DETERMINISTIC SAFETY ANALYSES OF BREST-OD–300 REACTOR FACILITY

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Abstract. As part of the power unit with reactor BREST-OD-300 project justification, deterministic safety analysis with imposition of postulated single failure of systems and equipment, or human error on the initiating event (IE) was held in JSC "NIKIET". Work was done for the version of the core with jacketless TVS. The calculations were performed using the verified dynamically bound neutron-physical and thermal-hydraulic software package DINAR.

Results of the safety analysis of the power unit with reactor BREST-OD-300 are presented in the paper for up to four OE violations in normal operation (VNO), one from each group of the internal effects listed below. Selected VNO initiating events, accompanied by the greatest disturbance and deeper relative to the nominal power deviations of parameters important to safety. Initial events of violations in normal operation were considered including the following groups of effects:

- Initiating events that lead to the unauthorized introduction of positive reactivity;

- Initiating events that lead to the disruption of the heat sink from the core;

Scenarios of IE with the imposition of a single failure of the safety systems, or human error take into account the requirements of the Russian nuclear power industry standards and regulations, according to which the power plant security must be provided in any of the initial event carried in project with imposition of one independent from the initiating event failure of any of the following safety systems: an active element or a passive element having mechanical moving parts, or a passive element with no moving parts, having a probability of failure of safety functions 10 -3 or more, or one independent from the initiating event of personnel mistakes in accordance with the principle of single failure. In addition to the one failure of the one of the elements listed above and independent from the initiating event, all failures resulting from this single failure or initial event, as well as non-detectable failures in the operation of the AC elements affecting the development of the accident were taken into account.

In conclusion, the article presents the results of the beyond project accident calculation with loss of power supply and the refusal of the two reactor shutdown systems.

As the security criteria of the reactor facility in violation of the normal operation the exceedance of the established design limits of the power unit parameters were taken.

Key Words: safety analyses, BREST-OD-300, safety systems, security criteria.

1. Introduction

Deterministic safety analysis with imposition of postulated single failures of systems and components, or human errors on an initiating event (IE) was held at JSC "NIKIET" in accordance with NP-018-05, par. 15 as part of the BREST-OD-300 nuclear power unit design justification. This work was carried out for the reactor core with shroudless fuel assemblies [1].

The calculations were performed using the verified dynamically bound neutronic and thermal-hydraulic software package DINAR [2], where three-dimensional kinetic equations are solved in the 26 group diffusion approximation using a spatial HEX-Z approach.

The reactor core is modeled with 169 FAs in central and peripheral areas, 4 equivalent channels in the side reflectors and channels of the passive feedback system (PFS). Each channel is described in the one-dimensional approximation with fragmentation of the reactor core height in 11 sections. Fuel elements are presented in a three-layer model that includes: fuel, gap and cladding. Thermal-hydraulic processes in four loops of the reactor primary circuit are described in the one-dimensional approach, taking into account the following components operation: steam generators (SG), reactor coolant pumps (RCP), the emergency core cooling system (ECCS) and others. Water and steam lines of the secondary circuit are also modeled in the one-dimensional approximation. Module for the secondary circuit calculation includes a description of the feedwater heater (FH), hydroturbine feedwater pumps (HTFWP), water and steam pipe systems, isolation feedwater valves (IFWV), isolation steam valves (ISV), steam generator safety devices (SGSD), main safety valves (MSV), control valves and others.

The report presents the results of the BREST-OD-300 unit safety analysis of four IEs resulting in anticipated operational occurrences (AOO), one from the each group of internal effects listed below. Selected AOO initiating events, accompanied by the greatest disturbance and deeper deviations relative to the nominal safety important parameters. IEs of AOO related to the following groups were considered:

- initiating events leading to the unauthorized introduction of positive reactivity;

- initiating events leading to the violation of the heat removal from the reactor core;

Scenarios of IEs with imposition of a single failure or human errors consider the requirements of the nuclear power rules and regulations which are specified in the appropriate documents (NP-001-15), according to which the NPP safety should be provided for any design-basis IE and in accordance with the principle of single failure with imposition of the one failure of any of the following safety systems independent from IE: an active element or a passive element having mechanical moving parts, or a passive element without moving parts, having a probability of safety functions failure equal to 10^{-3} or more, or one human error independent from IE. In addition to the one failure of one of the above elements independent from IE all failures resulting from this single failure or initial event, as well as non-detectable element failures in the operation of NPP affecting the accident development were taken into account.

In accordance with NP-082-07, par. 2.3.1.4 reactor facility basic design envisages two reactor shutdown systems, while each can independently transfer the reactor core in the subcritical state and maintain it, taking into account the principle of single failure or human error. Taking into account that the second shutdown system – emergency power reduction system (EPRS) has an efficiency significantly superior than an efficiency of seven control members of emergency protection system (EPS), further consideration of initiating events requiring EP activation takes into account only the operation of the EPS (EPRS initiation in response to EP signal is not considered). In addition a failure of the most effective EP control member is supposed (requirements of NP-082-07, par. 2.3.2.2).

Triggering the reactor shutdown systems conservatively assumes maximum delay times in their actuation and control members introduction into the reactor core.

For the most severe groups of initiating events for a reactor facility, namely: introduction of positive reactivity and deterioration of the heat removal from the reactor core, the imposition of multiple failures of systems and components is considered. In the last case, the scenario with complete non-availability of both reactor shutdown systems is allowed.

As the reactor safety criteria in case of AOO the power unit safe operation parameters were taken as being not exceeding the safe operation limits: maximum design limit of the fuel cladding temperature (from 800 to 900 °C for 100 min) and the safe operation limit of the coolant temperature at SG inlet (620 °C).

The basic technical approaches to the development of BREST-OD-300 are stated in [3-5].

2. Initiating events leading to a positive reactivity introduction

Unauthorized removing of two groups of automatic regulation (AR) control members

As an initial state the power unit stationary operation at the rated power is considered. This condition is characterized by the maximum specific volumetric and linear fuel element heat rating, as well as the minimum margins up to ultimate loads.

Operational reactivity margin at the rated power is assumed to be 0.65 β and corresponds to the reactor state till implementation of the neptunium effect (during the first 10 days). For the power control two groups of AR control members are provided, which are introduced into the reactor core at half of their travel, while reactivity compensator’s control members are withdrawn from the reactor core in accordance with the control algorithm.

So as IE there is considered the failure of control and protection system (CPS) when two groups of AR control members are withdrawn alternately with a maximum speed from the reactor core at half of its travel, that results in the positive reactivity (0.65 β) introduction during 30 s. Also the failure of automatic locking of continuous withdrawal of AR control members from the reactor core for more than 150 mm is imposed on the IE, otherwise the considered initial event can not occur.

Scenario of the IE development during design-basis operation of systems and components

Spontaneous insertion of positive reactivity causes a growth in power and the reactor core heat-up. To avoid the undesired development of a transient and its progression into an accident, the BREST-OD-300 design stipulates actuation of the following systems:

- emergency power reduction (EPRS) in response to a signal of the setup of 110 % N_{nom} being exceeded;

- EPRS in response to a signal of the setup of 115 % N_{set} being exceeded;

- EPRS when there is a mismatch between the power and the coolant flow rate (the pressure level);

- fast controlled power reduction (FCPR-4) when the coolant temperature at the outlet from the core central zone reaches 580 °C.

Actuation of at least one of these systems leads to a timely power reduction or the reactor shutdown without scram. However, the development of a transient may depend to a great extent on what immediately causes the spontaneous travel; thus, when the automatic control rod withdrawal has been initiated by a failure of the power controller (generates an overestimated value), the emergency power reduction system will not operate in response to the N_{set} signal, and the FCPR-4 malfunction may also be assumed as the potential accompanying failure since, in this case, power setback is also ensured by the AC rod operation.

The reactor will be shut down by the EPRS in response to a signal of the current power exceeding a setup of 110 % N_{nom}.

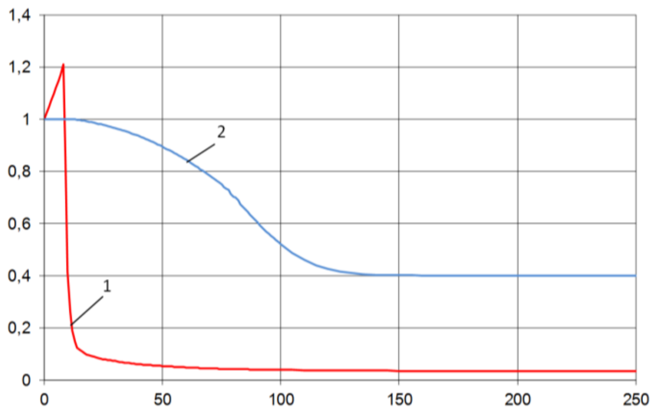
Scenario of the IE development accompanied by a simultaneous postulated SS component failure or personnel error

Omission of the EPRS signal (110% of N_{nom}) has been assumed to be the worst scenario with the IE being accompanied by a single SS component failure or a personnel error.

In accordance with the principle of a single failure along with a failure of the preliminary protection system (FCPR-4) and a potential EPRS system failure (with a setup of 115 % N_{set}), the omission of the EPRS signal of the setup of 110 % N_{nom} being exceeded was taken as the accompanying failure, and the reactor is shut down in response to an emergency protection signal of 120 % N_{nom}. The scram delay time was conservatively assumed to be 1 sec.

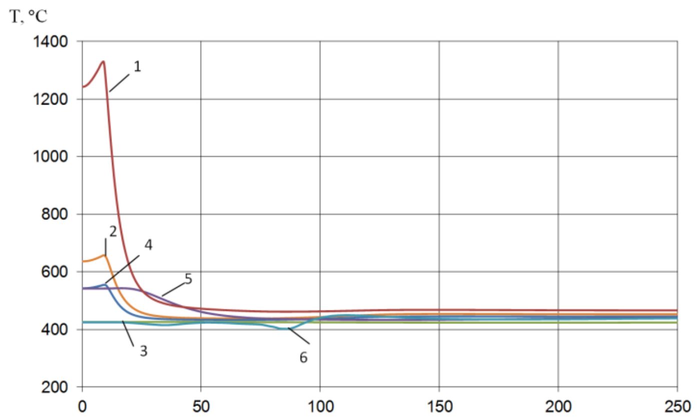
The transient calculation results (with a scram at 120 % N_{nom}) are shown in Figs. 1 and 2.

N;G, rel. units



Time, s

FIG. 1. Behavior of the reactor power and the coolant flow rate through the core: 1 – reactor power, 2 – coolant flow rate through the core.



Time, s

FIG. 2. Behavior of the fuel, fuel cladding and lead coolant temperatures at the core and SG inlet/outlet: 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at the core inlet, 4 – coolant temperature at the core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at SG outlet.

After the scram, with a power setback down to 15 % N_{nom}, the RCP flow rate decreases automatically to 40 % G_{nom}, with decay heat removed by the normal cooldown system (NCDS). The maximum fuel cladding temperature does not exceed 650 °C, and the lead coolant temperature at the SG inlet does not practically exceed the rated value. The maximum design fuel cladding temperature limit (from 800 to 900 °C for 100 min) and the safe operation limit for the coolant SG inlet temperature (620 °C) are not violated.

Scenario of the IE development accompanied by multiple failures of systems and components or personnel errors.

Operation of equipment in the event this scenario takes place has been assumed in accordance with Table 1.

TABLE 1: IE DEVELOPMENT SCENARIO

Sequence of system and component actuation events with regard for the design logic of their operation	**Actuation setup**	**Status of systems and components**	**Comment, expert appraisal of actions and consequences**
1) Actuation of FCPR-4	Mismatch between the power and the coolant pressure level	Failure	Core power and temperature growth
2) Actuation of EPRS	110 % N_{nom}	Failure	Core power and temperature growth
3) Actuation of EPRS	115 % Nset	Failure	Core power and temperature growth
4) EPRS	Mismatch between the power and coolant pressure level	Failure	Core power and temperature growth
5) Scram	120 % Nnom	Failure	Core power and temperature growth
6) Actuation of FCPR-4	580 °C at the reactor central zone outlet	Failure	Core power and temperature growth
7) Scram	600 °C at the reactor central zone outlet	Scram failure	Core power and temperature growth
8) Actuation of SGSD, SIV and FWIV, trip of the HTFWP	620 °C at the reactor central zone outlet	Design-basis operation	Core power and temperature growth, loss of heat removal from the primary circuit
9) EPRS	Signal to close all FWIVs	Failure	Core power and temperature growth

CONTINUED TABLE 1

Sequence of system and component actuation events with regard for the design logic of their operation	**Actuation setup**	**Status of systems and components**	**Comment, expert appraisal of actions and consequences**
10) Trip of RCPs	520 °C at the SG outlet	Design-basis operation	Switchover to natural circulation, deterioration of heat removal from the core, negative reactivity insertion by PFBS
11) EPRS	Trip of more than 2 RCPs	Design-basis operation	Reactor shutdown
12) NCDS	430 °C at the ECCS outlet	Failure	Primary circuit heat-up
13) ECCS	450 °C at the ECCS outlet	Failure of 2 ECCS loops	Removal of decay power

This scenario has the initiating event accompanied by failures of four EPRS signals and scram failures in response to two signals.

The calculation results are presented in Figs. 3-5. The fuel cladding temperature does not exceed 800 °C, and the SG inlet coolant temperature is ≈ 608 °C. The maximum design fuel cladding temperature limit and the safe operation limit as to the coolant SG inlet temperature are not violated.

Two ECCS loops (two have failed) are enough for the reactor decay heat removal.

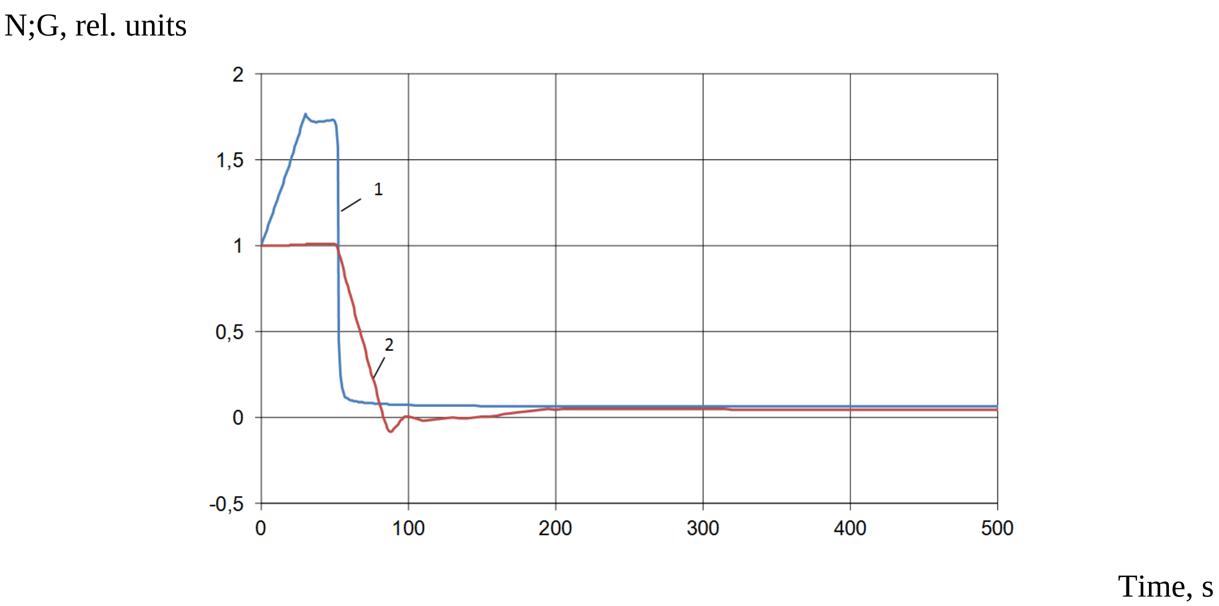


FIG. 3. Behavior of the reactor power and the coolant flow rate through the core

1 – reactor power, 2 – coolant flow rate through the core.

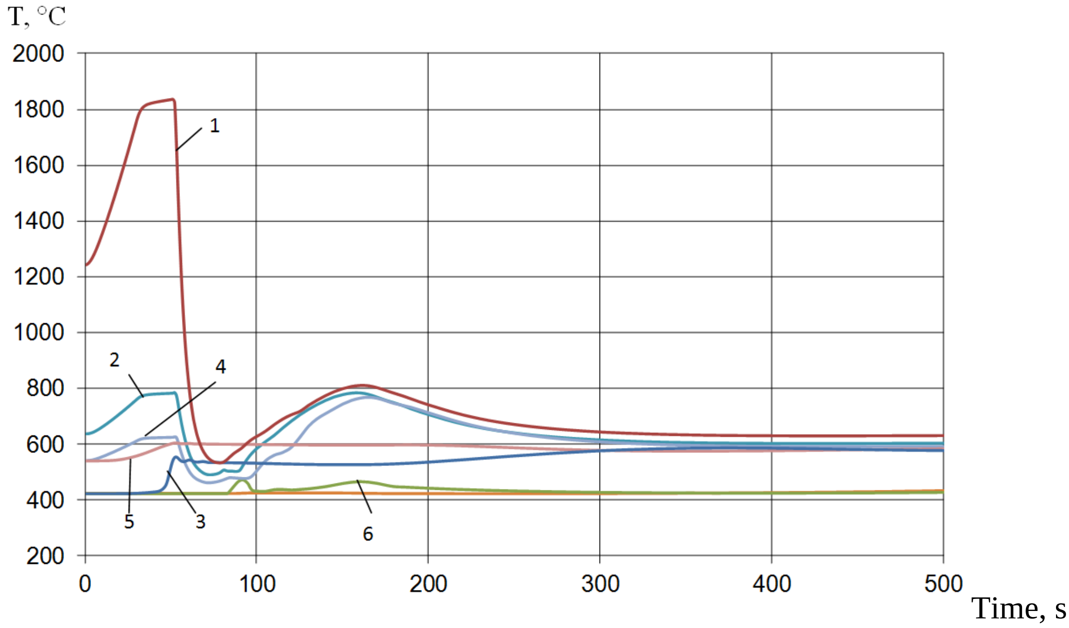


FIG. 4. Behavior of the fuel, fuel cladding and lead coolant temperature at the core and SG inlet/outlet: 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at the core inlet, 4 – coolant temperature at the core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at SG outlet.

3. Initiating event leading to deterioration of heat removal from the core

Loss of system unit power

The considered IE leads to the loss of power supply to normal operation consumers with all main components of the primary and secondary circuits being simultaneously disconnected.

Rated power operation of the unit is considered as the initiating event. In this condition, the loss of system unit power will lead to violent and deep perturbations of parameters and the greatest possible deterioration in the core heat removal.

Scenario of the IE development with design-basis operation of systems and components

The initiating event leads to scram, termination of power supply to all drivers of the CPS rods and their passive insertion into the core (including scram rods), and the reactor is shut down. All RCPs, feedwater pumps and the turbine are tripped. In addition to negative reactivity of the CPS rods, negative reactivity of the passive feedback system channels is introduced passively into the core. Power is supplied to the monitoring system equipment by the auxiliary power system. Diesel generators of the auxiliary power system are started by the automatics action to power consumers of the nuclear plant safety systems. Decay heat is removed by the emergency core cooling system (ECCS), while the NCDS does not operate in this mode.

Scenario of the IE development accompanied by a postulated safety system component failure or personnel error

A failure of a single ECCS channel (two loops) has been assumed as the worst scenario with the IE being accompanied by an SS component failure or a personnel error.

The reactor has been shut down by scram in response to a loss of auxiliary power. The scram signal is backed up by the EPRS signals in response to the RCP and feedwater pump trips.

And the core and primary circuit temperatures decrease abruptly without permissible limits being exceeded.

Figs. 5 – 7 show the transient of a long-term decay power removal by the ECCS channel remaining in operation. The ECCS is connected after ≈ 4000 sec as the coolant temperature at the emergency cooling channel inlet increases to 450 °C. Further, as can be seen from Fig. 5, the fuel, fuel cladding and lead coolant temperatures in the core and primary circuit do not exceed 590 °C in the process of operation of two ECCS loops (after 3600 sec). The maximum design fuel cladding temperature limit and the safe operation limit for the SG inlet coolant temperature are not violated.

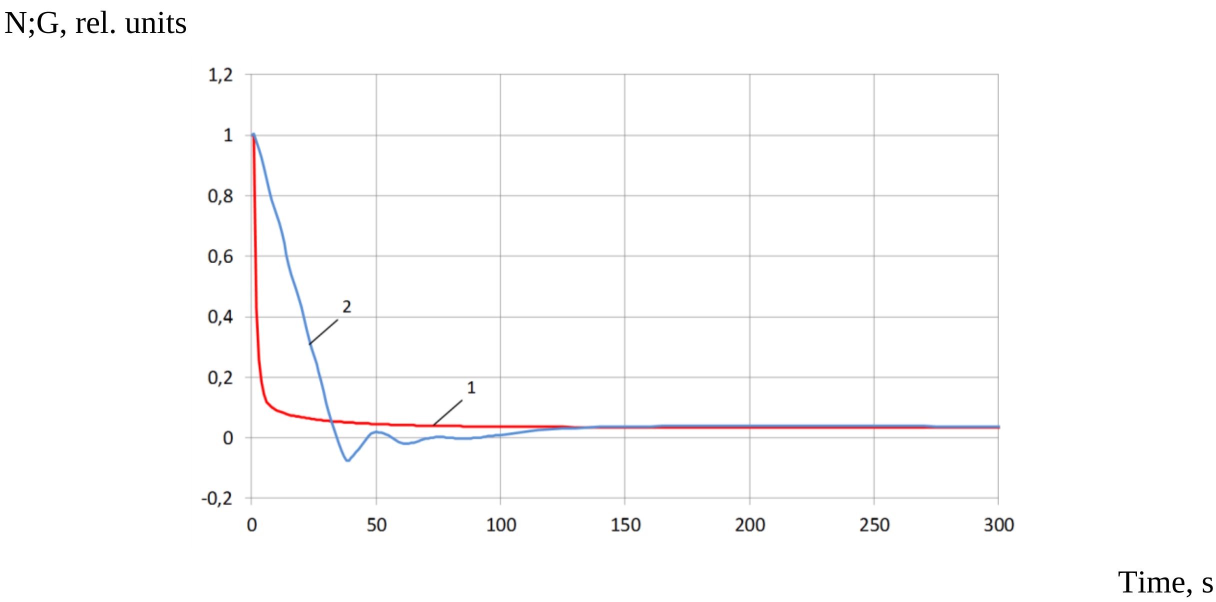


FIG. 5. Behavior of the reactor power and the coolant flow rate through the core: 1 – reactor power, 2 – coolant flow rate through the core.

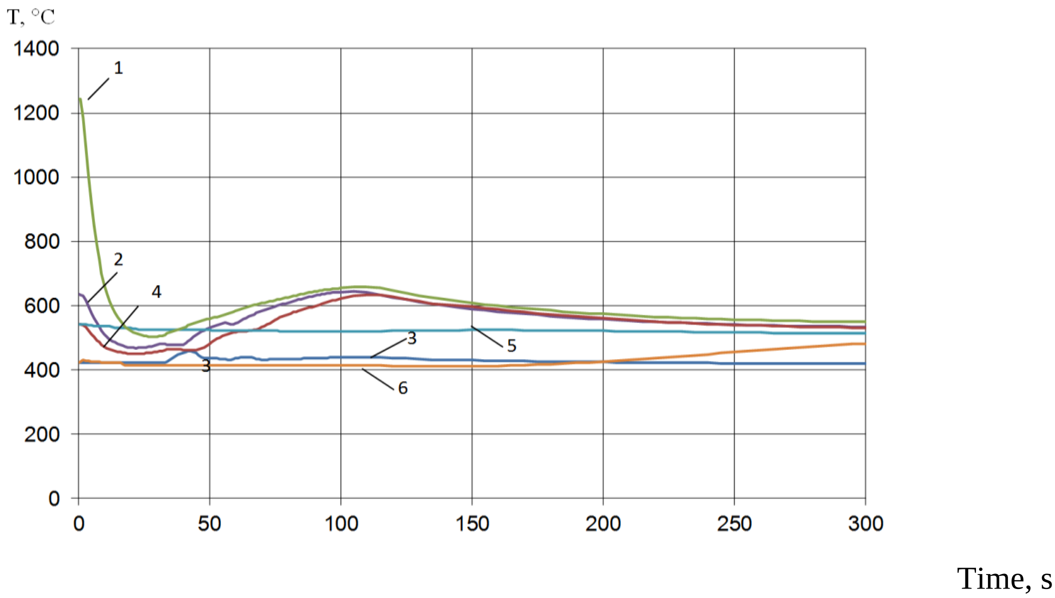


FIG. 6. Behavior of the fuel, fuel cladding and lead coolant temperatures at the core and SG inlet/outlet (until the 300th sec): 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at the core inlet, 4 – coolant temperature at the core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at SG outlet.

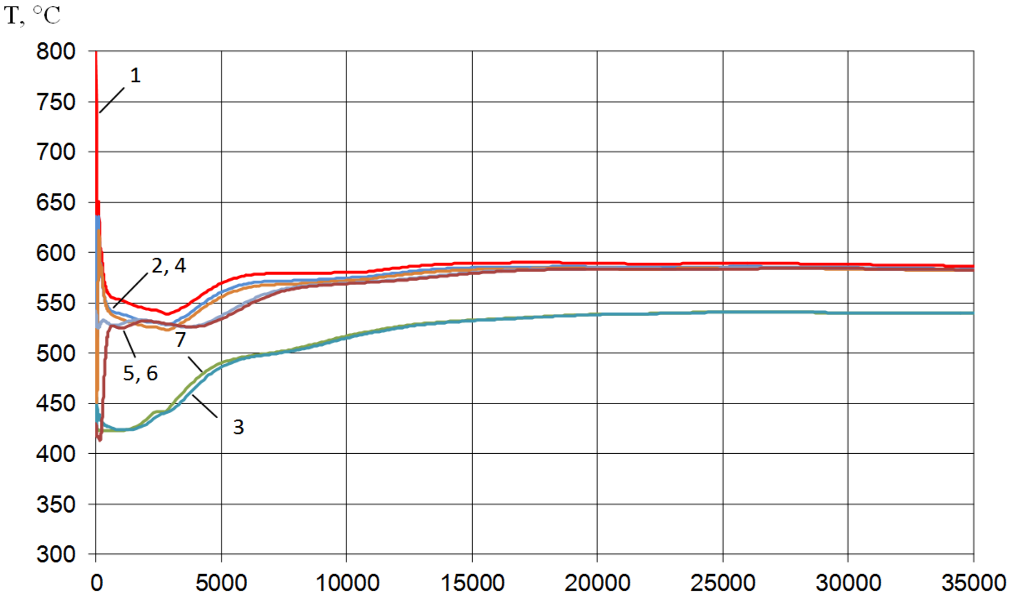


FIG. 7. Behavior of the fuel, fuel cladding and lead coolant temperatures at the core and SG inlet/outlet: 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at the core inlet, 4 – coolant temperature at the core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at SG outlet, 7 – coolant temperature at ECCS outlet.

Loss of unit's system power supply with simultaneous multiple failures.

In the initial state, the reactor operates at a rated power.

In this IE development scenario, failures of two ECCS loops and multiple EP and EPRS failures are postulated, in responce to the following signals:

a) EP actuation:

- increase of lead coolant temperature above 600 °C at the reactor core center outlet;
- loss of auxiliary power of the unit;
- faults of EP circuits in "two-out-of-three" channels of one set;
- loss of CPS bus voltage.

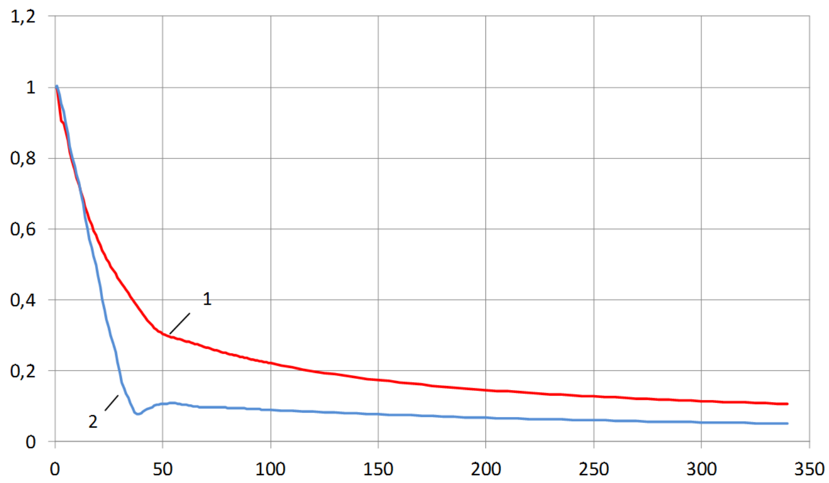
b) EPRS actuation:

- increase of power level up to the setpoint, which depends on the total coolant head level;
- shutdown of more than two RCPs;
- shutdown of all FWPs with failed start of the backup pump;
- shutdown of all HTFWPs with failed start of the backup pump;
- vacuum breaking in a turbine condenser;
- faults of EPRS circuits in "two-out-of-three" channels of one set.

Active gate valves of ECCS, which feed from an emergency power system (diesel generators) in the emergency conditions, in this case start operating according to the design conditions. Besides, ECCS will be brought into action by passive gate valves which are placed in parallel to the active ones. Regulating gates of ECCS with passive actuators are controlled by passive regulators in accordance with the air temperature in the data channels; the air temperature within the controlled range corresponds to the lead temperature at the outlets of the corresponding ECCS's heat exchangers in the range of 450 °C - 480 °C. Failures of two ECCS loops are postulated.

Due to high heat inertia of the primary reactor circuit (lead coolant, metal structure, steel lining of the primary circuit), fairly low heating-up rate is observed. Calculation results are shown in Figs. 16 – 18. The reactor power in the transient mode is decreased solely by the passive operation of PFBS and the reactor core's temperature feedbacks. The maximum fuel cladding temperature in the beginning of the transient process for no more than 50 s is greater than 800 °C (≈ 890 °C at maximum), subsequently, during operation of two ECCS loops (after 3600 s), the temperatures of the fuel claddings and the coolant at the reactor core outlet do not exceed 650 °C. In the considered scenario of IE development the maximum design limit for the fuel cladding temperature is not exceeded, but the safe operation limit in terms of coolant temperature at the SG outlet will be exceeded. In this case, safety systems are assumed to cut off the steam generators from water and steam (in response to coolant temperature at the reactor core outlet exceeding 620 °C), and the steam from the SG is assumed to be released through SG SD into the atmosphere, which prevents rupture of the steam generator tubes.

N;G, rel. units



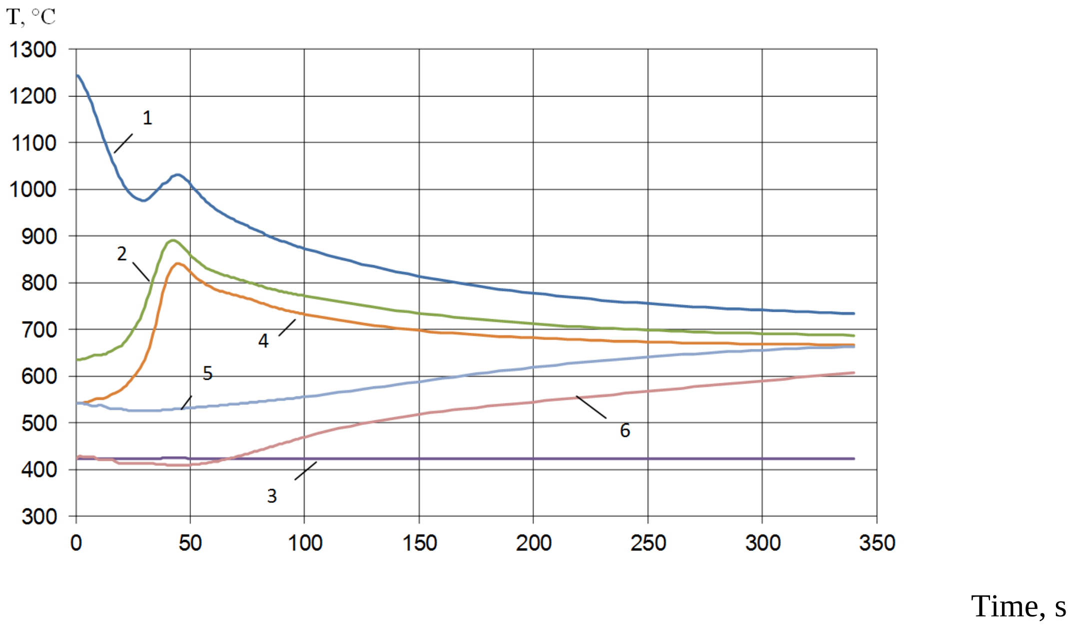
Time, s

FIG. 8. Behavior of reactor power and coolant flow rate through the reactor core: 1 – reactor power, 2 – coolant flow rate through the reactor core.

Conclusions

The presented results of the deterministic safety analysis show that all of the analyzed conditions of AOO caused by IEs with simultaneous postulated single failures of systems and components do not lead to violation of the design limits. This proves compliance of the design of the unit with BREST-OD-300 reactor with current safety norms and regulations.

A deterministic safety analysis of BREST-OD-300 in case of AOO IEs with multiple simultaneous failures of systems and components has also been performed at JSC NIKIET [6]. This work identified all IE development scenarios with maximum amount of simultaneous postulated failures of normal operation systems and safety systems not leading to violation of safe operation limits. Considered series of normal operation systems' and safety systems' failures consisted of a minimum of ten failures. This analysis showed high level of safety of the unit with BREST-OD-300 reactor and its resistance to accidents with multiple probable simultaneous failures of systems and components.



FIF. 17. Behavior of fuel, fuel cladding and lead coolant temperatures at reactor core's and SG's inlet and outlet (up until 350 s): 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at reactor core inlet, 4 – coolant temperature at reactor core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at SG outlet, 6 – time, s.

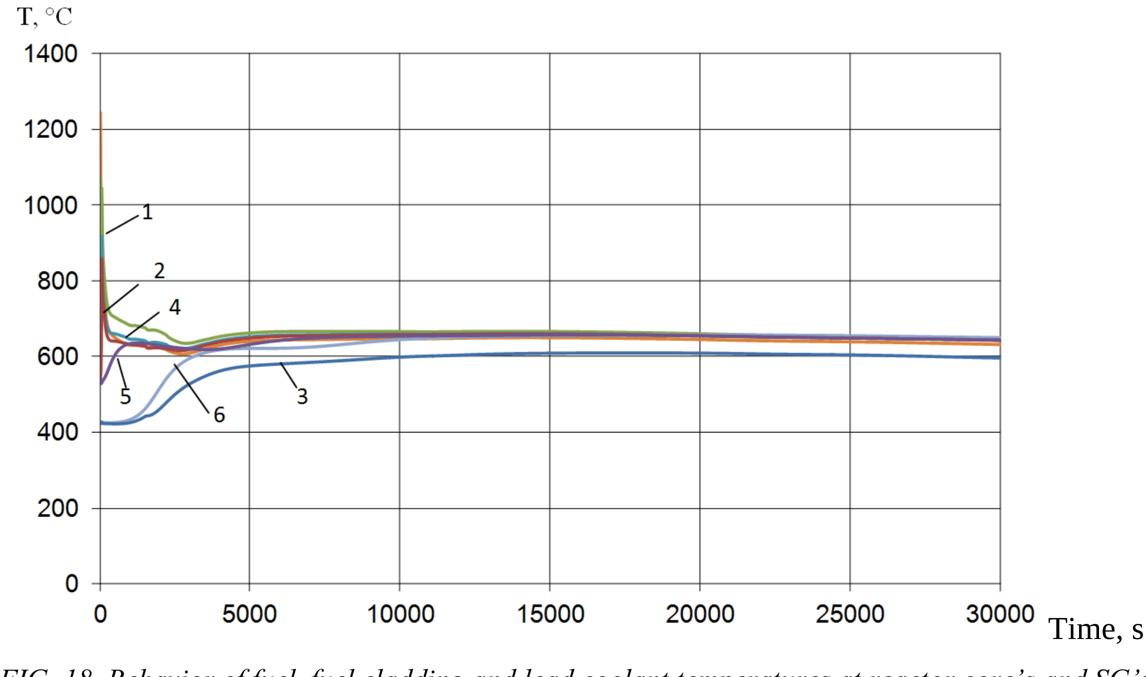


FIG. 18. Behavior of fuel, fuel cladding and lead coolant temperatures at reactor core's and SG's inlet and outlet: 1 – maximum fuel temperature, 2 – maximum fuel cladding temperature, 3 – coolant temperature at reactor core inlet, 4 – coolant temperature at reactor core outlet, 5 – coolant temperature at SG inlet, 6 – coolant temperature at ECCS inlet, 6 – time, s.

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