AtomFuture-2017
XIII International Youth Scientific and Practical Conference
"FUTURE OF ATOMIC ENERGY - AtomFuture 2017"
Volume 2017

Knowledge E
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Conference Paper

Computational Analysis of the Shutdown of One MCP of VVER-1000 During Operation of the Reactor Facility at Nominal Power

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Abstract

The article considers results of the computational analysis of process with a disturbance in the operation of VVER-1000 reactor facility, caused by the de-energizing of a single Main Circulation Pump (MCP-195).

Calculations of de-energizing of the MCP were made with the CORSAR/GP software package developed by Federal State Unitary Enterprise "Alexandrov RESE". This software package is a contour code which allows making calculations of emergency situations taking into account the operation of various systems, including safety systems. CORSAR/GP is certified and verified for facilities with water-cooled reactors including VVER-1000.

Developed nodalization scheme (computational scheme) of the first circuit of VVER-1000 allows adding or excluding the operation of protective safety systems and can also be used for a computational analysis of other disturbances of normal operation.

To perform the calculations parameters of the core were chosen to ensure conservative results for the parameters determining the current state of the reactor facility (fuel temperature, boiling point, etc.).

The computational analysis showed that in case of de-energizing of a single MCP when the reactor is operated at nominal power criteria describing the safety of the facility are fulfilled, protective automatic actions of safety systems transfer the reactor facility to a controlled safe state.

Keywords: shutdown of the MCP, safety, VVER-1000, contour code, off-nominal situation.

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Received: 23 December 2017
Accepted: 15 January 2018
Published: 21 February 2018

Publishing services provided by Knowledge E

Selection and Peer-review under the responsibility of the AtomFuture Conference Committee.

In a very short period of time since 1954, when the world's first nuclear power plant (NPP) was commissioned in Obninsk, nuclear power plants have become quite competitive in compare with thermal power plants (TPPs) using organic fuels. Compared with thermal power plants, the cost of electricity generated at nuclear power plants is much lower and environmental friendliness of nuclear power plants is higher. At the same time the development of nuclear power put forward a serious problem - the prevention of accidents at nuclear power plants. Technical systems of great complexity and high power, which include nuclear power facilities, create a certain degree of risk of accidents dangerous to humans and the environment [1].

During operation, extremely high reliability and safety of nuclear power plants must be ensured. Reliability is the property of a nuclear power plant (NPP) to retain the ability to generate electrical and (or) thermal energy of prescribed parameters by the required regime of energy production in radiation conditions permissible for normal operation with a given system of maintenance and repair of equipment during the time. Safety of nuclear power plants is not a component of reliability, it is an independent property provided by special means. Of course it depends on the reliability of the main equipment at some way, but not completely determined by it. Safety of nuclear power plants should be ensured not only and not so much in normal operation as in possible off-nominal situations caused by failures of the main equipment (related to its reliability) and caused by other reasons (personnel errors, natural disasters: earthquakes, floods, aircraft at nuclear power plants, etc.) [1].

The emergency operation modes of the nuclear power plant are reduced to two characteristic situations: a sudden increase in energy release with a constant heat removal and sudden deterioration of the heat removal at a constant power. The increase in energy release above the permissible level is a nuclear accident, and the deterioration of the heat removal is an accident related to equipment failure and loss of coolant. The first situation arises as a result of an uncontrolled increase of reactivity, for example when the control rods of the control system are jammed or they can't to enter into the core. Also this situation arises in case of sudden changes in the temperature and composition of the coolant, etc. The main reasons for the sudden deterioration of the heat removal are the switching off of the MCP, the depressurization of the main circulation circuit with loss of coolant, the decrease in the cross-section for the coolant in the parallel channels of the core due to the destruction of any nodes of the devices inside the vessel, what will lead to complete or partial "blockage" of single channels [2].

In the case of deterministic, computational safety justification of nuclear power plants emergency modes are divided into groups of characteristic effects on the change in parameters [3]:

1. modes with disruption of the systems that affect on reactivity;

2. disturbance of the coolant flow rate;

3. disturbance of cooling conditions from the side of the second circuit;

4. modes with depressurization of the second circuit;

5. modes with decompaction of the first circuit.

One of the serious design accidents is a disturbance of the coolant flow rate. The change in the coolant flow through the reactor core can be caused by disturbances in the operation of the main circulation pumps. If there are deviations in the power supply parameters from the nominal value, accidents in the power supply chain or mechanical damage to the MCP, then the coolant flow will reduce, what will lead to an increase of the coolant temperature [4]. All this can lead to insufficient cooling of the core and then to boiling of the coolant and to the beginning of the heat exchange crisis on the surface of the most heat-stressed fuel element. The most pessimistic final state of such emergency sequence with boiling of the coolant is the core meltdown. Decrease or complete lack of the circulation of the coolant through the core can be caused by the deenergization of a different number of MCPs when the reactor is operated at nominal power.

The de-energizing of one MCP during the operation of NPP at nominal power is an off-nominal situation. However, this event is usually not considered as an emergency. That's why the question of the consequences of such a situation is extremely important in the safety meaning and in the definition of the state of the NPP.

The deenergization of the MCP, as described in [5], can occur due to a malfunction of the power supply chain in the MCP itself and in the electrical chain of the auxiliary needs, and also it can occur due to wrong actions of the personnel. After de-energizing of one of the four operating MCPs, the coolant flow through the core decreases and also reactor power decreases due to the action of the automation and feedbacks of the reactor [6].

In this work it was taken into account that the primary coolant flow rate will decrease gradually because the MCP at rundown will rotate due to the inertia of the flywheels. Analysis of the cooling conditions of the core with the deenergization of one of the four operating MCPs was carried out on the basis of a change in the reactor thermal power, the coolant flow through the core, the coolant pressure in the upper and lower chambers of the reactor, and also in change of the temperature at the inlet to the core.

The initial data for the analysis of the investigated mode were taken as the design parameters of the VVER-1000 reactor: the nominal thermal power is 3000 MW, the primary coolant temperature at the reactor inlet is 290°C, the heating in the reactor is 30°C, the pressure in the reactor is 15.9 MPa, in the initial state of work there are all four MCPs [7].

At the reactor with VVER-1000, the MCP-195 is used as a pump. On the drive motor of this pump there is a massive flywheel. It provides a rundown of the MCP when the pump is deenergized, so in this situation the flow rate change gradually, not sharply (as, for example, in the case of a jamming of the MCP shaft). The flywheel is a steel circle with a diameter of 2.15 m, thickness of 0.3 meters and mass of 5.1 tons [8].

As a program complex for analysis of consequence of de-energizing of one MCP the contour code RK KORSAR/GP was chosen. KORSAR/GP is developed by Federal State Unitary Enterprise "Alexandrov RESE". Some part of modules for this code is developed in OKB "Gidropress". At present time this code is used in the OKB "Gidropress" for calculations in support of the safety of the VVER. RK CORSAR/GP is designed for numerical simulation of stationary states, transient and emergency modes of reactor facilities with light water moderator and coolant. RK CORSAR/GP provides coupled numerical simulation of non-stationary neutron-physical and thermal-hydraulic processes in the VVER reactor in operational and emergency modes [9].

RK CORSAR/GP may be used for a computational safety justification at all stages of the life cycle of NPPs with VVER. This contour code is used for: calculations of the dynamic of the reactor with VVER at the design and operation stages, for deterministic calculations of transient and emergency modes of a reactor with a VVER related to probabilistic safety analysis, for simulating thermal hydraulic processes in experimental facilities and assemblies with a water coolant [10].

Modeling of thermohydraulic processes in the RK CORSAR is carried out on the basis of a completely nonequilibrium two-fluid model (there are three conservation equations: of energy, of mass and of momentum for the water phase and also the same equations for vapor phase) in the one-dimensional approximation. Neutron-physical calculation is performed in a three-dimensional two-group diffusion approximation with 6 groups of delayed neutrons. The initial data is set by using the DLC language (Data Language for Codes) specially developed for the RK KORSAR [11].

The input file is a text file written in the form of a program in the DLC language in accordance with the user developed so called nodalization [12] (or computational) scheme. The file consists of a set of procedures that allow describing the relationships between elements and the single-valued conditions for each element of the nodalization scheme.

Model of contour thermohydraulic in RK KORSAR represents the coolant circuit as a set of thermohydraulic cells (control volumes) connected with each other. The relation of the two thermohydraulic cells with each other determines the connection. The scalar characteristics of the coolant flow, such as pressure, enthalpy of phases, gas content, etc., are tied to cell centers; vector characteristics, the main of which are phase velocities - to connections.

The elements of the thermohydraulic system are combined into a nodalization scheme. The developed nodalization scheme of the first circuit of the VVER-1000 is shown in Fig. 1. This scheme includes such built into the contour code elements as:

- ACCUM – hydraulic accumulator,
- CH – channel,
- COL – collector,
- SLVES – steam-water vessel under pressure,
- BVOL_T1 – predetermined boundary cell,
- HSC – heat conducting structure,
- CPUMP – Main Circulation Pump,
- VAL – valve,
- KINET – three-dimensional neutron kinetics,
- LREF – side reflector,
- FROD – fuel element.

Steam-water vessel under pressure element in combination with the specified boundary cell simulate the work of pressure compensator. The steam generator is modeled by setting the boundary condition for the heat exchange on the "HCS3" element. Newmann's boundary condition is set on every of four heat conducting structures, which simulate steam generators. Heat flow from surface of every simulator of steam generator equals one fourth part of thermal power of core.

In the core there are 163 channels divided into 20 cells in height each. Due to connection of every channel with its own fuel element "FROD" and its own heat conducting structure "HCS" water in channels heat up. The array from 163 elements is called "Nchaz1".

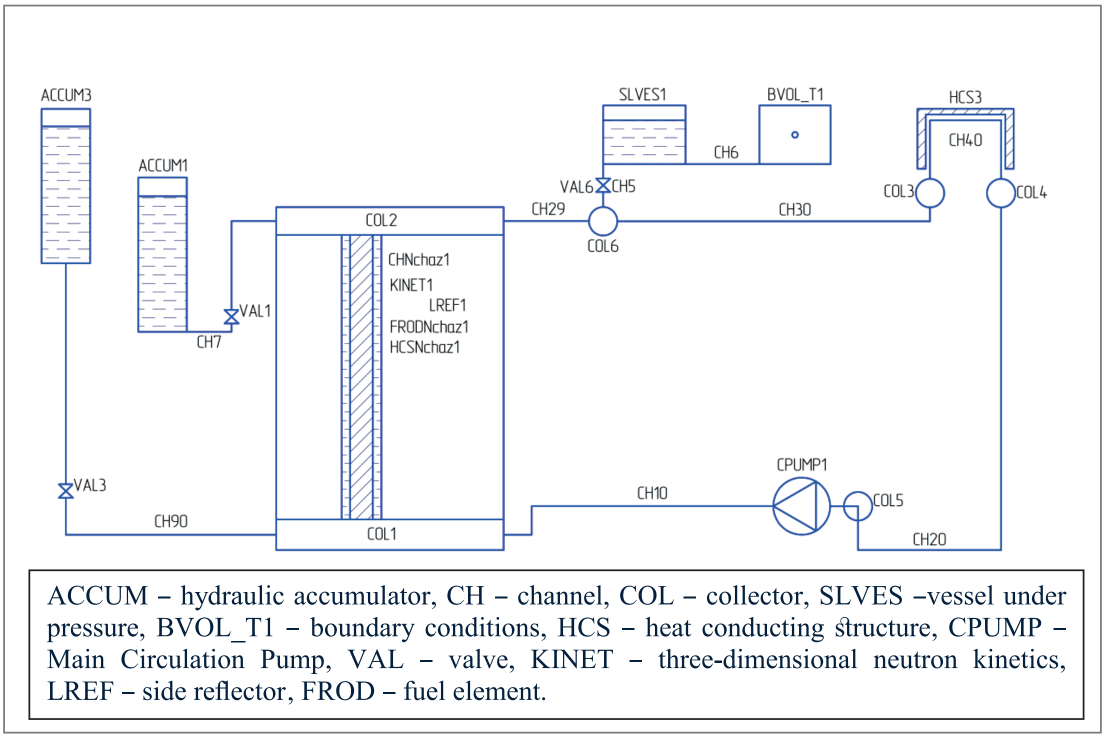

Figure 1: Nodalization scheme of the first circuit of VVER-1000 reactor.

1. Results of computational modeling

Figures 2 and 3 show the change in the parameters of the reactor facility operating at rated power in case of the de-energizing of one of the four operating MCPs.

Figure 2 shows the change of the flow rate of the coolant through the core and of the reactor thermal power in percentage of the nominal values. The abscissa is the time in seconds. The time *tau* = 0 is the time of beginning of the MCP stop. Line "*Ncore*" shows the change in thermal power, line "*Gcore*" – change in flow rate. De-energizing of one of the 4 working MCPs is cause of the rundown of the MCP, and then according to calculations flow rate will gradually decrease to a level of about 75% of the initial value. The decrease of the reactor power is performed predominantly by the operation of the automatic power limiting regulator (PLR), which reduces the power to a level corresponding to the number of MCPs remained in the work. As a result of the PLR action the power value and also the flow rate value decrease to about 75% of the nominal value.

Figure 3 shows the change in the coolant temperature at the outlet from the core (line "*T.out*"), the maximum temperature of the outside of the fuel cladding (line "*T.clad*"), and the maximum fuel temperature (line "*T.fuel*").

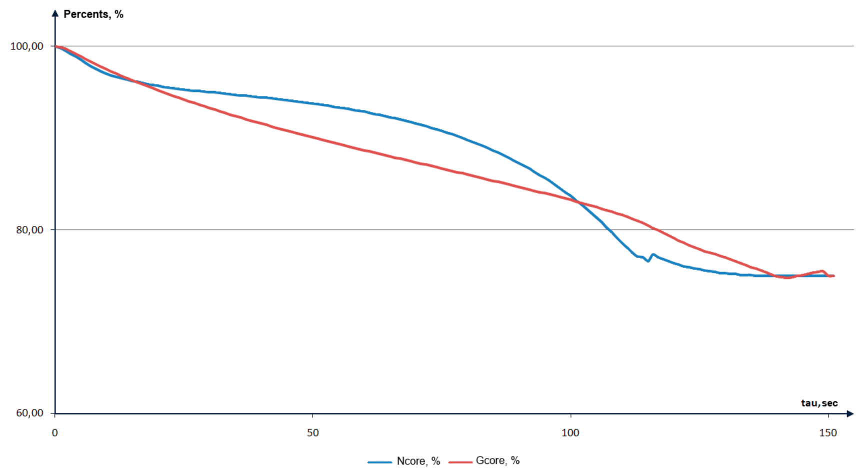

Figure 2: Change of flow rate and of reactor thermal power.

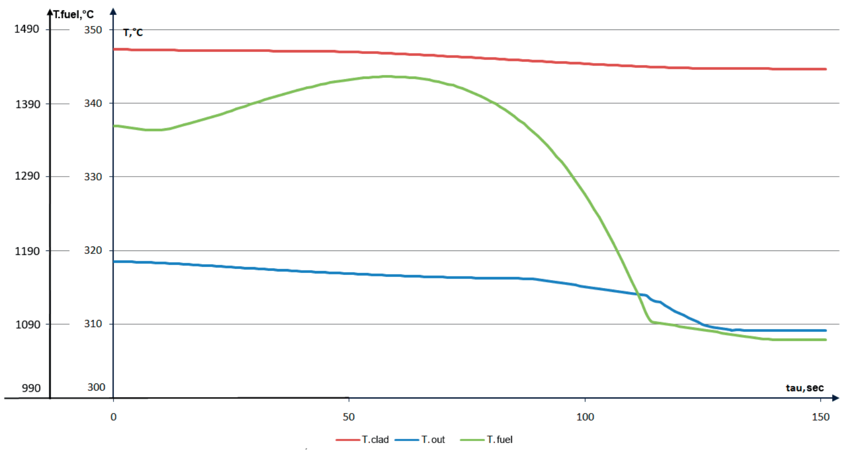

Figure 3: Change of main temperatures.

In the figure above (fig. 3) temperatures of water and cladding are shown on the inner vertical axis (range from 300°C to 350°C), fuel temperatures is shown on the outer axis (with a range from 990°C to 1490°C). To determine the maximum of fuel and cladding temperatures a special procedure was written in the input file. This procedure determines the maximum values from all 20 computational cells of each of the 163 channels. As a result of calculations it was found that the temperature of the coolant at the outlet from the core decreased from 320°C to about 309°C, the maximum temperature of the fuel decreased from 1380°C to 1080°C and the temperature of the cladding changed only for a couple degrees.

2. Safety justification

Safety of NPP operation in case of disturbance of normal operating conditions is assessed by the so called safe operation limits.

The safety justification of the reactor facility in the mode of de-energizing of one MCP is making on the basis of the analysis of the performance of the limits of safe operation of the VVER-1000 reactor on the technological parameters of the primary circuit in conditions of disturbance of normal operating conditions [4]. The list of parameters, their maximum values and obtained calculated values are given in Table 1. T_{nom} is a value of 320 degrees. Given values describe NPP integral parameters, which determine the limits of safe operation. These limits are involved in the operational management of the nuclear power plant. The dominant part of the limit values of the controlled parameters specified in this table determines the safety level of the main physical barriers of protection in the reactor: the fuel cladding and the fuel itself [4].

Table 1: Safety limits for the first circuit and calculated values.

Parameter name	Safe operation limits	Obtained calculated values
Maximum power value, not more than, % of Nom	<107	100
Maximum pressure in 1 circuit, not more than, MPa	<17,64	15,95
Maximum temperature in "hot" thread of circulation loop, not more than, °C	<(T_{nom}+8)	T_{nom}
Maximum pressure drop on the reactor core ΔP_{core} with temperature of 1 circuit is not less than 260°C, no more than, MPa	<0,46	0,2

The maximum value of the reactor power in the calculation was at the beginning of the process and was 100% of the nominal value. The calculation shows that later the power value only decreased. Comparison with the limits of safe operation shows that the power value does not go beyond the limit of safe operation in 107% of the nominal value.

The maximum pressure received by the calculation was 15.95 MPa, and did not go beyond the safe operation limit for the primary circuit pressure (17.64 MPa).

The maximum temperature in the "hot" thread of the circulation loop is obtained at the beginning of the process and was 320°C. Later this value decreases and does not overcome the initial value of 320°C.

The maximum calculated pressure drop across the core of the reactor is 0.2 MPa and it does not overcome the safety limits for pressure drop in the main circulation circuit (0, 46 MPa).

Calculation shows a decrease: of coolant temperature at the outlet from the core - 11°C; of maximum temperature of fuel cladding - 4°C; of maximum fuel temperature is 300°C.

In this way, the mode with de-energizing of one MCP during the operation of VVER-1000 at nominal power does not go beyond the limits of safe operation.

In conclusion it's must be told that made computational analysis of the situation with de-energizing of one MCP shows that criteria describing the safety of the reactor facility are fulfilled. In total work of systems, including safety systems, the reactor facility transfers into a controlled safe state.

It should also be pointed that developed nodalization scheme of the first circuit of VVER-1000 allows adding or excluding the operation of protective safety systems and can also be used for a computational analysis of other disturbances of normal operation.

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