Safety Assurance for BN-1200 Power Unit During Accidents

S.F. Shepelev, S.A. Rogozhkin, S.L. Osipov, V.I. Evseev, A.M. Anfimov

Joint Stock Company “Afrikantov OKB Mechanical Engineering” (JSC “Afrikantov OKBM”), Nizhny Novgorod, Russia

Abstract. The power unit with the BN-1200 reactor plant was analyzed considering requirements imposed to innovative nuclear technologies, including IAEA recommendations.

The BN-1200 design considers the following types of beyond-design accidents:

- heat-removal accidents with loss of power, failures of active systems of reactor shutdown, failures of normal cooldown systems and active elements of emergency cooldown;
- accidents with positive reactivity introduction and failures of active reactor shutdown systems;
- accidents with blockage of fuel assembly flow area.

Design safety analysis for beyond-design accidents was made by verified codes, developed under the Federal Target Program.

A complex approach, which analyzes main processes and phenomena taking place in the reactor plant and power unit rooms during accident, is used for safety assurance.

The analysis results show that evacuation of population is not required for the expected exposure doses outside the nuclear power plant.

Key Words: Accidents, Integral Computer Code, Liquid-Metal Sodium, Reactor BN-1200, Safety

1. Introduction

The BN-1200 reactor plant [1] is developed in compliance with the Federal Target Program “New-generation nuclear power technologies for the 2010-2015 period and till 2020” and “Rosenergoatom” Concern long-term program of action. The goal is to create a commercial fast reactor of increased safety.

The design of power unit with the BN-1200 reactor plant is an innovative design of nuclear technology and should meet the requirements of Generation IV International Forum - GIF [2].

The obligatory requirement of innovative nuclear technology is to prevent evacuation of population outside a nuclear power plant site in the case of a beyond-design accident.

The BN-1200 design considers three main types of beyond-design accidents:

- heat-removal accidents with loss of power, failures of active systems of reactor shutdown, failures of normal cooldown systems and active elements of emergency cooldown;
- accidents with positive reactivity introduction and failures of active reactor shutdown systems;
- accidents with blockage of fuel assembly flow area.

In the analysis of accidents the following main requirements for the BN-1200 design were considered:

- all initial events must be considered in the design; and safety, implemented both by combination of inherent self-protection properties and design decisions and by routine measures, must be proved;

- combination of an initial event with failures of all active elements of normal operation and safety systems, single failures of passive elements of systems, errors of personnel.

Design safety analysis of beyond-design accidents was made by verified Russian codes SOCRAT-BN and EVCLID, which analyze all basic processes and phenomena taking place in a reactor plant and power unit rooms during accident.

2. Safety Basis of BN-1200 Power Unit

High level of safety, requirements for which become more severe because of accidents occurred at nuclear power plants, could be achieved only by optimal combination of inherent self-protection and design decisions.

The basic criterion, which characterizes nuclear power plant safety, is limitation of possible effect of nuclear power plant on population in accidents to the level at which the necessity of population evacuation is eliminated. As nuclear power plant improves, requirements for safety become more severe, sizes of the area of possible evacuation are minimized.

New-generation nuclear power plants are given a task to prevent completely the necessity to evacuate population in the region of nuclear power plant location in any technically possible accidents. The term “technically possible accidents” is regarded as any initial events, even those of low probability, which are combined with failures of all active safety systems, singular failures of passive elements (being provided with mechanically driving parts), which are provided by the design to limit consequences of this initial event, failures of safety-related active systems of normal operation, and erroneous action of personnel.

At this approach, the BN-1200 design [3] is given a task to limit radioactive releases into the environment in such a way as not to exceed the level of population radiation exposure on the site boundaries and beyond them, set by regulatory documents, which requires population.

The BN fast reactors possess a number of inherent self-protection properties, which include the following:

- high heat capacity of the reactor, which provides a low temperature increase rate of sodium in the primary circuit in case of complete cessation of heat removal after emergency protection actuation;
- introduction of negative reactivity in case of sodium boiling, which provides reactor power decrease in a core overheating accident;
- stable negative coefficient reactivity of reactor power and core temperature, which provides limitation of power increase in case of unauthorized introduction of positive reactivity.

BN-1200 reactor safety (Figure 1) is also provided by the following design decisions:

- arrangement of the control and protection system which prevents simultaneous withdrawal more than one working member from the core, thus, prevents introduction of high positive reactivity;
- application of a passive system based on hydraulically suspended rods (PEP-H), which are suspended over the core in an uncoupled position and, when flowrate decreases lower than 50 % from the nominal one, drop into the core by gravity, providing reactor transition into the subcritical state. This system prevents dangerous consequences of temperature increase in the core in case of stoppage of main circulating pumps and emergency protection failure;

- application of an additional passive system of absorber rods based on temperature principle of operation (PEP-T), which are drop into the core in all emergencies after considerable temperature increase at the core inlet;
- connection of the emergency heat removal system directly to the primary circuit through autonomous heat exchangers placed in the reactor vessel. Air heat exchangers are connected with autonomous heat exchangers by the intermediate sodium circuit. In the all three circuits of each channel (primary (sodium), intermediate (sodium), and air one) heat is removed by natural circulation of coolants. In each air heat exchanger there are active and passive gates, in the autonomous heat exchangers there are ball valves, which open sodium circulation path directly through the core by gravity in case of cessation of forced sodium circulation through the reactor.

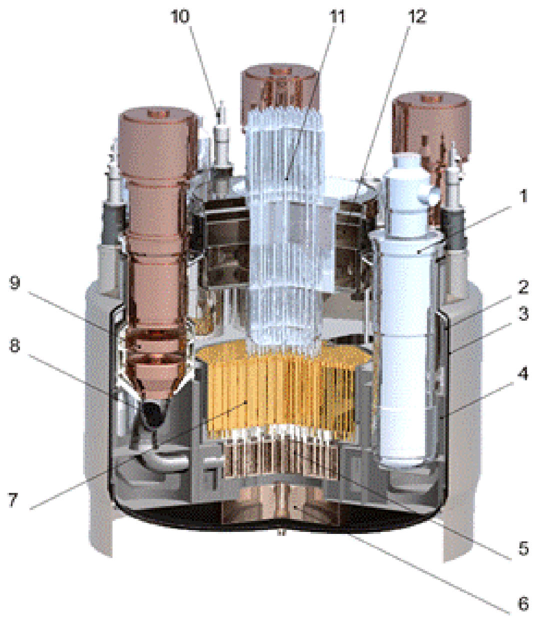

Figure 1. BN-1200 reactor

1 – intermediate heat exchanger; 2, 3 – main and guard vessels; 4 – support skirt; 5 – pressure chamber; 6 – fuel collection device; 7 – core; 8 – pressure pipeline; 9 – main circulating pump of primary circuit; 10 – autonomous heat exchanger; 11 – control rod drive mechanisms; 12 – rotating plugs

The BN-1200 safety level is increased due to application of new engineering solutions (Table 1).

TABLE 1: ENGINEERING SOLUTIONS FOR SAFETY ASSURANCE.

Engineering solutions for safety assurance	BN-600	BN-800	BN-1200
Sodium circuits:			
1) intermediate sodium/sodium circuit	+	+	+
2) jacketing of vessels with radioactive sodium	+	+	+
3) jacketing of pipelines with radioactive sodium	+	+	Eliminated pipelines
4) jacketing of secondary pipelines	-/+ (partial)	-/+ (partial)	-/+ (partial)

Emergency protection: 1) active 2) passive based on hydraulically suspended rods 3) passive based on temperature principle of operation	+ - -	+ + -	+ + +
Emergency heat removal system: 1) as a part of tertiary circuit 2) air heat exchangers are connected to the tertiary circuit 3) air heat exchangers are connected to the primary circuit	+ + (on one loop) -	+ + -	 +
Melted fuel retention system	-	+	+
Additional pressure safety device in the primary circuit	-	-	+
Fuel type	UO_2	UO_2, $(U,Pu)O_2$	$(U,Pu)O_2$, (U,Pu)N

The complex of inherent self-protection properties and design decisions prevents heavy core damage with large-scale fuel melting. Such event probability is estimated by the value $5 \cdot 10^{-7}$ per reactor per year.

Despite of inherent self-protection properties and made design decisions the BN-1200 design considers low-probability accidents with failure of active and passive elements and safety systems.

3. Brief Description of Integral Computer Codes SOCRAT-BN and EVCLID

Computer codes SOCRAT-BN and EVCLID [4, 5] are intended for end-to-end computations of safety-related reactor plant parameters considering simulation of main equipment of reactor plant and main power unit rooms in transients and emergency modes including heavy accidents with core melting.

The new-generation SOCRAT-BN and EVCLID codes consider mutual effects of main processes and phenomena taking place in transients and emergency modes, thus increasing accuracy of simulation of these processes. These codes permit to perform complex analysis of:

- neutronic and thermal-hydraulic processes in the core;
- thermal-hydraulic processes in reactor plant circuits;
- thermal-mechanical behavior of fuel elements;
- sodium boiling in reactor, melting and transfer of melt of fuel element claddings and fuel within reactor boundaries;
- high-energy thermal interaction of fuel element with coolant;
- possibilities of generation of repeated criticality in the process of melt transfer;
- fission product transfer in the reactor and main power unit rooms.

The SOCRAT-BN and EVCLID codes have been verified on Russian and foreign experimental data, including BN-600 operation data. Results of code verification confirmed adequacy of simulation of basic processes and phenomena taking place in accidents.

The reactor plant computation model [6] describes main equipment of the primary circuit, pipelines and equipment of the secondary circuit, pipelines and equipment of the emergency heat removal system.

4. Heat-removal Accident with Full Power Loss

The analysis of this accident postulates failures of all active reactor shutdown systems and active elements of the emergency heat removal system (outlet gates of air heat exchangers). In addition, it postulates failures one of rods of passive emergency protection based on hydraulic and temperature principles of operation.

The accident was analyzed assuming reactor plant cooldown by two of four channels of the emergency heat removal system.

Loss of power results in disabling of main circulating pumps (MCP) of primary and secondary circuits and cessation of feed water supply to steam generators (SG). Reactor power is decreased to the level of residual heat due to action of passive emergency protection rods. In the process of primary MCP coastdown the check valves of autonomous heat exchangers and some outlet gates of air heat exchangers open passively. As a consequence, natural circulation of coolants develops in the reactor and circuits of the emergency heat removal system (Figure 2).

The fuel element cladding temperature achieves values close to normal operation limit for a short period (Figure 2). The results of thermal-mechanical behavior of fuel elements show that in accident, as a result of different temperature extensions, a gap arises between fuel and fuel element cladding. In this case effective stresses in fuel element cladding do not achieve ultimate strength, thus preventing depressurization of claddings.

The reactor vessel temperature is near the nominal value (Figure 2). It means that reactor vessel integrity is kept during accident.

Coolant temperature increase in the reactor leads to decrease of the reactor gas cavity volume and actuation of the hydraulic lock of the reactor vessel. In case of actuation of the hydraulic lock of the reactor vessel radioactive fission products come into power unit rooms and into the environment. Activity, released into the environment in case of actuation of the hydraulic lock, is mainly caused by accumulation of inert radioactive gases in the pressurizer vessel during normal operation.

In 11 hours after accident beginning, temperatures in reactor begin to decrease (Figure 2), radioactive releases into the environment cease.

In accident, the exposure dose of a person of population will not exceed 0.046 mSv. In this case there is no necessity to take any protection measures for population living outside of the nuclear power plant site.

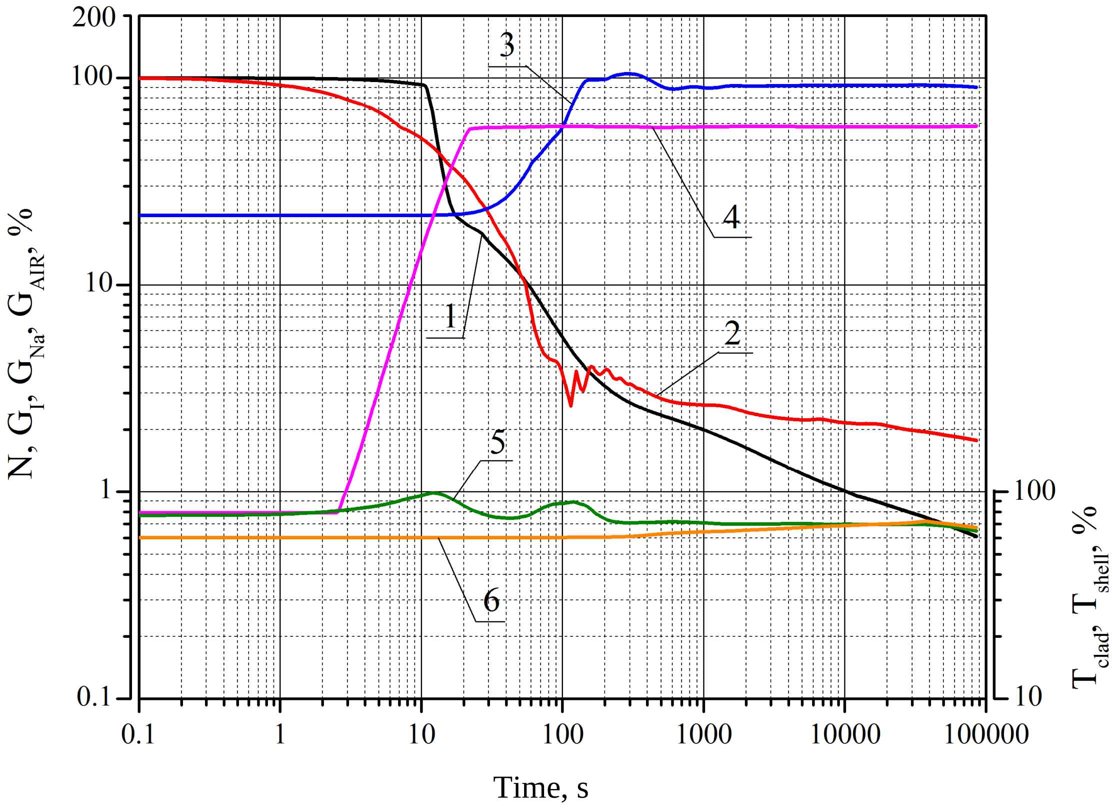

Figure 2. Main parameters in heat-removal accident

1 – reactor power (N); 2 – coolant flowrate through the core (G_I); 3 – coolant flowrate in intermediate circuit of emergency heat removal system (G_{Na}); 4 – coolant flowrate in air circuit of emergency heat removal system (G_{AIR}); 5 – ratio of fuel element cladding temperature to maximum allowable one (T_{clad}); 6 – ratio of reactor vessel temperature to maximum allowable one (T_{shell})

5. Reactivity accident

Sequential withdrawal of control rods from the core of the reactor operated on nominal power as a result of control in response to a false signal with failure of active reactor shutdown system is accepted as an initial event. Main circulating pumps of the primary and secondary circuits keep nominal speed. Heat is removed from the reactor through steam generators with keeping feed water supply in them. In addition, failure of one of temperature-based passive emergency protection rods is postulated

As a result of withdrawal of control rods from the core, the positive reactivity is introduced (Figure 3), and, as a consequence, reactor power begins to increase (Figure 4). The maximum power level ~ 220 % is achieved in 12 s after accident beginning. The power increase is limited by negative temperature effects (Figure 3), in which the Doppler effect contributes most of all.

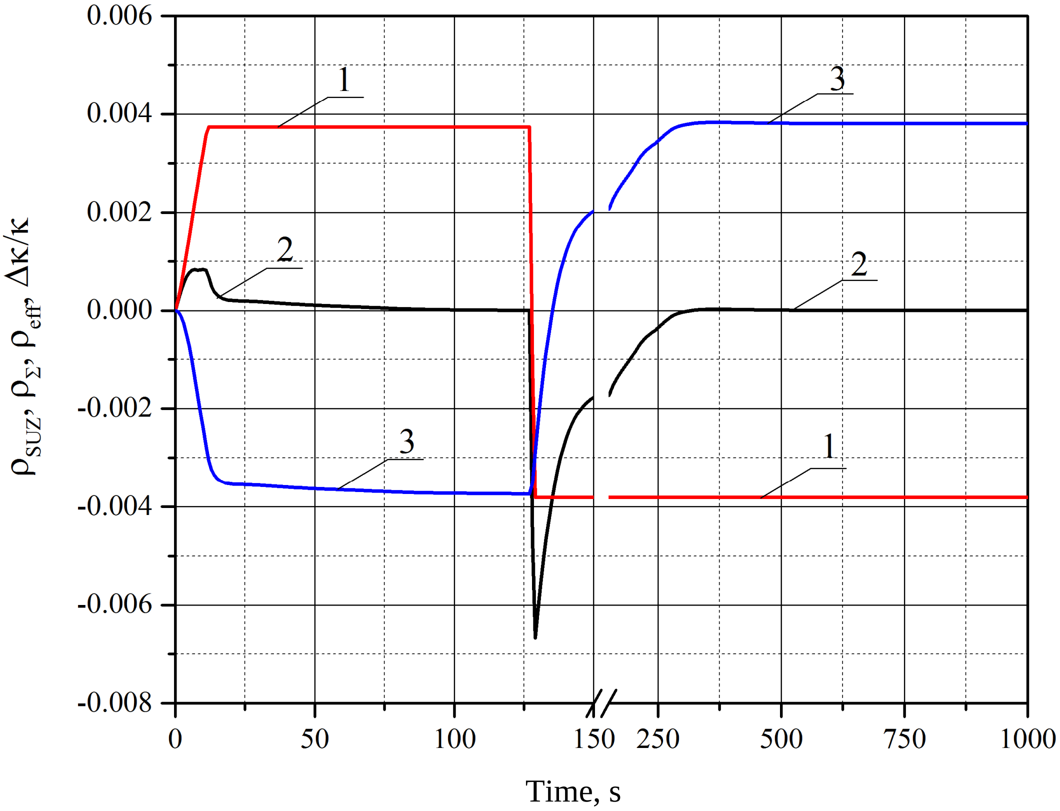

Figure 3. Reactivity in accident with withdrawal of control rods from core

1 – reactivity of control rods (ρ_{suz}); 2 – total reactivity (ρ_{Σ}); 3 – reactivity effects (ρ_{eff})

Reactor power decreases after actuation of temperature-based passive rods. Passive rods are introduced into the core by gravity after melting of a retaining device located above the core. After actuation of passive rods, reactor power decreases approximately to 35 % of the nominal value.

In the initial period of accident the fuel element cladding temperature could exceed the normal operation limit (Figure 4); equivalent stresses in claddings achieve ultimate strength. It leads to cladding depressurization. In some fuel elements fuel melting is possible in the core center area. After actuation of passive rods, temperatures in the reactor decrease lower than nominal values.

As result of depressurization of some fuel element claddings, fission gases (Xe, Kr) and volatile fission products (I, Cs) come into the primary coolant. As a consequence of degassing and evaporation processes, fission products come into the reactor gas cavity.

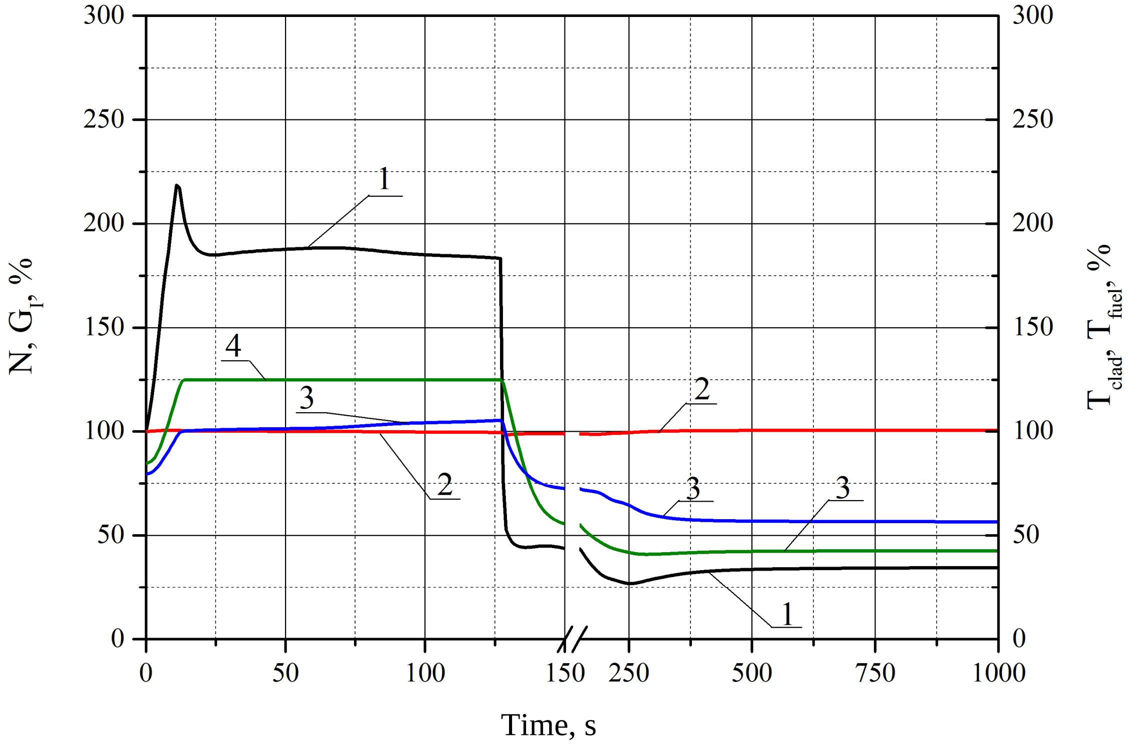

Figure 4. Main reactor parameters in accident with withdrawal of control rods from core

1 – reactor power (N); 2 – coolant flowrate through the core (G_I); 3 – ratio of fuel element cladding temperature to maximum allowable one (T_{clad}); 4 – ratio of temperature at axis of fuel stacks to temperature of fuel melting (T_{fuel})

In case of actuation of the hydraulic lock of the reactor vessel, radioactive fission products come into power unit rooms and into the environment. A main part of fission product coming out from fuel elements is retained in coolant. A major part of volatile fission products coming out into the reactor gas cavity is deposited on cold surfaces of reactor internals. Release activity is governed by inert radioactive gases.

In accident, the exposure dose of a person of population will not exceed 0.12 mSv. In this case there is no necessity to take any protection measures for population living outside of the nuclear power plant site.

6. Accidents with Blockage of Fuel Assembly Flow Area

Accident with complete instantaneous shutdown of a flow area of one of fuel assemblies during reactor operation at nominal power is considered. It is supposed that the flow area at the fuel element bundle inlet of a fuel assembly is disabled.

Fuel element blockage leads to rapid boiling of sodium and further melting of the assembly. As a result of melting of fuel element cladding, fission products are released into the upper mixing chamber of the reactor.

Emergency protection actuates by a signal of the delayed-neutron-based cladding tightness monitoring system, which sensors are located in the area of inlet ports of intermediate heat exchangers.

After emergency protection actuation, reactor power decreases to the level of residual heat. The reactor is cooled down as per the design pattern through the tertiary circuit.

Melt of fuel and steel in the emergency fuel assembly flows to the assembly bottom part, where it solidifies. In the process of transfer, fuel melt contacts with the fuel assembly wrapper tube. Approximately in 15 s after accident begins, fuel melt burns through the wrapper tube of the emergency fuel assembly and comes out into the inter-cell space. Thermal interaction results in fuel melt solidifying on the outer surface of neighboring fuel assemblies (Figure 5). Fuel melt does not move further.

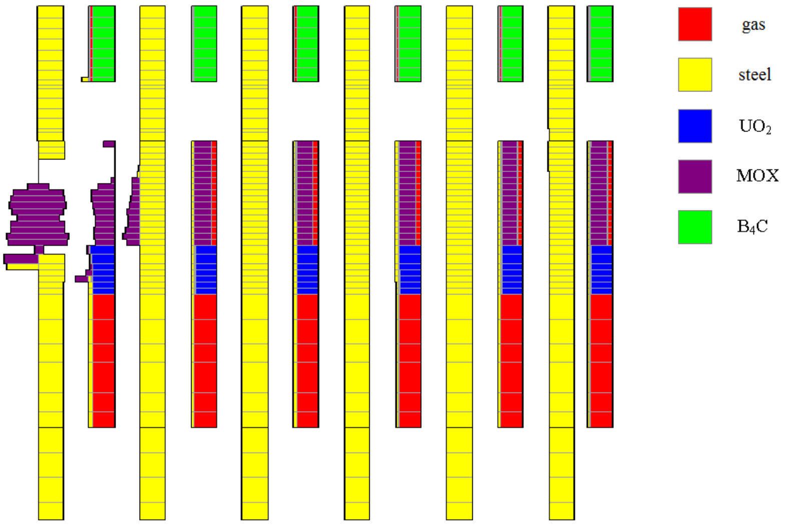

Figure 5. Distribution of materials in structural elements of the core in the accident with blockage of fuel assembly flow area (at the 75th s of transient)

The accident is followed by fission product release from damages fuel elements into coolant, and then, into the reactor gas cavity. Activity releases into power unit rooms and into the environment as a result of gas leakage from the reactor gas cavity through clearances.

In accident, the exposure dose of a person of population will not exceed 0.24 mSv. In this case there is no necessity to take any protection measures for population living outside of the nuclear power plant site.

7. Conclusion

The complex analysis has been performed for three types of beyond-design accidents, which scenarios consider low-probability initial events with failure of active and some passive elements and safety systems.

The accident with fuel assembly blockage is most important from the viewpoint of core technical condition. The scope of core damage is limited by melting of emergency fuel assemblies.

The considered accidents produce no considerable release of radioactivity into the atmosphere. Exposure doses of population are considerably less than the regulated value of 5 mSv/person per a year. In this case there is no necessity to take any protection measures, moreover, to evacuate population.

The performed analysis of beyond-design accidents shows the high safety level of the power unit with the BN-1200 reactor plant, which meets the requirements for innovative designs of nuclear power plants.

REFERENCES

[1] SHEPELEV, S.F. “Development of the new generation power unit with the BN-1200 reactor” // Paper for this conference.

[2] MAROVA, E.V. **“**Evaluation results of BN-1200 compliance with the requirements of GENERATION IV” // Paper for this conference.

[3] VASILEV, B.A., OSIPOV, S.L., SHEPELEV, S.F. Optimal decisions. Combination of inherent self-protection properties and design measures to ensure safety of the BN-1200 reactor // REA Rosenergoatom. – No. 1. – 2013.

[4] RTISHEV, N.A., CHALYY, R.V., TARASOV, A.E., et al., “Development of SOCTAR-BN Code”, Proceedings of the “International Scientific and Technical Conference - 2012”, Moscow, ENTEC (2012) 348-359.

[5] RTISHEV, N.A., CHALYY, R.V., SEMENOV, V.N., et al., “Validation of SOCRAT-BN Code on the Base of Reactor Experiments”, 10th International Topical Meeting on Nuclear Thermal Hydraulics, Operation and Safety (NUTHOS10), Okinawa, Japan, 14-18 December 2014.

[6] OSIPOV, S.L., GORBUNOV, V.S., ANFIMOV, A.M., et al., “Application of integral calculation code SOCRAT-BN for safety justification of BN-1200 RP”, Innovative designs and technologies of nuclear power: Third International Scientific and Technical Conference, Moscow, 7-10 October 2014, Proceedings. – Moscow, JSC “NIKIET”, 2 (2014) 164-169.