ANALYSIS OF ACCIDENT SCENARIOS WITH HETEROGENEOUS BORON DILUTION IN THE COOLANT OF THE PRIMARY CIRCUIT


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Introduction

In European Union, the safety requirements to the design of the new Nuclear Power Plants (NPP) include a justification in the design that all accident sequences that could lead to early or large radioactive releases are practically eliminated with a high degree of confidence [1].

At international level, this requirement was first introduced by INSAG 10 [2] in 1996 and then detailed in 1999 by INSAG 12 [3].

The IAEA SSR-2/1 [4] states that the accident sequences with a large or early release can be considered to have been practically eliminated:

(1) if it is physically impossible for the accident sequence to occur or
(2) if the accident sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is ‘practically eliminated’. The design shall be such that for design extension conditions, protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures.

The list of the accident situations that are considered as to be “practically eliminated” could include:

- Accident sequences involving containment by-pass,
- Reactivity accident resulting from fast introduction of cold or deborated water,
- High pressure core melt situations,
- Global hydrogen detonations and steam explosions threatening the containment integrity,
- Fuel melting in the spent fuel pool.

It should be mentioned that Belarussian regulations do not include “practical elimination” approach into the safety justification. This raises couple of questions during regulatory safety review regarding the completeness of safety assessment.

In the frame of EC INSC project BE/RA/08 “Support and assistance to strengthen the capabilities of the Belarusian Nuclear Regulatory Authority MES/ Gosatomnadzor (GAN) in the field of licensing and supervision of construction of the Belarusian Nuclear Power Plant (NPP)”, European experts provided methodology and examples of application of “practical elimination” approach for safety justification of new NPP designs.

A Case Study on analysis of Heterogeneous boron dilution (HBD) accident sequences was proposed and performed by Belarussian partners and supported via EC INSC project.

The justification of “practical elimination” should be examined on a case-by-case basis, using deterministic considerations, complemented by a probabilistic assessment.

In pressurized water reactors of VVER type, boric acid is used as a soluble neutron absorber in coolant water of primary circuit. Main functions of boric acid are to compensate the reactivity changes caused by the fuel burnup and Xenon poisoning during normal operation and to provide the required sub-criticality of reactor shutdown during refueling, maintenance and repairing as well as under accident conditions.
An inadvertent dilution of primary coolant with deborated water leads to decreasing of boron concentration and an insertion of positive reactivity in the core. As a result, it may lead reactor into a critical state and in case of additional failures of safety systems could develop an accident situation up to the core damage.

In the frame of safety assessment of pressurized water reactors, two types of boron dilution could be considered:

- homogeneous boron dilution
- heterogeneous boron dilution (HBD)

In case of homogeneous dilution, the boron concentration remains roughly the same in the volume of primary system and gradually decreases in time.

Heterogeneous dilution of boric acid occurs when the low borated slug is formed in certain parts of the primary circuit and transported to the reactor core, while the boron concentration in the rest of the primary system is unchanged. The deborated water plug is injected locally into the volume of the core. As a result, the boric acid concentration in the latter becomes to be heterogeneously distributed over the volume. In this case, a rapid increase in reactivity could occur due to reduction or absence of a neutron absorber (boric acid) and low coolant temperature at the inlet of the core. Both of them may result in fuel rod damage.

In this study the term "deborated water" (DW) means water with a low or zero concentration of boron acid (H$_3$BO$_3$). It includes water used in primary and secondary circuits such as condensed water located in storage tanks; chemically demineralized water; coolant of intermediate cooling circuit; deborated water formed due to boiling and subsequent condensation during the accident; secondary circuit coolant; primary coolant at the end of reactor campaign.

Goals and tasks of the analysis of accident scenarios with heterogeneous boron dilution in the coolant of the primary circuit

Goals of this paper are to analyze the beyond design basis accident related to the heterogeneous dilution of boric acid in the reactor core of the AES-2006 design. The result of this analysis was:

- Description of possible scenarios of formation of DW plugs and definition of system configurations, failure of system elements and human errors that could lead to the plug formation;
- Description of scenarios for the DW plug transport to the core and definition of system configurations, failures of system elements and human errors leading to the plug injection into the core;
- Estimation of parameters of interest (pressure, temperature, speed and volume) associated with the maximum possible DW plug formation and transportation;
- Estimation of the main parameters of the primary coolant and the core at the moment of the DW slug injection into the reactor core;
- Development of event trees and fault trees using the RiskSpectrum code for selected HBD scenarios as a result of a deborated water injection into the core through the systems connected with the primary circuit;
- Quantification of the probability of realization of accident scenario identified;
- Development of fuel assembly models and full core models in the Serpent and DYN3D codes;
- Determination of the critical volume of injected water plugs in various scenarios, leading to a critical state of the reactor;
- Analysis of transient processes in the core, leading to exceeding design limits and lack of acceptance criteria for fuel.

Description of possible scenarios of formation and transport of diluted water slug to the reactor core

One of the possible HBD scenarios in emergency modes assumes the formation of a DW plug in the main coolant pump (MCP) pipeline head during an emergency boiling-condensation process in the primary circuit. The break of one of the four circulation loops of the main circulation pipeline
Ø850 with a two-way outflow of the coolant may be the initial event of occurrence of this emergency process.

A rapid drop in pressure in the primary circuit occurs due to a large leak Ø850 of one of four circulation loops of the main circulation circuit (MCC). The leak of the cold loop pipe is the most conservative variant for such accident for the following reason. The stock of remaining coolant in the primary circuit will be less than one for the leakage of any of the hot loops due to the lower coolant level established in the reactor.

The steam formed above the core and above the level established in the reactor will be pressurized into the hot parts of the circulation loops. Then the vapor enters the steam generator, where it will be condensed while cooling with the feed water entering SG. Finally, it accumulates as deborated water plug in U-like sections of pipelines of cold train of circulating loops and gradually injects the reactor [5]. The maximum possible volume of deborated water plug for the U-like section of the main circulation pipe is 4.0 m³.

The second possible scenario for the occurrence of an emergency process is a possibility of forming a DW plug upon the transportation of the secondary coolant to the primary circuit during the accident of type "Leak from the primary circuit to the second one". In such accident, the emergency protection system (EPS) is triggered and the power unit is stopped by the signals of the level increase in the steam generator and increasing activity in the secondary circuit. Following the EPS operation, the accelerated cooling of the reactor starts at a speed of 60 °C/h, and, accordingly, a decrease of the pressure in the primary circuit begins.

There is a significant amount of chemically demineralized deborated water in the damaged steam generator of the secondary circuit. The cross-flow of the deborated water into the primary and formation of a DW plug in U-like section of pipelines as well as its following entering the core might occur due to operator error in the accident management actions. Three types of such leaks are considered: a small leak Ø13 is a break of a collector tube in the steam generator, the mean (Ø13-100) and large (Ø100-850) leakages can be caused by the break of several collector tubes in the steam generator, the breakaway of the collector cover or the collector break at the inlet to the steam generator.

In addition, it is necessary to take into account the situation when an increase in the pressure in the emergency steam generator from the side of the secondary circuit may occur during cooling the reactor. This can be caused by the unauthorized inclusion of emergency make-up electric pump (EMEP), the non-inclusion of make-up electric pump (MEP), the inclusion of the auxiliary MEP, the non-opening of quick-acting pressure reducing plant of air discharge to the atmosphere (QPRP-A) during reducing the pressure in the primary and secondary circuit. All of the above causes can lead to the formation of a DW plug and its injection the reactor core.

The pressure of a deborated water plug for the scenario will correspond to the pressure in the secondary circuit and will be within the range of 2.5-8.8 MPa according to the conditions created in the primary and secondary circuits. The maximum possible volume of deborated water plug for the U-like section of the main circulation pipe is 4.0 m³.

The stage of formation of a DW plug in pipelines of ECCS HP, ECCS LP, ECCS HA is supposed to be present in one of the scenarios for heterogeneous boron dilution in the primary coolant [5, 6].

During the reactor power operating, the safety system of the ECCS does not operate and is in the standby mode (in a state of readiness in case of an accident of the first circuit coolant leak). ECCS HP, ECCS LP and ECCS HA are connected to primary circuit and isolated from it by means of check valves. In standby mode, the ECCS system's electrically driven pipe fittings are in the open position. Then a non-detactable small compensated leakage of the primary coolant into the pipelines of the ECCS LP system may occur due to the leakages of the check valves. The concentration of H₃BO₃ in the primary coolant is minimal at the end of the fuel campaign. In this case, the formation of a DW plug can occur in the ECCS HP and ECCS LP pipelines due to leakage of check valves.
The pressure of the deborated water of the ECCS LP will be 2.45 MPa. The pressure of the deborated water plug of the ECCS HP will be 7.9 MPa. The temperature of the plug corresponds to the temperature of the ECCS HP and ECCS LP pipeline during normal operation of the reactor unit and is conservatively assumed to be 27 °C. The maximum possible volume of deborated water plug in ECCS HP is 4.6 m³. The maximum possible volume of deborated water plug in ECCS LP is 8.0 m³.

In normal operating conditions of the reactor unit, including during power operation, a periodic check of hydraulic tightness of check valves and the efficiency of motor valves are performed. The water level is regulated by draining the borated water into the drainage system of the reactor building equipment and feeding a concentrated boric acid solution from the make-up and boron control system to the tanks and also controlled by monitoring sensors. Monitoring of $\text{H}_3\text{BO}_3$ concentration is not carried out in pipelines from HA to the connection with primary circuit. Filling with the primary coolant the HA ECCS pipelines from the points of their connection with the primary circuit to the accumulator becomes possible during the standby mode due to a small undiagnosed leakage of the ECCS check valves, and the motor valves of bypass lines of the check valves. The pressure of a probability will be equal to the pressure of the activation of the ECCS HA – 5.9 MPa. The temperature of the DW plug corresponds to the temperature of the ECCS HA pipelines under normal operation of the reactor – 27-50 °C. The maximum possible volume of probability plug for this scenario is 2.6 m³.

For all identified scenarios of the DW plug formation, an analysis of the possibilities of the DW plug transporting to the reactor core was made:

- Boiling-condensation process in the primary circuit. After the formation of the deborated water plug, the transport of plug to the reactor core is possible in two ways - natural circulation and unintended starting of the RCP with the casting of a DW plug into the core.
- Human errors during “Leakages from the primary circuit to the second circuit” accident management.

During the cooling down of the reactor installation an emergency speed of 60 °C/h, the pressure in the primary circuit decreases. Under normal accident management, the main task of the operator is to minimize the flow of coolant into the second circuit by equalizing the pressure between the circuits. However, due to the operator error in the accident management, it is possible to reduce the pressure in the primary circuit below the pressure in the second and formation of deborated water plug in the U-shaped section of the main circulation pipe. In the event of an unintended connection of the MCP of this loop, it is possible for the DW enters to the primary circuit.

- Introduction of a deborated water plug through connecting pipelines of ECCS LP, ECCS HP, ECCS HA in the case of accident with coolant leakage of the first circuit and the ECCS triggering.

**Analysis results of possible scenarios of formation and transport of diluted water slug to the reactor core**

In order to determine the probability of the deborated water plug formation in the ECCS HP, ECCS LP and ECCS HA, possible scenarios for filling the pipelines with diluted coolant as a result of check valve leaks and additional valve failures were investigated by modelling fault trees and event trees.

The example of fault tree fo ECCS HA is presented in Figure 1. Filling of pipelines with deborated water is possible in case of leakage of check valves JNG50(60,70,80)AA602, AA 601 and simultaneous leakages of motor valves JNG51 (61,71,81) AA101, AA102 or JNG52(62,72,82)AA101, AA102 on the drainage line. Failure probability of deborated water plug formation is $2.1 \cdot 10^{-10}$. 
The results of the consideration of scenarios for the DW plug formation in the ECCS HP, ECCS LP and ECCS HA show that the DW plug formation may occur as a result of valve failures and safety systems triggering in case of primary coolant leaks.

Total probability for all considered scenario for the formation of a DW plug in the ECCS is \(1.1 \cdot 10^{-8}\). The maximum deborated water plug volume, formatted in the ECCS LP system, is 8.0 m\(^3\).

The temperature of the formed deborated water plug is conservatively assumed to be equal to the minimum temperature in the reactor containment (27–50 °C), the pressure of the plug is taken to the pressure of the corresponding system.

The dominant probability is \(7.2 \cdot 10^{-8}\) for the scenario of the DW water formation in the pipelines of the passive part of the ECCS system.

When considering the scenarios for the formation of a DW plug in the pipeline of the ECCS, the Human Errors (HE) and the probability of Common Causes Failures (CCF) were not taken into account due to the absence of reliable data on equipment failures for the Belarusian NPP at the moment.

**General description of the used reactor core neutron-physical model**

To study the heterogeneous boron dilution phenomenon using a conservative approach to the analysis of emergency processes, it is necessary to determine initial conditions for the core that will lead to the most unfavorable consequences. For accident scenarios with heterogeneous boron dilution in the coolant of the primary circuit, it is assumed that the state of the reactor core at the beginning of the fuel load will be the most conservative.

A four-loop model of the core is considered. The reactor core is divided into four parts and initial thermohydraulic parameters are set for each of the four loops. We assume conservatively that a water slug passes through one loop and enter the ¼ core without mixing with the main coolant.
The neutron-physical calculations of the scenarios were performed using DYN3D reactor diffusion code [7]. It is a best estimate code for the three-dimensional simulation of steady states and transients’ processes in Light Water Reactors. DYN3D uses a three-dimensional core model for dynamic and depletion calculations in light water reactor cores with quadratic or hexagonal fuel assembly geometries and possesses its own thermal-hydraulics core model and thermomechanical fuel rod model. Two-group macroscopic cross sections library was calculated using the Monte-Carlo code Serpent.

When working with the Serpent code, a common version of the library of microscopic cross sections ENDF/B-VII, adapted for this code by its developer, was used.

For the fuel assemblies of the VVER-1200 reactor, state parameters included fuel temperature, moderator temperature (water), moderator density, boric acid concentration, fuel burnout, and the position of absorbing rods. For each type of fuel assembly, a library of macro sections was prepared, taking into account their dependence on all the parameters listed above.

For VVER reactor fuel assemblies, the state parameters include fuel temperature, coolant temperature, coolant density, boric acid concentration, fuel burnup as well as the position of control rods. For each type of fuel assembly, a cross section library was prepared, taking into account their dependence on all the parameters listed above.

2D Serpent model of a single fuel assembly (FA) in reflective boundary conditions used in Serpent code is shown as an example in Figure 2. Spacer grids are included in this FA model as an additional thickness of fuel rod cladding (it is possible due to the same material composition of the fuel rod cladding and spacer grid). Such consideration of the spacer grids in the model corresponds to the main physical factor - effective reduction of the coolant volume inside the fuel assembly due to the presence of spacer grids.

![Figure 2 – 2D model of a single fuel assemblies in Serpent code](image)

90° sector of the full core model is shown in Figure 3. This model was developed taking into account the detailed geometry of VVER-1200 reactor core design.
The axial reflector model is shown in Figure 4. Top and bottom axial reflectors consist of three homogenized layers [8]. They were modelled as 2 hexagonal nodes on both sides of the 3D model of any FA, periodic lateral and black axial boundary conditions were used. Ratio between material composition of each layer was calculated for TVS-2M fuel assembly.
Radial reflectors were modelled as FA-sized hexagons containing real geometry of baffle and vessel. The supercell method was used to calculate the macroscopic cross sections of the radial reflector. The radial reflectors were modeled in the form of hexagons in size equal to the size of fuel assemblies containing the actual geometry of the baffle and the reactor barrel. The supercell includes a reflector cell surrounded by so-called background cells, among which the fuel assembly must necessarily be present as a source of neutrons (see Figure 4). Given the 30° symmetry of the baffle for a VVER-type reactor, 5 types of radial reflectors can be modeled [8, 9]. We used a two-layer model of a radial reflector in the code Serpent. The second layer allows to use black boundary condition for the core in DYN3D model instead of set of albedo coefficients that could also depend on the parameters. Third layer is shown to be insignificant for the modeling [9,10]. In this paper, all five types of reflector were considered in the model of one supercell being the 90-degree sector of the reactor core shown in Figure 4.

3D full core model in Serpent code consists of 163 FA of 1st loading for AES-2006 project with radial and axial reflectors (see Figure 5).

![Figure 5 – 3D full core model in Serpent code](image)

To conduct neutron-physical calculations of accident scenarios with heterogeneous boron dilution in the coolant of the primary circuit, a complete cross-section library was created, including few-group cross sections for all types of fuel assemblies and for the reflector.

**Results of neutron-physical calculation for accident scenarios**

When analyzing the design basis accidents and abnormal operation mode, the following acceptance criteria should be demonstrated for reactor in operation, taking into account the deviations of the parameters for the worse and the principle of single failure in safety systems:

1) the highest fuel cladding temperature achieved in emergency conditions does not exceed 1200 °C;
2) fuel pellets do not melt even locally (temperature less than 2540 °C for burnt fuel and less than 2840 °C for fresh fuel);
3) the enthalpy of fuel averaged over the cross-section of the fuel pellet does not exceed 963 J/g for fresh fuel and fuel with depletion less than 50 MW*day/kgU and 691J/g for fuel with depletion greater than 50 MW*day/kgU in any cross-section of fuel element height;
4) the pressure in the primary circuit coolant system and the steam pipelines of the SG is lower than 115% of the design value;
5) the heat transfer crisis is not achieved with a 95% probability for the hottest fuel element (DNBR > 1.0);
6) the number of damaged fuel rods does not exceed 1% or 10% (depending on whether the design basis accidents or abnormal operation mode is considered) of the total number of fuel elements in the reactor core.

The acceptance criteria associated with the change in pressure in the primary and secondary circuits are not taken into account in the RIA analysis because all calculations are carried out using a three-dimensional neutron-physical code without coupling with thermohydraulic code.

Accident scenarios associated with leaks of coolant as a result of pipeline ruptures of various diameters and leaks from the first circuit to the second were under consideration. From the probabilistic safety analysis, it follows that the greatest amount of water and the probability of the realization of this event relate to the accident associated with the primary coolant leaks.

Below, the calculation results are given on the example of an accident scenario associated with small and medium coolant leaks as a result of rupture of the primary pipelines of equivalent diameter of 100 mm (DN 100) and 300 mm (DN 300), respectively.

In the first case an accident associated with small coolant leaks as a result of the pipeline rupture of the primary circuit with an equivalent diameter (DN) of 100 mm is considered. When the ECCS HP pump is activated, it is possible to get a deborated water plug into the reactor core.

To simulate this scenario, a core model was prepared in DYN3D code, in which the following input parameters: flow rate, pressure, coolant temperature and boric acid concentration, were set for ¼ part of the core.

A DW plug enters the quarter of the core. Conservative values for the reactivity coefficients and the efficiency of emergency protection at the nominal power level were taken for calculations.

The events chronology obtained is presented in Table 1.

<table>
<thead>
<tr>
<th>Time point, s</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>Leak DN 100 from the cold loop No. 4</td>
</tr>
<tr>
<td>6.2</td>
<td>Formation of the first signal for triggering the SCRAM (signal is skipped): By coincidence of signals: the pressure at the reactor outlet is less than 15.2 MPa and the reactor power is more than 75% ( N_{\text{nom}} ).</td>
</tr>
<tr>
<td>12.5</td>
<td>Formation of the second signal for triggering the SCRAM: Reduction in reserve to boiling (difference between saturation temperature and temperature in any of the hot cycles of MCT) is less than 8 °C.</td>
</tr>
<tr>
<td>12.6</td>
<td>Shutdown of all MCP and their mechanical run-out</td>
</tr>
<tr>
<td>14.5</td>
<td>The beginning of the SCRAM insertion</td>
</tr>
<tr>
<td>18.5</td>
<td>The end of the SCRAM insertion</td>
</tr>
<tr>
<td>105.0</td>
<td>Start of boron solution supply from high pressure pumps</td>
</tr>
<tr>
<td>110.0</td>
<td>The beginning of the DW plug movement at the entrance to the core</td>
</tr>
<tr>
<td>150.0</td>
<td>End of calculation</td>
</tr>
</tbody>
</table>

Due to the absence of a coupling of the neutron-physical/thermohydraulic codes, the temperature, pressure and mass flow dependencies were taken from the project [11]. The water plug temperature is taken equal to the main coolant temperature (this is not conservative assumption).

The calculation was carried out for different DW plug volumes entering in the core. To study the dynamics of the process, the following volume of plugs was taken: 0, 2, 3, 4, 6.5, 8 and 10 m³. From this, the minimum volume of the plug, which takes the reactor to a critical state, is found. For the case of passing a plug with a volume of 10 m³, the parameters for the conditions of non-
violation of the acceptance criteria for the given problem were checked. Below, main design parameters are presented in Figures 6 – 8.

Figure 6 – Reactivity during the transient for different DW plugs volumes.

Figure 7 – Neutron power during the transient for different DW plugs volumes.
From the calculations follows that in this scenario, with the plugs volume from 0 to 10 m$^3$, the reactor core does not reach the critical state, hence all the parameter values do not exceed the acceptance criteria presented above.

In the second case an accident associated with medium coolant leaks as a result of the pipeline rupture of the primary circuit with an equivalent diameter (DN) of 300 mm is considered. When the ECCS HP pump is activated, it is possible to get a deborated water plug into the reactor core.

The events chronology is presented in Table 2.

Table 2 – Chronology of the accident events associated with medium coolant leakages as a result of the pipeline rupture of the primary circuit with an equivalent diameter of 300 mm.

<table>
<thead>
<tr>
<th>Time point, s</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>Leak equivalent diameter DN 300 from the cold loop No. 4</td>
</tr>
<tr>
<td>14.0</td>
<td>Formation of the signal for triggering the SCRAM</td>
</tr>
<tr>
<td>16.0</td>
<td>The beginning of the SCRAM insertion</td>
</tr>
<tr>
<td>20.0</td>
<td>The end of the SCRAM insertion</td>
</tr>
<tr>
<td></td>
<td>Shutdown of all MCP and their mechanical run-out</td>
</tr>
<tr>
<td>35.0</td>
<td>Start of boron solution supply from high pressure pumps</td>
</tr>
<tr>
<td>40.1</td>
<td>The beginning of the DW plug movement at the entrance to the core</td>
</tr>
<tr>
<td>150.0</td>
<td>End of calculation</td>
</tr>
</tbody>
</table>

The results of calculations with plug volumes similar to the first case are shown in Figures 9-11.
Figure 9 – Reactivity during the transient for different DW plugs volumes.

Figure 10 – Neutron power during the transient for different DW plugs volumes.
From the calculations follows that in this scenario, with the plugs volume from 2 to 10 m$^3$, the reactor core reaches the critical state, which lasts less than 1 s. Analysis of the calculations results shows that all the parameters values do not exceed the acceptance criteria.

**Conclusions**

As a result of the study the accident scenarios was developed and correspondent neutronphysical and PSA level 1 models were built in order to calculate the behavior of most critical safety parameters and the probability of realization of accident scenarios.

The analysis performed shows that the main mechanisms for the formation of a DW plug are leaks through electric and manual valves, check valves, diaphragms, heat exchangers leaks, as well as operator errors in system configuration.

Analysis of beyond design basis accident related to the heterogeneous dilution of boric acid in the core of the VVER-1200 reactor within the framework of probabilistic analysis, the following tasks were performed:

- the analysis of HBD scenarios in accident transients with the possibility of accumulation of a DW plug in the discharge part of the pipeline of the main circulation pump during boiling-condensation accident transients in the primary circuit, as well as in the case of an accident "Leak from the primary circuit to the secondary one";
- an analysis of possible scenarios for the formation of a DW plug in normal operation and abnormal operation: the accumulation of the drainage water plug as a result of leakage of check valves in the connecting pipelines between the ECCS LP, ECCS HP, ECCS HA and the primary circuit pipelines. The temperature of the formed condensate plug is conservatively assumed to be equal to the minimum temperature under the container (27 °C), the plug pressure is equal to the triggering pressure of the corresponding systems;
- the analysis of scenarios of transportation of a DW plug to the reactor core with determination of system configuration, failure of system elements leading to delivering the plug into the reactor core was performed;
• the maximum possible volumes of deborated water plugs generated in the systems are determined. Maximum volume of DW plug is 8.0 m$^3$ that could be accumulated in the pipeline of ECCS LP system due to leakages of check valves;
• failure trees and event trees were modelling using the RiskSpectrum code for the selected HBD scenarios as a result of the arrival of deborated water through systems associated with the primary circuit;
• numerical values of probability of realization of the accident transients are calculated. Total probability for all considered scenario in normal operation is $11.4 \times 10^{-8}$. The dominant failure probability is $7.2 \times 10^{-8}$ for ECCS HA accident scenario: leakage of check valves and simultaneous leakage of motor valves on the drainage line.

At present, there is not enough data for the analysis of operator errors for the Belarusian NPP, therefore this analysis should be carried out as the operating documentation data will be available. Failures of equipment systems for a common cause in this work were not taken into account due to lack of relevant data.

To calculate the scenarios associated with the heterogeneous dilution of boric acid, we used a two-group library of macroscopic sections in the $I_{wqs} = 2$ formula format for the DYN3D code prepared in the Monte Carlo Serpent code based on the ENDF/B-VII library, adapted for this code by its developer. When calculating the diffusion coefficients, a transport correction is applied, which takes into account the anisotropy of scattering of neutrons by hydrogen.

The neutron-physical calculations of transient processes of accident scenarios with heterogeneous boron dilution in the reactor core coolant, carried out using the DYN3D code, taking into account a number of assumptions, show that a critical state is reached in the reactor. However, there is no excess of the acceptance criteria (fuel cladding temperature, fuel temperature, fuel enthalpy averaged over the fuel pellet cross section) used in neutron-physical analysis of accidents caused by heterogeneous boron dilution in the primary circuit coolant.

As the result of the study several open issues were identified:
1) Consideration of human actions and CCF in PSA models;
2) Need for detailed analysis of the accident scenarios for heterogeneous boron dilution in the coolant using the DYN3D/ATHLET code coupling;
3) Uncertainty and sensitivity analysis for transient processes in the reactor core for HBD scenarios.

It is planned to continue the study of scenarios associated with heterogeneous boron dilution to solve open questions.

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