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Research Article

A study into the modes of the VVER-1000 RCP starting in an earlier inoperative loop*

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Abstract

To simulate the mode of the RCP starting in an earlier inoperative loop, KORSAR/GP, a code supporting coupled numerical modeling of neutronic and thermal-hydraulic transients in a VVER reactor plant in operating and emergency conditions, was chosen as the computational tool.

Studying these modes using thermal-hydraulic codes makes it possible to analyze the course of transients and certain emergency processes without using commercial test procedures, which contributes to laying the groundwork for addressing the issues involved in ensuring the reliability, operating safety and efficiency of nuclear power plants.

Increased requirements to the safety of NPPs identify the need for avoiding excessive conservatism in the analysis based on which requirements to safety systems are formulated, as well as for enhancing the knowledge of the regularities of thermal-hydraulic transients based on advanced computer programs (or codes) designed for improved computational analysis of non-stationary thermal hydraulics in the water-cooled reactor circulation circuits in emergency and transient modes relying on inhomogeneous non-equilibrium mathematical models of two-phase flows and on a detailed description of the physical transient regularities.

The purpose of the study is to analyze computationally the starting of a VVER-1000 RCP in an earlier inoperative loop at different reactor plant power values. To do this, one requires to develop the VVER-1000 reactor primary circuit computational pattern to model the transient taking place as one RCP is started, to conduct a further analysis, and to compare the key monitored reactor coolant and core parameters (power, temperature, flow rate, etc.).

Keywords

Mode, reactor coolant pump (RCP), circulation loop, reactor plant, scram, reactivity coefficient, safety margin, reactor plant power

1. Introduction

The prime objective in ensuring safety at all NPP lifecycle stages is to take effective measures aimed at preventing severe accidents, and protecting personnel and the population by avoiding or limiting the release of radioactive products into the environment at any circumstances. Absolute reliability and safety of nuclear power plants must be ensured during operation. Reliability is a property of a nuclear power plant to maintain, over time, the capacity to produce

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electrical and/or thermal energy of the prescribed parameters in accordance with the desired generation schedule in such radiological conditions as permitted for normal operation with the given equipment maintenance and repair system. The NPP safety is not a constituent property of reliability but an independent property ensured by special means. It certainly depends, to a certain extent, on reliability of key components but is not defined by it in full. The NPP safety shall be ensured not only and not so in normal operation as in emergencies associated with failures of key components (i.e., its reliability) or caused otherwise (personnel errors, natural disasters including floods, aircraft impacts, etc.) (Ivanov 1994; Afrov et al. 2006; Baklushin 2011).

Situations involving failures to observe the NPP normal operation limits are reduced to two typical cases: a sudden increase in the power density with an invariable heat release and a sudden loss of normal heat release with a constant power. A power density growth to above the permissible level is a nuclear accident, and the worsening of heat release is an accident associated with equipment failure and loss of coolant. The former results from an uncontrolled increase in reactivity, e.g., when the CPS control rods become seized or cannot be inserted into the core, the coolant temperature and composition changes abruptly, etc. The major reasons for a sudden worsening of heat release include the RCP trip, loss of the main circulation circuit pressure with a coolant leakage, or a reduction of the coolant flow area in the core parallel channels due to some parts of the core internals having failed, which may result in some of the channels to be fully or partially clogged (Bukrinsky 1982; Nigmatulin and Nigmatulin 1986; GOST R 50088-92 1994;Margulova 1994; Vygovsky et al. 2011; Baklushin 2012; Belozerov and Zhuk 2012).

For the deterministic computational justification of the NPP safety, emergency modes are divided by groups of the typical impacts they have on parameter variations (Vygovsky et al. 2011):

- malfunction of reactivity affecting systems;
- coolant flow rate disturbance;
- loss of normal secondary circuit cooling;
- loss of the secondary circuit pressure;
- loss of the primary circuit integrity.

2. Connection of an RCP in an earlier inoperative loop

One of the major anticipated operational occurrences is modes with malfunction of the systems that affect the in-core reactivity change. One of such potential modes with malfunction of the reactivity change affecting systems is the starting of an RCP in an earlier inoperative loop. The initial event for this malfunction is the NPP operation at a level of 70% (+ 7% for the calculation of the mode involving the starting of this RCP) of the rated power after one of the RCPs trips. The starting of an RCP in an earlier inoperative loop is possible in response to an erroneous signal to connect the idle circulation circuit, which leads to a violation of standard operating procedures. As the flow direction in the started loop changes and the coolant flow rate through the reactor core becomes greater, the core coolant temperature lowers. With the neutronic performance of the core corresponding to the negative temperature reactivity coefficients of the coolant, the reactor power will increase (Vygovsky et al. 2011).

The amount and the rate of the reactor power increase and the change in the primary and secondary circuit parameters (Bukrinsky 1982) depend on the coolant temperature and fuel temperature reactivity coefficients that may have different values throughout the lifetime. The reactor neutron power increases to the scram setting values. The scram is followed by the secondary circuit pressure decreasing down to the level of the TG stop and control valve closure, and, further on, the primary and secondary circuit parameters are maintained by operation of the secondary circuit steam relief valves with discharge to the atmosphere (BRU-A) and to the turbine condenser (BRU-K).

The most dangerous mode is that in which the scram setting is not achieved, that is, when the operator has not brought the setting to a new (lower) power value after the RCP trip. In this case, the power surge in the cold sector is the maximum one but there is no departure from nucleate boiling. However, this may lead to the fuel rod integrity to be lost due to an increased gas permeability of the cladding in some of the rods. To avoid this fuel damage, the connection of one RCP to the three others in operation requires the reactor power to be cut back in advance to 20 and 30% of the rated power value respectively (Vygovsky et al. 2011).

The mode that influences the reactivity change, namely, the connection of an RCP in an earlier inoperative loop is an abnormal situation. However, this event is not normally regarded an emergency. It is exactly for this reason that the consequences of such situation are vitally important in terms of influence on safety and the reactor plant state identification.

For the purpose of the safety analysis calculations, calculation data for the connection of an RCP in an earlier inoperative loop is included in the technical safety analysis report (TSAR). Technical safety analysis report is the document that includes the bench and field test data for different components (for instance, fuel rod, RCP, valves, etc.), but, to a greater extent, a technical safety analysis report is based on the computational analysis of operating and emergency modes with various equipment malfunctions and erroneous actions of operating personnel (Vygovsky et al. 2011).

Computational analyses for the connection of an RCP in an earlier inoperative loop were performed as part of various studies. Some of these involved fairly conservative assumptions.

3. Presentation of the VVER-1000 primary circuit in the KORSAR code

A thermal-hydraulic system to be calculated using the KORSAR code is broken down into a number of components.

The KORSAR code's circuit thermal-hydraulics model represents the coolant circuit as a set of interconnected thermal-hydraulic cells (test volumes). The interconnection of two thermal-hydraulic cells is defined by the respective junction. The *scalar* characteristics of the coolant flow (pressure, phase enthalpy, gas content, etc.) are tied-in to the cell centers; the *vector* ones, the key of which are phase velocities, are tied-in to junctions.

Thermal-hydraulic cells are divided into *boundary* cells, the parameters of which are the input ones for the circuit thermal-hydraulics model, and *computational* cells, the parameters in which are determined in the model operation process.

The components of a thermal-hydraulic system are united into a nodalization scheme. The connections among the nodalization scheme components are coded in the input data file as part of the LAYOUT procedure using the permitted presentation forms of components based on program names of component types, component numbers and component constituents, and based on particular rules (Baklushin 2011).

A large number of component types are used in the KORSAR, including channels, gate valves, tanks, local resistances, heat conducting structures, given mass sources, etc. These are what a single nodalization scheme is composed of, based on which thermal-hydraulic parameters are calculated.

The developed nodalization scheme of a VVER-1000 reactor plant primary circuit is shown in Fig. 1.

We shall describe in brief the nodalization scheme components.

SMASS_T2 is designed to specify the flow rate and phase enthalpies of the fluid fed into the channel's computational cell (CC) and is used to calculate the coolant flow rate and temperature at the RCP outlet. The component (item 1) describes the operation of the three RCPs, and the component (item 2) describes the fourth RCP as the boundary conditions change over time.

BLJUN2 defines the direction of the phase introduction into the channel cell, which are taken into account when the cell parameters are calculated.

CH calculates the parameters of the coolant flow portion between branching points and other components of the nodalization scheme in a two-phase one-dimensional approximation. It is possible to describe by a single element the flow in a pipeline or in a fuel channel of the reactor core FA, the reactor inlet annular space, a vessel (distributed, in a one-dimensional approximation), etc. A system of channels can be used to describe the reactor upper and lower mixing chambers, a primary or secondary circuit steam generator, circulation pipelines, etc. This element (CH1) also describes the axially distributed core flow area.

The header (COL) is designed to calculate the coolant parameters in a volume having more than two connections to the circuit. The header space is connected to the circuit via channels and/or degenerated channels. The header parameters are calculated in a point approximation practically in the same assumptions as the channel's computational cell parameters.

The boundary cell (BVOL_T1) is designed for changing, by the well-known law, the fluid scalar characteristics (pressure, enthalpy, and fraction volume ratios) that are used as boundary conditions for the circuit thermal hydraulics.

The heat conducting structure (HSC1) is designed to calculate the temperature field distribution in a solid body with the preset boundary conditions for the heat exchange on the body surfaces and the capability to preset the power density within the body assuming that the geometrical characteristics and the relative power density distribution in the fuel channel are constant. As part of this scheme, it serves a heat structure in the form of a fuel rod (axially distributed fuel and cladding).

The boundary condition for the fuel rods (BHEAT) is designed to preset the conditions of heat exchange of the first, the second, and the third kind on the surface of the heat conducting structure. In this scheme, it serves a boundary condition on the rod's left surface (we preset the zero flow condition at the center).

The moderator (MOD1) is designed to process the coolant parameters (temperature, steam quality, etc.), to calculate the feedback reactivity from the coolant parameters, and to transmit data into the reactor kinetics calculation unit.

FUEL1 serves to calculate the temperature reactivity effect from the average-weighted temperature of the fuel rods and the power density in these. It is required to process the fuel temperature, to calculate the feedback reactivity from the fuel temperature, and to transmit data into the reactor kinetics calculation unit.

CORE1, the point neutron kinetics unit, is designed to calculate the reactor core neutron and thermal power. Neutron power is calculated in a point approximation with regard for delayed neutrons (Weinberg and Wigner 1961; Safety of Nuclear Power 1994; RELAP5/MOD3 Code Manual 1995; Belozerov et al. 2008; Kazantsev et al. 2009; KORSAR/GP Code 2009; Dragunov et al. 2019). The reactivity margin, coolant temperature and fuel temperature feedbacks, and control rod reactivities are taken into account in reactivity calculations. No reactor xenon and samarium poisoning is taken into account in this paper. After the reactor neutron kinetics is calculated, the obtained thermal power data is transmitted into the fuel rod heat structure (HSC1).

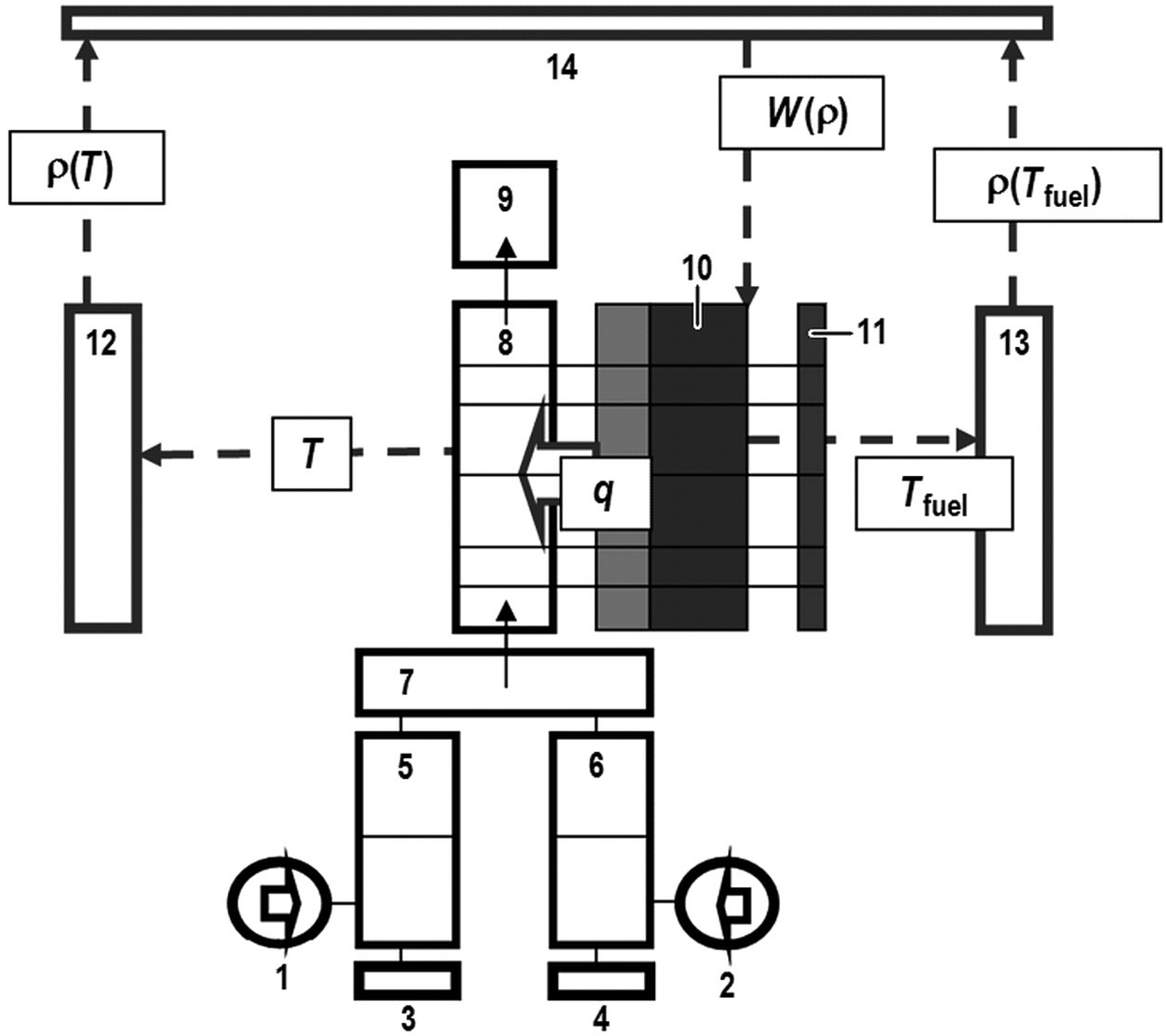

Figure 1. Nodalization scheme of the VVER-1000 primary circuit: SMASS_T2 – boundary conditions for RCP (1, 2); BLJUN2 – boundary conditions for supplying channels (3, 4); CH2 – channel (5, 6); COL1 – header (7); CH1 – core flow area (8); BVOL_T1 – given boundary cell (9); HSC1 – heat conducting structure (10); BHEAT – boundary condition for fuel rods (11); MOD1 – moderator (12); FUEL1 – fuel (13); CORE1 – pointwise neutron kinetics (14).

4. Computational simulation results

The computational simulation involved studies on the RCP starting in an earlier inoperative loop at different reactor power levels, as well as simulation with the coolant temperature change in the cold leg of the formerly idle loop (Petrosyants 1981; Belozerov et al. 2008; Gordon 2009; Asmolov et al. 2010).

Further on, the paper presents the change in the reactor plant key parameters in response to the RCP starting in an earlier inoperative loop at the VVER-1000 reactor thermal power of 75, 53, 38, 34, and 30% of the rated value (3000 MW). The simulation was also performed at 23 and 75% of the rated power (3000 MW).

The core coolant flow rate is the same for each mode when the RCP is started.

Fig. 2 presents the core coolant flow rate change in response to the RCP starting in an earlier inoperative loop. The time (in sec) is plotted along the X axis. Time = 0 corresponds to the RCP starting onset. As shown by the calculation results, the RCP starting in an earlier inope-

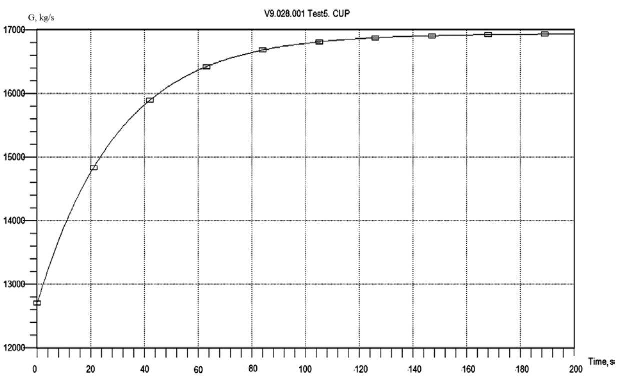

Figure 2. Core coolant flow rate change (a monitor screen portion during the KORSAR operation).

rative loop will lead to a smooth increase in the coolant flow rate through the reactor core. At the end of the mode, the flow rate becomes equal to 16900 kg/s (80400 m^3/h) which corresponds to the rated operation of all four RCPs.

Figs 3, 4 present the change in the reactor plant key parameters in response to the RCP starting in an earlier inoperative loop by 30% (900 MW) of the reactor rated power (3000 MW).

Fig. 3 shows the change in the core outlet coolant temperature. As the RCP is started, the core coolant flow rate increases due to which the coolant temperature is decreasing for the initial 30 s and subsequently equalizes. The value $\Delta T_{t.out}$ in this mode is equal to 1 °C.

Fig. 4 shows the reactivity inserted into the reactor thanks to the coolant temperature feedback. Because of a negative coolant temperature reactivity effect, we get that a positive reactivity of 0.00021 *dk*/*k* is inserted into the reactor as the temperature decreases.

To compare the obtained data when considering the mode of the RCP connection in an earlier inoperative loop at different power levels, the values of the reactor key parameters during and after this mode have been tabulated (Tables 1 through 3).

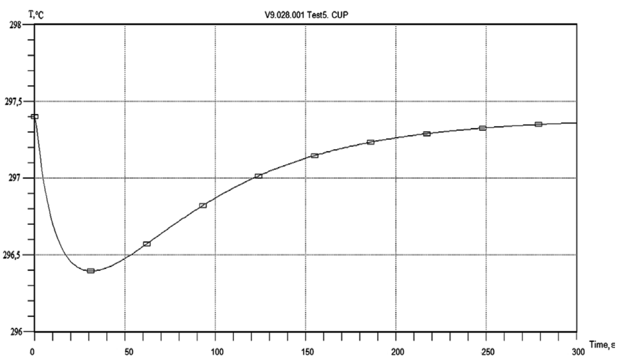

Figure 3. Core outlet coolant temperature change (a monitor screen portion during the KORSAR operation).

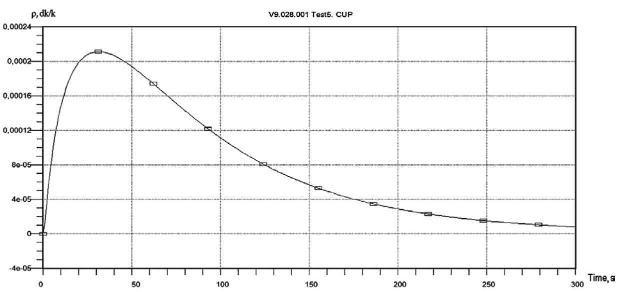

Figure 4. Reactivity inserted into the reactor (a monitor screen portion during the KORSAR operation).

5. Checking the safety criteria fulfillment

The obtained calculated values of the NPP key parameters require to be compared with their limiting values that ensure safe operation of the primary circuit. This will make it possible to evaluate the safety criteria fulfillment.

Table 1. Values of the reactor key parameters after the RCP connection in an earlier inoperative loop.

Initial reactor plant power,% N_{nom}	30	34	37.5	53	75
Initial core coolant flow rate, m^3/h	60300	60300	60300	60300	60300
Final core coolant flow rate,m^3/h	80400	80400	80400	80400	80400
Core outlet coolant T change at the RCP starting time, °C	1	1.1	1.1	1.2	1.3
Core outlet coolant T, °C	297.4	299.1	300.7	306.9	316.1
Inserted reactivity, dk/k	0.00021	0.00022	0.00023	0.00024	0.00026
Final reactor plant power,% N_{nom}	39.9	44.1	50	70	100
Final fuel cladding temperature, °C	313.6	317.4	321	340	364
Change of the minimum critical power ratio, relative units	2.6	2.3	2	1.4	1
Final minimum critical power ratio, relative units	8	7.1	6.4	4.5	3

Table 2. Simulation results for the RCP starting in an earlier inoperative loop at 23% of the reactor plant power.

Loop 4 cold leg coolant T, °C	100	200	250	290
Final reactor plant power, MW	7800 (260%)	4300 (140%)	2450 (82%)	900 (30%)
Inserted reactivity, dk/k	0.0031	0.00195	0.0011	0.00019
Core inlet coolant T, °C	250	272	281	290
Core outlet coolant T, °C	331	317	308	300
Core outlet saturation T, °C	347	347	347	347

Table 3. Simulation results for the RCP starting in an earlier inoperative loop at 75% of the reactor plant power.

Loop 4 cold leg coolant T, °C	100	200	250	280
Final reactor plant power, MW	19000 (630%)	6500 (217%)	4215 (140%)	3414 (114%)
Inserted reactivity, dk/k	0.0018	0.0012	0.0006	0.00037
Core inlet coolant T °C	257	272	284	289
Core outlet coolant T, °C	347	335	325	322
Core outlet saturation T, °C	347	347	347	347

Prior to the RCP starting in an earlier inoperative loop, it is important to bring the scram setting to a new level more reliable for the purpose of the reactor plant safe operation. As a rule, it is brought to a level which is 10% as high as the current power (when the RCP is started at 30% of N_{nom}, the setting is equal to 40% of N_{nom}).

Proceeding from the general considerations and based on the combination of all of the considered factors, a conclusion can be made that the RCP starting in an earlier inoperative loop at all power levels, beginning with 34% of N_{nom} and ending with 75% of N_{nom}, is outside the safety criteria limits. A growth in the reactor plant power to above the scram setting values is observed in these modes. It should be noted that the end temperature of the fuel cladding reaches 364 °C in the mode with the RCP starting in an earlier inoperative loop at a power level of 75% of N_{nom}, this leading to the fuel cladding mechanical properties to be lost.

Prior to the RCP startup in an earlier inoperative loop, it is more reasonable to lower the reactor plant power to a level of < 34% of N_{nom}. The fulfillment of the safety criteria will be observed in these modes, and, therefore, safe operation of the reactor plant as the whole will be ensured.

Table 4. Safe operating limits and calculated values of the mode with the RCP starting in an earlier inoperative loop at a power level of 30% of N_{nom}

Parameter	Safe operating limit	Actual calculated value
Maximum permissible reactor thermal power, % N_{nom}	100 ± 2	39.9
Maximum loop temperature difference, °C	30	13.4
Minimum critical power ratio, relative units	Not less than 1	8
Core coolant flow rate, m^3/h	80400 ± 1696	80400
Core inlet coolant T in any operating loop, °C	£ 290	284
Core coolant pressure, MPa	15.9 ± 0.2	15.9
Scram setting, % N_{nom}	107	40
Maximum fuel cladding temperature, °C	350	313.6

As an example, we shall compare the obtained data of the RCP starting mode at a power level of 30% of N_{nom} with the limiting values (Table 4).

The post-transient value of the reactor thermal power amounted to 39.9% of the rated value which does not exceed the maximum permissible safe operating limit for the 100% power with four RCPs in operation.

The maximum temperature difference in the loop was 13.4 degrees which is less than the operating limit of 30 degrees.

The minimum critical power ratio throughout the transient was less than a unity, and its minimum value was equal to eight.

The post-transient coolant flow rate through the reactor core was 80400 m^3/h which does not exceed the maximum permissible safe operating limit of 80400 ± 1696 m^3/h.

The coolant temperature at the reactor core inlet in any of the loops in operation does not exceed a value of 284 degrees which is 6 degrees as low as the safe operating limit.

The maximum pressure was 15.9 MPa which does not exceed the maximum value of the safe operating pressure of 15.9 ± 0.2 MPa with regard for the permissible scatter of values of 0.2 MPa.

The scram setting was changed to a power level of 40% of N_{nom}. We shall note that no scram takes place as the reactor's post-transient final thermal power of 39.9% of N_{nom} is reached, this having a positive effect on the safe reactor plant operation.

The maximum fuel cladding temperature was 313.6 degrees which does not exceed the operating limit of 350 degrees.

Therefore, the safety analysis criteria for the RCP starting in an earlier inoperative loop at a power level of 30% of N_{nom} are fulfilled, and the heat-engineering reliability of the reactor core cooling is ensured.

It can be noted based on the simulation results that the reactor plant power increase is the greater the lower is the coolant temperature in the loop with the earlier idle RCP upstream of the core inlet. Accordingly, the resultant action may lead to a number of negative effects, e.g., to the coolant boiling and the occurrence of a departure from nucleate boiling. There is a major deviation from the safe operating limits in such modes. In this case, no safe operation of the reactor plant takes place.

6. Conclusions

The KORSAR code was used to investigate the modes of the RCP starting in an earlier inoperative loop.

In accordance with the calculated values obtained, it can be stated that, when the reactor plant power decreases to 30% of N_{nom} prior to the RCP starting in an earlier inoperative loop, the heat-engineering reliability of the reactor core cooling and safe operation of the VVER-1000 reactor plant is ensured in this mode.

The nodalization scheme of the VVER-1000 primary circuit developed for the computational analysis makes it possible to add or to exclude the action of protective safety systems. This scheme can be used to analyze computationally the effects of other anticipated operational occurrences and emergencies such as, for example, modes involving the primary coolant flow rate disturbances and primary circuit breaks.

References

- Afrov AM, Andrushechko SA, Ukraintsev VF, Vasilyev BYu, Kosourov KB, Semchenkov YuM, Kokosadze EL, Ivanov YeA (2006) VVER-1000: Physical Principles of Operation, Nuclear Fuel, Safety. Logos Publ., Moscow, 488 pp. [in Russian]
- Asmolov VG, Blinkov VN, Kovalevich OM (2010) Fundamentals of the NPP Safety. MEI Publ., Moscow, 93 pp. [in Russian]
- Baklushin RP (2011) Operation of NPPs. Part 1. Operation of NPPs in Power Supply Systems. Part II. Handling of Radioactive Waste. Training book. NIYaU MIFI Publ., Moscow, 304 pp. [in Russian]
- Baklushin RP (2012) Operating Modes of NPPs. Izdatelskiy Dom MEI Publ., Moscow, 532 pp. [in Russian]
- Belozerov VI, Sergeyev VV, Kazantsev AA, Pozdnyakov AN, Kanyshev My (2008) A neutronic and thermal-hydraulic model of a VVER-1000 reactor for personnel training. Izvestiya Vysshykh Uchebnykh Zavedeniy. Yadernaya Energetika 2: 99–106. [in Russian]
- Belozerov VI, Zhuk MM (2012) VVER-1000 Reactor Physics and Operating Conditions. Dom Pressy Publ., Minsk, 144 pp. [in Russian]
- Bukrinsky AM (1982) Emergency Transients at NPPs with VVER Reactors. Energoizdat Publ., Moscow, 142 pp. [in Russian]
- Dragunov YuG, Bykov MA, Vasilenko VA, Migrov YuA (2019) Experience of the application and development of the KORSAR code for the VVER NPP safety analysis. http://www.gidropress.podolsk.ru/files/proceedings/mntk2005/Конференция/Сторонние_организации/ФГУП%20НИТИ/07_Мигров%20Ю.А.pdf [accessed Apr 10, 2019] [in Russian]

- Gordon BG (2009) Fundamentals of Safety Regulation in the Use of Atomic Energy. Training book, MIFI Publ., Moscow, 263 pp. [in Russian]
- GOST R 50088-92 (1994) Water-Cooled Water-Moderated Nuclear Power Reactors (VVER). General Requirements to Physical Calculations. Gosstandart Rossii Publ., Moscow, 13 pp. [in Russian]
- Ivanov VA (1994) Operation of NPPs. Energoatomizdat Publ., St. Petersburg, 384 pp. [in Russian]
- Kazantsev AA, Sergeyev VV, Belozerov VI, Yefremov AYu (2009) Simulation of transients for a VVER-1000 reactor. Izvestiya Vysshykh Uchebnykh Zavedeniy. Yadernaya Energetika 1: 98–104. [in Russian]
- KORSAR/GP Code (2009) Software Certification Passport No. 263, dated 23.09.2009. NTTs YaRB Publ, Moscow. [in Russian]
- Margulova TKh (1994) Nuclear Power Plants. University training book (5th edn). IzdAT Publ., Moscow, 296 pp. [in Russian]
- Nigmatulin IN, Nigmatulin BI (1986) Nuclear Power Plants. Energoatomizdat Publ., Moscow, 168 pp. [in Russian]
- Petrosyants AM (1981) Nuclear Power. Nauka Publ., Moscow, 272 pp. [in Russian]
- RELAP5/MOD3 Code Manual (1995) Code Structure, System Models, and Solution Methods. National Engineering Laboratory, Idaho, 1418 pp.
- Safety of Nuclear Power Plants (1994) Handbook. Kontsern Rosenergoatom Publ., Moscow, 250 pp. [in Russian]
- Vygovsky SB, Ryabov NO, Semenov AA, Chernov YeV, Bogachek LN (2011) Physical and Design Features of Nuclear Power Plants with VVER Reactors. Training book. NIYaU MIFI Publ., Moscow, 376 pp. [in Russian]
- Weinberg A, Wigner E (1961) Physical Theory of Nuclear Reactors. Inostrannaya Literatura Publ., Moscow, 733 pp. [in Russian]