pressure waves in the primary circuit of the VVER-1000 reactor at instant stop of the MCP

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Abstract. The initial stage of the accident caused by the instantaneous stop of the main circulation pump (MCP) of the first loop of the VVER-1000 reactor plant (RP) is considered. The appearance of pressure waves and their propagation in the equipment of the primary circuit of the RP is shown. For the calculation, the parameters of the typical RP V-320 (VVER-1000), in particular, the 3rd unit of the Kalinin NPP, are used. All initial data for the calculation is obtained from the international standard problem Kalinin-3 [1-2]. Calculations were carried out using the computational best estimated code ATHLET [3], developed the society for reactor safety (Gesellschaft für Anlagen- und Reaktorsicherheit-GRS), Germany and certified in Russia for use in calculations to justify the safety of reactors with water coolant [4]. In case of an accident, the failure of the load-off and reactor power controller (LRPC), automatic reactor power controller (APC), warning protection systems is assumed. Emergency protection reactor (PR-1) only works on the 2nd signal. The considered emergency situation is included in the list of mandatory reports for different types in the justification of VVER safety [5]. The obtained results can be used as boundary conditions for the analysis of equipment strength in this transition process.

1. Brief description of the calculation code.
The thermohydraulic system code ATHLET (Analysis of Thermal-hydraulics of LEaks and Transitions) was developed in Gesellschaft für Anlagen- und Reaktorsicherheit (GRS mbH) and was originally intended for analysis of the entire spectrum of leak and transient analyses in PWR and BWR reactors. However, experience with it has shown that it can be successfully used to the full extent for Russian reactors such as VVER and RBMK. ATHLET consists of several basic modules that allow describe different phenomena in the behavior of thermal hydraulic systems: thermal hydraulic module (TFD), heat exchange and thermal conductivity module (HECU), neutron-kinetic module (NEUKIN) to describe point and one-dimensional kinetics, module to describe the operation of the equipment (GCSM) and fully implicit module for numerical integration (FEBE). Other independent modules can be connected via the main interface.

The TFD module is based on the use of five or six equations (the General equation for the moment of motion of the mixture with the drift flow) to describe the two-fluid model. In addition to light water, the module allows calculations with other types of reactor coolants, such as: heavy water, liquid metals (lead, lead-bismuth, sodium), gases (helium), simulate the behavior of non-condensable gases (air, nitrogen, argon, hydrogen, helium, oxygen, user gas-when the user sets its
properties), nitrogen dissolution and describe the transport of boron in systems with light and heavy water.

The plant cooling system is modeled by connection of basic thermodynamic objects (TFO), besides, there is a special possibility for calculation of cross flows in system of the parallel connected channels. Quite widely used capabilities of the ATHLET code for linking with various three-dimensional neutron-physical programs in the calculation of the spatial distribution of energy release fields and the spatial distribution of the coolant parameters in the reactor core (up to the sub-cassette), an example is the work [6-10].

2. The design scheme of the simulated power plant.

The design scheme of the plant simulation presented below can be attributed to a group of schemes for the analysis of the behavior of reactors of VVER-1000 type (model 320), which were developed for the code of ATHLET. Some results of the work of this group and comparison with the results of modeling by other programs are given in [11-17]. Below is a brief description of the main elements of the scheme used in this work.

The description of the first circuit includes the following basic elements:

- Reactor (its details will be discussed further)
- Cold (divided into 22 parts) and hot (divided into 8 parts) loops of main circulation pipelines with main circulation pumps (MCP) - his work is described by four quadrant characteristics
- Pressure compensator (divided into 6 parts) with heaters, different injection systems, safety valve system
- Hot (divided into 7 sections/parts) and cold (divided into 7 sections/parts) steam generator collector
- Steam generator tube (7 parallel channels in height, each of which is divided into 6 control volumes)
- makeup and blowdown systems of the primary circuit
- blowdown system of SG in the secondary circuit;
- systems associated with emergency boron supply from the pumps and passive systems of core emergency cooling.

Let's focus more on the reactor model, which is described as follows:

- 6 parallel-connected channels in the standpipe area and the space between the bottom of the reactor and a perforated bottom sides of the core. Of these channels, four are connected directly to the cold loops-1st, 2nd, 3rd and 4th, and the other two are located between the 1st and 2nd and 3rd and 4th plots, respectively. This splitting is caused by azimuthal nonuniform distribution of both cold and hot loop threads along the perimeter of the reactor vessel VVER-1000. The angle between the 4th and 1st and 2nd and 3rd pipes is 55 degrees, and between the 1st and 2nd, as well as 3rd and 4th pipes 125 degrees.
- 7 parallel connected channels in the subzone space between the perforated bottom of the shell and the lower support grid of cassettes-in accordance with the subsequent splitting of the core
- 7 parallel connected / disconnected channels in the reactor core (six peripheral and one central) consisting of assemblies’ groups. In the calculation with the point kinetics, you can select any group where three assemblies with different high-altitude power distribution are, in one of the “hottest” fuel rod was allocated. In case of joint calculation of hydrodynamics in the core with three-dimensional neutron kinetics it is possible to use 211 parallel hydraulic channels in the core (163 channels for assemblies and 48 channels for bypass), while one control volume for hydraulics accounts for one neutron-kinetic node in physics.
- 7 parallel connected channels at the outlet of the cassettes (unheated area to the upper support grid of the cassettes) - in accordance with the previous splitting of the core.
• 6 parallel channels simulating a bypass of the active zone/core (at the rate of change of reactor power in the point kinetics) in accordance with the peripheral channels of the active zone.
• 7 parallel connected channels between the lower and the middle grating of the protective tube block-in accordance with the peripheral channels of the core.
• 1 volume between the middle and upper grille of the protective pipe block.
• 1 volume between the upper grille of the protective tube unit and the upper cover of the reactor.
• 6 parallel connected channels in the space between the perforated shell of the protective tube block and the perforated shell of the reactor shaft – similar to the channels in the lower section of the reactor.
• 6 parallel coupled channels in the space between the perforated shell of the reactor shaft and the reactor vessel at the reactor outlet - similar to the channels in the lower section of the reactor.

The calculation scheme of the second circuit consists of the following major groups of elements:
• System of steam lines from steam generator to turbine with safety valves, BRU -A, BRU -K, BZOK, BRU-SN, check valves (about 30 control volumes relative to each steam generator).
• Piping system, from the main feed water pumps, auxiliary and emergency pumps to the steam generator, including the system of control and shut-off valves (about 20 control volumes relative to one steam generator).
• The internal volume of the steam generator is modeled by 6 interconnected elements that allow, in turn, to model the separator, internal circulation. The scope of the stills of the steam generator modeled is the 7th volume in the vertical direction. A total of 16 control volumes are used to describe SG.
• All thermophysical objects, both on the first and on the second circuit are provided with thermal structures where they take place.
• The operation control of all necessary equipment elements, which are involved in the process, is modeled.
• An additional control system is developed, which allows to set all the necessary parameters of the equipment operation (mass flow rate in loops, temperatures, pressure levels in the steam generators and pressure compensator, etc.) before the start of the transition process on the zero-transient.
• In the modeling of the process, it was assumed that the first signal for the operation of the emergency protection is skipped according to one of the operation requirements [5], as we already mentioned this above.
• The behavior of the secondary circuit equipment is not presented, although, it was taken into consideration; in this case, it plays a secondary role, especially for the initial stage of the transition process.
• Below is the scheme of splitting the lower section, the rest of the reactor, including the lower mixing chamber, the active zone (reactor core) and the Supervisory space up to the upper reactor lid in figure 1. In figure 2 shown the nodalization scheme of reactor objects in the primary circuit.
Figure 1. To the upper left side there is a cross-section along the axis connecting leg. To the Bottom – the core partitioning scheme by seven groups of parallel hydraulic channels (six peripheral ones and central one) – To the right Reactor of Unit 3, Kalinin NPP ‘cross-section of in-core area’.

Figure 2. Nodalization scheme of reactor objects in the primary circuit.
In addition, the scheme of splitting the first circuit of the first loop (figure 3) is presented, which is used as an emergency loop when considering these transients. The scheme of splitting the first loop of the first circuit from the output (V-UP4) of the mixing chamber and to the input (V-DC0) of the reactor mixing chamber. In the first group of 24 cassettes 21 with average parameters and three "hot cassettes" with different distribution of energy by height and each "hot" cassette is allocated a hot fuel rod.

Figure 3. Nodalization scheme of major internals objects.

It should be noted that different modifications of this calculation scheme were usually used earlier, as mentioned above, for the analysis of transient processes in RP models B-320 (VVER-1000) [11-17].

Figure 4 shows the model of the first circuit of the first loop, which is the same for other three circuits, except for pump. Also, the model for all components is the same in normal operation and emergency case, except for pump. The hydraulic behavior of a pump in the different states of operation is generally described by empirically developed sets of curves relating pump head and torque to the volumetric flow through the pump and the angular speed of the pump impeller. These curves are a four quadrant curves. Generally, they are supplied by the pump manufacturer for both the pump head and torque.

For normal operation, single phase homologous head curve and Single-phase homologous torque curve are used. But for simulating the pump stop, Pump Model with Speed Control and Single-phase homologous head curve are used.
3. **Description of the process (Accident Scenario).**

The considered emergency in this work is instant stop of one of the four operating MCPs, the MCP in the first loop of the primary circuit, which is assumed to start at $t=0$. Before that, a normal operation conditions were considered. Figure 5 shows a drop in the pressure difference in MCP No.1 during the whole process in 600 seconds after the emergency.

![Figure 4](image.png)

**Figure 4.** The scheme of splitting the first circuit of the first loop.

![Figure 5](image.png)

**Figure 5.** The pressure difference in MCP No.1 within 600 seconds after the emergency.
As shown in figure 6, there are fluctuations in the pressure in the reactor inlet from the emergency loop in the first period of the whole process, which shown here during 600 seconds after the emergency.

![Figure 6](image1.png)

**Figure 6.** The pressure in the reactor inlet from the emergency loop (1st) within 600 seconds after the emergency.

For this reason, considering the same previous parameters within the first 10 seconds after the emergency. Figure 7 shows the pressure difference in MCP No.1 also figure 8, the pressure in the cold leg No.1 reactor inlet, both figures illustrate in more details the oscillation in the pressure.

![Figure 7](image2.png)

**Figure 7.** The pressure difference in the MCP 1 within the first 10 seconds after the emergency.
Figure 8. The pressure in the cold leg No.1 reactor inlet within the first 10 seconds after the emergency.

From the previous dissection, it can be generally indicated that a strongest amplitude and frequency of pressure fluctuations on NPP main component are happened during the initial period of the accident. Thereby, to see certainly more details about the consequences of this emergency, tracking the pressure in the emergency loop No.1 in first second is considered.

The few following figures show the change in one of the most important parameters, as a consequence of an emergency, which can affect the performance of the main equipment elements. Figure 9 shows the pressure before the MCP.1 (after the steam generator exit collector side), and after the MCP.1 (before the reactor inlet pipe side).

Figure 9. Pressure changes before and after MCP .1 within the first second after the emergency.
The pressure drops in the core, reactor, pump and steam generator during the emergency loop, is shown in figure 10.

**Figure 10.** Pressure difference in core, reactor, MCP.1 and Steam Generator within the first second after the emergency.

As the first second after the pump stops shows the highest fluctuations in pressure difference in the main components, also the strongest amplitude and frequency of pressure fluctuations in inlet to cold leg, outlet of hot leg, inlet and exit of the steam generator shown in figure 11 in the emergency loop.

**Figure 11.** Pressure in cold leg, hot leg, steam generator inlet and exit through the 1st loop within the first second after the emergency.
In order to know how this drop in the pressure reflects on the other important parameters, the mass flow rate of the coolant from the four loops is shown in figure 12. It is observed a rapid drop in the emergency circuit just after the pump stop.

![Figure 12](image12.png)

**Figure 12.** The mass flow rate in the cold leg for the 4 loops in the primary circuit within 600 seconds after the emergency.

For more details, the behavior in the first 10 seconds after the emergency is considered. The mass flow rate from the 4 cold legs, figure 13, shows that the flow from the emergency loop (1st) reversed its direction back within only 1 second after emergency.

![Figure 13](image13.png)

**Figure 13.** The mass flow rate in the cold leg for the 4 loops in the primary circuit within 10 seconds after the emergency.

As mentioned before, the first second in the considered emergency shows many important details about consequences happened in the main components, thus the mass flow rate from the four loops is
show in figure 14 during this first second. The reversed flow is formed by the coolant from the other intact loops, mainly the closest loops No. 4 then No. 2.

Figure 14. Mass flow rates into the core from the four coolant loops within the first second after the emergency.

This change in the mass flow from the emergency loop effect, as expected, by decreasing on the total mass flow into the core, figure 15.

Figure 15 Mass flow rate into the core within the first 10 seconds after the emergency.
Taking into consideration the important function of the coolant, to remove the heat from the reactor core, figure 16, shows the change in coolant temperatures at inlets and outlets of the reactor. The reversed flow -as shown before- is passing through the steam generator, it is getting cold even more, which effect on the output temperature in the hot leg No.1.

![Figure 16. The coolant temperature changes in inlet and outlet of the reactor within the first 10 seconds after the emergency.](image1)

The inevitable result of all this change of the coolant quantity, direction and temperature within the pipelines of the cooling circuits, the change in the average coolant temperature in the core is observed as shown in figure 17. Additionally, a change in average temperature of the fuel, figure 18, is noticed too.

![Figure 17. Average coolant Temperature in the core within the first 10 seconds after the emergency.](image2)
Figure 18. Average fuel Temperature in the core within the first 10 seconds after the emergency.

Based on the referred changes on the temperatures of the coolant and fuel, the reactor reactivity effected by it. As shown in figure 19 its behavior in the whole process during 600 seconds after the emergency.

Figure 19. The Reactor total reactivity within 600 seconds after the emergency.

As expected consequence, also the reactor power follows exactly this behavior as shown in figure 20.
Figure 20. The reactor power within 600 seconds after the emergency.

It has also been adopted the pressure and flow rate in the first 10 seconds for more details, figure 21 shows the reactor total reactivity, and figure 22 shows that the reactor power also has the same behavior in changing with time. It can be notice that the reactor power drop to 80% from the normal operation within 1~2 sec.

Figure 21. The reactor total reactivity within 10 seconds after the emergency.
Figure 22. The reactor power within 10 seconds after the emergency.

Hence, the change in total reactivity is due to the change in its components, figure 23 shows the change in the components of reactor reactivity. Doppler effect has a positive reactivity coefficient, the temperature and density have a negative reactivity coefficient. This proves that the temperature reactivity is the predominant for that affect, which leads to a decrease in the reactor power.

Figure 23. The reactor reactivity in details within the first 10 seconds after the emergency.
4. Conclusion

The details of the initial period of the MCP instantaneous stops accident were analyzed, in which the strongest amplitude and frequency of pressure fluctuations on NPP elements, especially the MCP and SG, is observed. Strongest amplitude and frequency of pressure can lead to significant dynamic loads on the structural elements of these objects. This can be estimated either by: Joint strength and hydrodynamic calculations, or it is possible to use the results obtained in this work as boundary conditions for the calculation of dynamic loads.

It is necessary to consider a similar process for the other MCPs specially the pump in 3rd circuit in which the pressurizer connected. In the safety case, emergency situations are analyzed, which can also lead to significant pressure fluctuations in the first stage, in particular, these are accidents with instantaneous rupture of the main circulation pipeline.

It should be noted that the emergency protection in this case did not work (see power change), because it was assumed that the first signal for it is skipped (in this case, the first signal was the pressure drop on the emergency pump).

So, the next step in the direction of analyzing the occurrence of pressure waves will be just the study of such accidents at breaks in different parts of pipelines.

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