Computational modeling of boric acid droplet entrainment during water-water energetic reactor operation in emergency boiling mode

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Abstract. The results of a computational modeling of the processes of droplet entrainment of boric acid by a vapor stream in the event of an emergency at a water-water energetic reactor (WWER) nuclear power plant associated with a guillotine rupture of the main circulation pipeline and the loss of all alternating current sources are presented in the article. The dependence received in the course of computational modeling makes it possible to estimate the maximum size of the droplets carried away by the steam stream during the emergency process in the WWER reactor unit during the operation of the complex of passive safety systems in the pressure range 0.2-0.5 MPa. It was established that to the end of 72 hours of an accident the maximum size of droplets carried away from reactor is 0.11 mm.

1. Introduction

To ensure the reliable operation of modern nuclear power plants, a set of systems ensures the normal operation of the plant. A special role is given to safety systems. There are such systems as active (dependent on external energy sources) and passive, based on only the fundamental laws of nature [1].

In the event of a possible accident at nuclear power plants (NPPs) with a rupture of the main circulation pipeline, the coolant will leak from the core. Modern NPPs with a reactor installation WWER have a passive core flooding system of hydraulic accumulators (HA) of the first (HA-1 system), second (HA-2 system), and third (HA-3 system) stages (Figure 1). It consists of vessels of different volumes filled with boric acid solutions with a concentration of 16-20 g/kg [2]. Such a system is triggered when the pressure drops in the core below a certain level. The HA-1 system is designed for a core cooling immediately after an accident. The second stage of hydraulic tanks containing a solution of boric acid is capable of cooling the reactor within 24 hours after the start of the accident. The HA-3 system started in operation after completing the coolant flow from the HA-2 system's tanks. It can provide heat removal from the reactor over the next 48 hours. Thus, the systems of the passive core flooding provide an emergency cooling down three days after the accident [3].

Hydraulic tanks of this system contain a solution of boric acid (H₃BO₃), to maintain the reactor in a subcritical state. However, in the case of prolonged cooling of the core, there are problems with the crystallization of boric acid. The main reasons for this are: the volume of H₃BO₃ entering the core from the hydraulic tanks system within three days after the start of the accident; boiling of the coolant and...
low content of boric acid in the vapor phase. $\text{H}_3\text{BO}_3$ can be deposited on the surface of the fuel elements and impair heat transfer from their surface [4].

A similar problem could occur at PWR reactors during the recirculation phase of Loss of Coolant Accident (LOCA) [5]. During the LOCA, water with boric acid is injected by the Emergency Core Cooling System into the reactor vessel to remove the decay heat from the core. Injection continues into the cold legs during the safety injection phase. After the sump switchover, when water with boric acid from the containment sump is recirculated through the primary circuit after Refueling Water Storage Tank liquid decreasing – the so-called recirculation phase. This phase could last for several days to maintain adequate reactor cooling. The continuous vaporization of the water in the core during this phase could increase the $\text{H}_3\text{BO}_3$ concentration in the reactor, where the boric acid could reach the solubility limit and precipitates.

The removal of boric acid from the core with steam or due to drip entrainment can significantly reduce the risk of its crystallization in the core. Therefore, the study of processes associated with the presence of boric acid and a boiling core is essential for the calculation of emergency processes at nuclear power plants with new generation WWER reactors.

A review of performed experimental studies of $\text{H}_3\text{BO}_3$ solubility in vapor [6-8] showed that the results presented in the literature do not completely cover the range of main coolant parameters in case of an accident on WWER nuclear reactor. Thus, experiments dedicated to the determination of the solubility of $\text{H}_3\text{BO}_3$ in vapor in a wide range of pressures and boric acid concentrations have been done at Leypunsky Institute for Physics and Power Engineering [9]. The relation between the distribution coefficient of $\text{H}_3\text{BO}_3$ between liquid and vapor phases on its concentration in the reactor model has been experimentally received.

Experimental studies of processes of boric acid entrainment were conducted by various investigators. For example, in [10, 11], the results of experiments on the study of the process of long-term boiling of boric acid carried out in Finland on the REWET-II, VEERA, and modified VEERA facilities are presented. These facilities include different regions of the reactor vessel and steam generator. The volumetric scale of the REWET-II facility is 1:2333 relative to the WWER-440 reactor, and the height scale is 1:1. The main purpose of the research was to measure the concentration of $\text{H}_3\text{BO}_3$ in the core, reactor’s pressure chamber, and in the outgoing vapor. In the tests have been shown that in the reactors simulator, it is possible to achieve very high concentrations of boric acid, which could lead to crystallization and blocking the circulation of the coolant in the reactor.

Also, it was shown in [11] that droplet entrainment had affected the rate of increase of the boric acid concentration in the reactor. In the experiments without a separator device to control a droplet entrainment, the concentration of $\text{H}_3\text{BO}_3$ in the reactor’s core simulator reached the constant level below the saturation point, and crystallization did not occur. The analysis showed that the reason for this was

![Figure 1. WWER passive core flooding system: 1 – reactor vessel, 2 – HA-1, 3 – HA-2, 4 – HA-3, 5 – steam generator, 6 – spring check valve, 7 – main circulation pump.](image)
that the vapor carried away the same amount of boric acid that came with the water into the reactor. The concentration of $\text{H}_3\text{BO}_3$ reached the saturation point after 14 hours of the experiment with a separator on the vapor removal line.

A massive program of experimental studies of droplet entrainment and separation of vapor was done in the USSR under the study of the efficiency of steam turbines for thermal power plants. The formulas characterizing the motion of a two-phase flow received based on the analysis of these experiments are shown in [12].

In articles [13, 14], the results of computational and experimental studies of $\text{H}_3\text{BO}_3$ droplet entrainment for the reactor CAP-1400 in China are presented. The tests were carried out on the FATE test facility, which has a scaling ratio of 1:1. The facility has the models of the fourth stage Automatic Depressurization System (ADS-4) pipelines and hot leg of the main circuit made of transparent plastic. The tests were carried out at a pressure of 0.1 MPa. The processes of entrainment of highly concentrated boric acid solution with vapor through the tube of the ADS-4 system connected to the hot leg of the main circuit of the reactor were investigated. Boric acid solutions with various additives were used in the tests. As a result of the studies, it was concluded that the capture of $\text{H}_3\text{BO}_3$ droplets was not as intense as when using pure deionized distilled water. These results mean that the presence of boric acid in the coolant has a significant impact on the process of droplets entrainment in case of an accident.

So, in the result of the review of publications about boric acid droplet entrainment studies, it can be concluded that the experiments were conducted at the main parameters not equal for a loss of coolant accident at Nuclear power plant with WWER in case operation of passive core cooling systems. These results imply the need for computational and experimental studies of the entrainment process of boric acid droplets during WWER operation in accident boiling mode.

2. Modeling of changes of parameters of the reactor in case of an accident

During an accident with a rupture of the main circulation circuit, the pressure in the reactor rapidly decreases, the effluent coolant instantly boils, evaporates, and enters the containment volume. Almost all boric acid entering the reactor from the HA-1 system in the first minutes after the accident is also poured into the containment and does not take part in mass transfer processes in the core. Therefore, for the simulation, the time range after 1,000 seconds accident was selected, when the pressure in the primary circuit begins to stabilize at a new level.

To assess the rate of evaporation of the coolant and determine the speed and flow rate of steam, it is necessary to know the power of heat generation in the core after the accident. Given that the rated power of the WWER-1200 reactor is $N_0 = 3200$ MW, the law of residual energy release can be written as follows [15]:

$$N_{hr} = 6.5 \cdot 10^{-2} \cdot N_0 \cdot \tau_c^{-0.2} = 20.8 \cdot 10^7 \cdot \tau_c^{-0.2},$$

where $\tau_c$ is an accident time change.

The calculated range of the accident time change $\tau_c$ is in the range from 1,000 to 259,200 seconds, i.e., within three days of the emergency process.

Previous calculations showed that during the accident under consideration, the pressure in the core would be in the range of 0.2-0.5 MPa [16].

The volumetric flow rate of steam through the core can be determined from the following ratio:

$$Q' = \frac{N_{hr}}{r \cdot \rho'},$$

where $r$ is the heat of vaporization, J/kg; $\rho'$ is the vapor density, kg/m$^3$.

To assess the speed of the steam rising in the reactor vessel, it is necessary to know the vessel's passage section at the level of the hot nozzles' axis, taking into account the presence of internals. Knowing the inner diameter of the shell of the block of protective pipes, the number of pipes, and their diameters, we define the flow area for the steam generated in the core: $F = 4.46$ m$^2$.

Then the superficial steam velocity can be calculated by the formula:
Thus, dependencies were obtained that describe the main parameters during the operation of the reactor in a boiling mode during an accident with a rupture of the main circulation pipeline. This data allows determining the maximum droplet size that can be carried out of the reactor with a vapor stream.

3. Determination of the maximum diameter of the soaring droplets

The moisture content of the steam is determined by the number of droplets being thrown to the height where the steam outlet channels (hot reactor tubes) are located and transported by the steam stream from the steam space. At high altitudes of the vapor space, transportation has the main effect on the entrainment of droplet moisture. However, the ratio between the discharged and the transported moisture substantially depends on the flow rate (steam load). The height of the drop of transported moisture drops is not limited. Conventionally, it can be taken equal to the height at which the absolute velocity of the drops becomes equal to the velocity of the vapor. The further movement of the droplet is determined only by the dynamic effect of the vapor stream. Thus, the transported humidity of the steam does not depend on the height of the steam space. The height between the cold and hot nozzles of the WWER reactor exceeds 1 m, taking into account the boiling of the boric acid solution in the active zone at the level of cold nozzles, the steam humidity depends only on the transported humidity. To determine it, you need to know the size of the soaking drops.

In [17], it was noted that droplets are freely transferred by the flow when its superficial velocity is higher than the terminal velocity \( w_{\text{term}} \) – relative velocity of the droplet, at which the friction forces balance the weight of the droplet:

\[
\frac{4}{3} g \cdot \frac{d_{\text{drop}}}{\xi} \cdot \frac{\rho' - \rho^*}{\rho^*} \equiv w_{\text{term}}
\]

where \( \xi \) is the coefficient of friction; \( g \) is gravity acceleration, m/s\(^2\); \( d_{\text{drop}} \) is the diameter of the droplet, m; \( \rho^* \) is the density of saturated water, kg/m\(^3\).

If the terminal velocity is higher than the superficial vapor velocity and the height at which the drop is raised is less than the height of the vapor space, the drop falls back onto the evaporation mirror.

The dependence of the coefficient of friction on the Reynolds number lying in the range from 2 to 1,000 (this condition is satisfied in this case) can be found in [17].

\[
\xi(\text{Re}) = \frac{25.3}{\text{Re}^{0.6}}
\]

where \( \text{Re} = \frac{w_0^* \cdot d_{\text{drop}}}{\nu} \).

To receive expression for calculation of maximum diameter of soaring droplets \( d_{\text{drop}} \) we should equate the superficial steam velocity (3) to the terminal velocity (4), substituting equation (2) into (3) and formulas (5), (6) into (4). Using simple mathematical transformations, we obtain the dependence of the change in the size of the soaking droplets on the power of the residual energy release:

\[
d_{\text{drop}}(N_{hr}) = (N_{hr})^\frac{5}{8} \cdot C_1
\]

where

\[
C_1 = \left( \frac{25.3 \cdot \nu^{0.6}}{\frac{4}{3} g \cdot \frac{\rho' - \rho^*}{\rho^*}} \right)^\frac{4}{5} \cdot \frac{1}{(r \cdot \rho' \cdot F)^\frac{7}{8}}
\]

To obtain the dependence of the soaring droplet size on time in (7), we substitute the law of residual energy release (1):
Thus, we obtained two dependences (7) and (8), which allow us to determine the maximum size of the soaring droplets depending on the power of the residual energy release and time at constant pressure at primary circuit at the range 0.2-0.5 MPa (Figures 2 and 3).

\[ d_{\text{drop}}(\tau_c) = \left( \tau_c^{-0.2} \right)^{\frac{7}{8}} \cdot C_2 \]  

(8)

where \( C_2 = \left( \frac{25.3 \cdot v^{0.6}}{4 \cdot g \cdot \rho - \rho} \right) \left( \frac{6.5 \cdot 10^{-2} \cdot N_0}{r \cdot \rho \cdot F} \right)^{\frac{7}{8}} \)

4. Conclusion

In Leypunsky Institute for Physics and Power Engineering a calculated analysis of droplet entrainment process in the case of WWER loss of coolant accident has been performed. As a result of calculations the maximum size of liquid drops that can be carried out from WWER core with the flow of steam-water mixture into the steam generator and into the containment volume was obtained. The results can be used in the calculation of emergency processes in WWER reactors equipped with a complex of passive safety systems, when substantiating their safety.

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