M-SG test facility was constructed. The rig incorporates: buffer tank, SG model with scale is 1:46, PHRS heat exchanger.

Experiments at the test facility have been performed to investigate condensation mode of operation of SG model at the pressure 0.4 MPa, correspond to VVER reactor pressure at the last stage of the beyond design basis accident. The report presents the test procedure and the basic obtained test results.

1 INTRODUCTION

The principle of technological diversity including the combination of active and passive safety systems is applied for achievement of qualitatively new level of safety in the NPP project with VVER-1200 reactor (reactor plant V-392M). To a number of new passive systems stipulated by the project, the system of passive core reflooding from hydro-accumulators of the second stage (HA-2) and passive heat removal system (PHRS) from steam generators are related.

These systems can take part in elimination of wide class of accident modes, such as: ensuring heat removal from accident core, maintenance of coolant reserve in the reactor; integrity of containment and restriction of failure radiological consequences in the project limits. Thus, the rules of operation of these systems are defined by the accident specifics.

So, in spite of active safety systems refusal, at a leakless circuit of reactor plant (RP) heat removal from the fuel in the core to environment can be carried out by the PHRS system practically during an unlimited period of time. In the process of heat exchange between the coolant flowing in the primary circuit inside tube bundle and water in intertube space, a steam generation in the secondary circuit of SG takes place. Due to the natural convection in steam-condensate circuit of PHRS system, steam enters the air-cooled heat exchangers, mounted on the outside surface of containment in special air conduits. Steam is condensed, giving back its energy to air. The formed condensate due to gravity flows back into SG tube space.

At beyond design basis accident associated with breakage of RP primary circuit and refusal of ECCS (emergency core cooling system) active part, the joint operation of PHRS and HA-2 systems is supposed. The loss of coolant, going away by steam and saturated water through the rapture, is compensated by injection of water from HA-2 system with decreasing in steps flowrate after decrease
of pressure in the reactor below the threshold meaning of 1.6 MPa. After the PHRS system transition into the regime of cooling (by opening the slide valves on the air path of the PHRS heat exchangers), a cooling of the SG secondary circuit starts up. After pressure decrease in the secondary circuit below the pressure in the primary circuit, steam generators enter into the regime of condensation of steam coming to the SG tube bundle from the reactor. Thus, the PHRS system capacity is determined by the pressure in SG and temperature of atmospheric air. The formed condensate through the non-damaged loops comes back to the reactor. The HA-2 system operation together with the condensate make-up from SG compensates loss of coolant in the reactor and, thus, secures the fuel cooling.

The analysis of various failure schemes has shown, that the accident associated with the instant break-up of reactor coolant pipe (RCP) with nominal diameter of 850 mm, together with simultaneous break of a.c. power supply and, as a result, loss of the operation ability of active safety systems is crucial for operation of HA-2 and PHRS passive safety systems. According to the results of theoretical analysis of the processes taking place in the VVER-1200 reactor with reactor plant V-392M, using a code SCDAP/RELAP5/MOD3.3, in case of a such type accident, the pressure in the primary circuit drops sharply because of the significant coolant leakage through the rapture up to the level of 0.4 MPa already at ~100-th second after the accident beginning [1]. As a result, the pressure in secondary circuit exceeds that in the core. On the 30-th second after the accident an opening of the PHRS system gates, established before and after heat exchangers, takes place. In the draught shafts the circulation of atmospheric air develops that leads, as a result, to the intensive increase of the PHRS system heat exchangers effectiveness, that results in the beginning of SG secondary circuit cooling.

As a sequence, approximately at the 3000-th second after the accident, the pressures in the both primary and secondary circuits of the reactor facility are equalized at a level of 0.3-0.4 MPa and steam generators enter the condensing operation mode. The water supply to the reactor from HA-2 system together with condensate flow from SG provides a reliable cooling of fuel in the core during the first twenty four hours after accident.

An operation of VVER reactor steam generator in off-design regime of steam condensation demands additional study. For this purpose in the State Scientific Center of the Russian Federation – IPPE a large-scale thermal hydraulic test rig HA2M-SG (Hydro Accumulators of 2nd stage Modernized and Steam Generator) was constructed.

2 EXPERIMENTAL FACILITY

A HA2M-SG is a large-scale experimental facility intended for the complex research of passive safety system operation for the new generation VVER reactor [2]. This facility includes a VVER reactor steam generator model, tank-accumulator with a system of steam supply from thermo-electric power plant, and simulator of PHRS system heat exchanger cooled by service water. All the facility systems are connected with the pipelines and equipped with valve armature. The altitude marks of the facility are the same as for the real project design. For the thermal losses to be reduced, the equipment and technological pipelines thermally insulated. The basic parameters of the facility are presented in Table 1.

A SG model is a recuperative heat exchanger consisting of vertical knock-down body, two vertical collectors (cold and hot ones) and integrated into their pipe bundle consisting of horizontally spaced helical pipes.

The length of heat exchanging pipe is equal to the average length of natural SG pipe. The distance between extreme pipe rows of the SG model corresponds to that of PGV-1000 steam generator, as well. For the primary circuit coolant drainage from pipe bundle to be secured at the accident situations, the helicoids are made with some slope relative to the collectors with altitude difference of 20 mm.

An arrangement of base equipment of the HA2M-SG test facility is shown in Figure 1, and its schematic diagram is presented in Figure 2.
### Table 1: Basic parameters of the HA2M-SG test facility

<table>
<thead>
<tr>
<th>Name</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Working medium</td>
<td>Water, steam, steam-water mixture</td>
</tr>
<tr>
<td>Maximum pressure, MPa</td>
<td>1.6</td>
</tr>
<tr>
<td>Maximum temperature, °C</td>
<td>200</td>
</tr>
<tr>
<td>Basic Equipment of the Facility</td>
<td></td>
</tr>
<tr>
<td>Steam Generator Model</td>
<td></td>
</tr>
<tr>
<td>Scale</td>
<td>1/46</td>
</tr>
<tr>
<td>Maximum power, MW</td>
<td>1.0</td>
</tr>
<tr>
<td>Number of pipes (rows)</td>
<td>248 (62)</td>
</tr>
<tr>
<td>Pipe diameter, mm</td>
<td>$16 \times 1.5$</td>
</tr>
<tr>
<td>Pipe length, m</td>
<td>10.19</td>
</tr>
<tr>
<td>Vertical step of pipes, mm</td>
<td>36.5</td>
</tr>
<tr>
<td>Material of tube bundle</td>
<td>Stainless steel</td>
</tr>
<tr>
<td>PHRS System Heat Exchanger Simulator</td>
<td></td>
</tr>
<tr>
<td>Maximum power, kW</td>
<td>800</td>
</tr>
<tr>
<td>Cooling medium</td>
<td>Service water</td>
</tr>
<tr>
<td>Tank–Accumulator of Steam</td>
<td></td>
</tr>
<tr>
<td>Volume, cu m</td>
<td>16</td>
</tr>
</tbody>
</table>

*Figure 1: Arrangement of base equipment of the HA2M-SG test facility.*
Figure 2: Schematic diagram of the HA2M-SG test facility. 1 – SG model, 2 – PHRS heat exchanger simulator, 3 – tank-accumulator of steam.

The instrumentation used at the test facility allowed the following basic parameters to be registered:

– pressures in the primary circuit, in the tank-accumulator, and at the SG model input;
– pressures in the secondary circuit, at the SG model output, and at the PHRS simulator input;
– temperatures of primary circuit medium, in the tank-accumulator, and at the SG model input;
– temperatures of secondary circuit medium, at the input/output of SG model, and at the input/output of the heat exchanger of PHRS system heat exchanger simulator;
– temperatures of service water at the input/output of PHRS system heat exchanger simulator pipelines;
– flow rate of primary circuit medium (steam) at the SG model input;
– flow rate of service water at the input into the PHRS system heat exchanger simulator;
– water levels in the tank-accumulator, in the SG model, in primary circuit water seal, and in downcomer region of secondary circuit pipeline.

The scheme of gauges and measuring devices layout of the HA2M-SG test facility are shown in Figure 3.

The pressures along the facility circuit were measured by piezoresistive transducers METRAN-100-DI type. Measurements of water levels were carried out by hydrostatic method using transducers of pressure difference METRAN-100-DD type. For the temperature measurement the cable chromel-copel thermocouples with a diameter of 1.0 mm were used. The service water flow rate through the PHRS system heat exchanger simulator was determined by the measurement of pressure drop at the orifice flowmeter combined with the METRAN-100-DD transducer. The frequency of scanning the measuring channels was determined by computer system at a level of 1 Hz.

The measurement of steam flow rate at the SG model input was carried out by the vortex steam flow rate counter METRAN-332 type with frequency of 0.125 Hz.

The SG model was equipped with more than 100 thermocouples enabling the temperatures of primary and secondary facility circuits to be controlled in both pipe bundle and in the intertube space of steam generator.

A thermocouple layout diagram at the SG model is presented in Figure 4.
Figure 3: Scheme of instrumentation layout of the HA2M-SG test facility.

Figure 4: Thermocouple layout diagram at the SG model.
The thermocouples $T_{16} \div T_{30}$ measured a temperature of SG body outside surface that permits the thermal losses into atmospheric air to be estimated. The superficial thermocouples such as chromel-copel type $2 \times 0.5$ mm covered with silica protective shell were used.

By the thermocouples $T_{32} \div T_{35}$, a measurement of primary circuit medium temperature was carried out along the height of a "cold" collector of the SG model.

The measurement of temperature of secondary circuit medium was carried out in 10 sections along its height by the thermocouples $T_{36} \div T_{55}$. Each of these thermocouples was fixed inside the SG body with a help of two rigidly mounted vertical bars. The cable chromel-alumel thermocouples with diameter of 1.5 mm were used.

The measurement of temperature of the steam generating pipes external surface was carried out in 7 rows along the pipe bundle height by thermocouples $T_{56} \div T_{103}$ welded to the walls of upper and lower generatrix lines of pipes in three sections along the length.

3 THE RESULTS OF EXPERIMENTS

The experiments on study of SG model condensation capacity were carried out on the test facility. These experiments were realized at steam pressure in the primary circuit of $0.39 \pm 0.01$ MPa. The condensing power ranged from 160 up to 310 kW.

On the basis of treatment of the experimental results, obtained at the HA2M-SG facility, the correlation between SG model condensation ability and pressure in the secondary circuit presented in Figure 5 was obtained. Mean condensing power of the steam generator was defined by the steam flow rate at the SG model input, under condition of its complete condensation in pipe bundle and, as a sequence, absence of its exit through the water seal of primary circuit. The level of liquid in water seal was defined by the level gauge readings.

Pressure in the HA2M-SG test facility circuits for the test 016-4-1/12 ($P_1 = 0.388$ MPa, $N_{con} = 248$ kW) after achievement of steady-state conditions are presented in Figure 6.

Figure 5: Correlation between SG model condensation capacity and pressure in the secondary circuit.
Temperature fields of water in intertube space of SG model over the height of SG pipe bundle for the same test are shown in Figure 7. The temperature difference between the top and bottom rows of steam generator model pipe bundle did not exceed 1.5 °C what said about uniformity of water temperature field in the secondary circuit of SG in the process of the experiments carrying out.

During the experiments, the features of processes taking place at SG operation under given conditions were determined. It was established, that the temperature difference determining the intensity of heat exchange between primary circuit (condensing steam) and secondary circuit (boiling water) is insignificant and does not exceed 3 °C. The record of temperature fields over the height of steam generator pipe bundle for the test 016-4-1/12 is presented in Figure 8.

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**Figure 6:** Pressure in the HA2M-SG test facility circuits for the test 016-4-1/12 $(P_t=0.388 \text{ MPa, } N_{\text{con}}=248 \text{ kW})$.

**Figure 7:** Temperature distribution in intertube space for the test 016-4-1/12; $h$ – height from SG bottom.
Figure 8: Temperature fields over the height of SG pipe bundle for the test 016-4-1/12.

Such small temperature differences stipulate for the peculiarities of the heat exchange processes taking place in the SG model and values of heat fluxes, as well. The dependence of heat exchange coefficient upon heat flux value for the experiments in the secondary circuit pressure of $P_2=0.34\pm0.02$ MPa is shown in Figure 9.

Figure 9: Correlation between heat exchange coefficient and heat flux ($P_2=0.34\pm0.02$ MPa).
It is seen from these figures, that the value of heat flux under boiling condition does not exceed 2500 W/Sq m and the value of heat exchange coefficient is less than 550 W/(sq m·K).

On the base of the experimental data treatment, the empirical correlation Eq (1) for the heat exchange coefficient was obtained for water pool boiling on the multi-row horizontal pipe bundles under natural circulation of heating steam, applied at the pressure 0.34±0.02 MPa:

\[ \alpha = 10.9q^{0.5}, \]  

where: \( \alpha \) – heat exchange coefficient, W/(sq m·K); \( q \) – heat flux, W/sq m.

The character of this correlation allows making a conclusion about features of heat exchanging processes taking place in SG model. The dependence of heat exchange coefficient upon the heat flux in the power of 0.5 shows, that the regime studied is an intermediate one between single-phase convection (\( \alpha \sim q^{0.3} \)) and developed nucleate boiling (\( \alpha \sim q^{0.7} \)). Thus, in this case, a non-developed nucleate boiling takes place, caused by the bundle geometry (62 horizontal rows) and natural steam circulation in the primary circuit.

The comparison between experimental and calculated, using Eq. (1), heat exchange coefficients is shown in Figure 10. It is seen from this figure, that the discrepancy between values does not exceed 5%.

![Figure 10: Experimental vs. calculated heat exchange coefficient.](image)

4 CONCLUSIONS

The experiments carried out on the large-scale thermal-hydraulic test facility HA2M-SG have proved operability of SG model for the reactor VVER in condensing mode.

Experiments were carried out at pressure in the primary circuit of 0.4 MPa, corresponding to the pressure in reactor plant in the case of reactor coolant pipe break-up. As a result, the dependence of SG condensing power on the pressure in secondary circuit was obtained.

The features of processes of heat exchange taking place under SG operation in steam condensing mode was elucidated. The empirical correlation describing processes of water pool boiling on horizontal pipe bundles was received.
REFERENCES

