Research Article

Analysis of a severe beyond design basis accident for the EGP-6 reactor of the Bilibino NPP. Radioactive source term determination[*]

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Abstract

Analysis was performed of dynamic phase of severe accident of the EGP-6 reactor of the Bilibino NPP, due to uncontrolled reactivity insertion initiated by withdrawal of two pare of automatic control rods with followed by full failure of reactor emergency protection system. This initial event leads to promt increasing of reactor core power up to 450% of nominal value with short period, coupled with rise of temperature of fuel, pressure and temperature of coolant. These factors lead to crisis of heat exchange with subsequent ruptures tubes of fuel assemblies and coolant blow down into graphite stack. All its lead to rise of pressure in reactor shell and damage of it, outflow of steam-water mixture through up-reactor area to ventilation system, communication corridors and reactor hall and further – to atmospheric release. Transient processes were calculated using code RELAP5/Mod3.2. It was considered stages of processes of fuel damage and evaluated dynamic of a number and degree of damaged fuel assembles. They were grouped on burn-up and for each group it was performed analysis of dynamic of damage values. Further it was considered processes of yield of fission products from damaged fuel with models, based on experimental data on yield of fission products from fuel material, used in assembles of Bilibino NPP fuel type (fuel tubes with steel cladding, where fuel material is grits of uranium dioxide in magnesium), under condition of severe accident, especially performed in SSC IPPE. Transport of fission products with steam and air up to release points was evaluated with models, based on experimental data of fission product transport through graphite stack under conditions of severe accident, also especially performed in SSC IPPE. Evaluation of source term was performed in accordance with accident dynamic and assumed modes of release for conservative and most possible approaches. It was noted good self-protection property of EGP-6 reactor under severe beyond design basis accident condition.

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Introduction

Bilibino NPP (BiNPP) consists of 4 reactor blocks of EGP-6 reactor, with total electrical power equal to 48 MW and up to 100 GCal/h of heat production. Draft design of BiNPP was prepared in 1964, detailed design - in 1965. Construction of the BiNPP was begun in October 1966. The NPP was constructed and is operated in extremely complex climatic and transportation conditions of Arctic region. Upgrading systems and equipments aimed at the enhancement of reliability and safety of operation was implemented during the period of operation of power units of the BiNPP (Bilibino NPP. Unit 4 2016). Structural feature of the NPP design is single central hall with windows for all power units.

Thermal hydraulic processes in the BiNPP building

Accident with the following initiating event (IE) is postulated: uncontrolled reactivity insertion initiated by withdrawal of two pare of automatic control rods with followed by full failure of reactor emergency protection system. Analysis of the worst (and the least probable) scenario of this beyond design basis accident (in accordance with NP-001-2015, RB-001-05 2005, Deterministic Safety Analysis for Nuclear Power Plants 2009) was published in (Bilibino NPP. Unit 4 2016). Power level prior to the accident was assumed to be equal to 100%. Reactor power dynamic is reflected in Fig.1. Estimation of reactor power dynamic was performed using the model of point kinetics of RELAP5/MOD3.2 code (Fletcher and Schultz, RELAP5/MOD3.2 2012).

Monotonous growth of reactor power occurs during 20 sec from the initial events and leads to reach maximum power level of 450% of rated reactor power (see Fig. 1). It results in the increase of temperature of fuel and rich
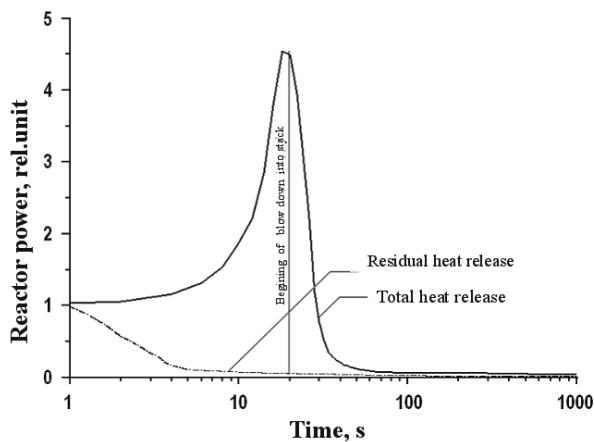
to heat exchange crisis in fuel assemblies. Heat exchange crisis causes to realization of conditions for damage of internal fuel cladding with bearing coolant pressure. When temperature of coolant achieves 900°C under design pressure in the main circuit (MC), damage of fuel cladding in 126 fuel assemblies arise on the 20-th sec, causing blow down of coolant in the reactor stack. Max flow rate of coolant blow down rises up to 1200 kg/sec on the 42-th sec. So total mass of coolant in MC is about 22 tons so long blow down with the rate is impossible.

Blow down of coolant in the reactor stack results in pressure rise in the reactor space (RS) and follows rupture of reactor shell on the 50-th sec with subsequent outflow of steam-water mixture in installation space, space above reactor and central hall (CH). Steam-water mixture outflow from the RS into the installation space is significant and comparable with flow rate of blowdown from the MC. Significant level of flow rates leads to fast reduction of mass of the substance inside the volumes from which the blow down is happening and after 70 sec from the beginning of the accident they end up to be fairly moderate which leads to the rate of outflow reduction of radioactive substances from the reactor core into the central hall of NPP building (Fig. 2).

Outflow of steam-water mixture from the installation space in the space above the reactor core takes place through the annular gap between the concrete overhead slab and the reactor top plate. Following this, outflow of steam-water mixture in the central hall (CH) takes place causing increase of pressure, disruption of normal air exchange ensured by the ventilation systems and destruction of CH windows.

Outflow of steam-water mixture in the CH from the space above the reactor (Fig. 3) proceeds relatively evenly in time ~ 100 sec. Total mass of outflow mixture is not more than 10% of the mass originally contained in the cooling circuit of the damaged reactor which could be explained with liquid phase for main part of coolant and separation of steam-water mixture outflow along the path to the central hall realized sequentially. There are other
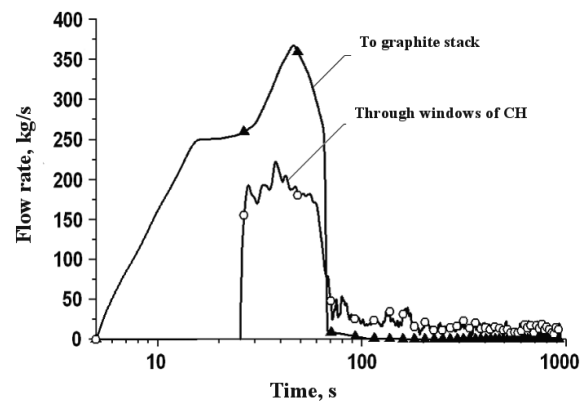

Figure 1. Dynamics of relative variation of reactor power compared with rated power

Figure 2. Flow rates of coolant blow down in the RS and through the reactor room windows
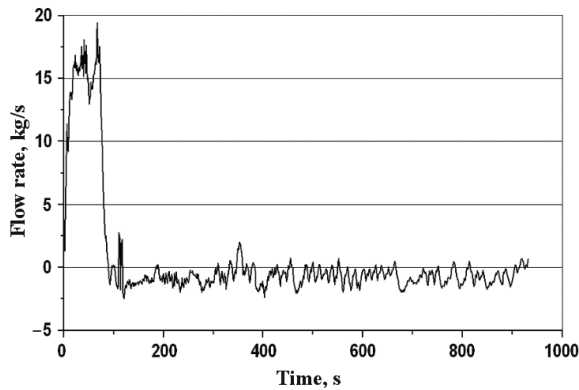
Figure 3. Flow rate of steam-water mixture from the space above the damaged reactor into the central hall

Figure 4. Flow rate of steam-water mixture through the windows of central hall

losses of this flow on main safety valve, system for localization of "wet" accident, drainage and ventilation.

Release of steam-water mixture in atmosphere corresponds to the dynamics of flow rates of outflow steam-water mixture to CH. Flow rate of steam-water mixture from windows of the central hall is presented in Figure 4.

Yield of radioactive products from damaged fuel

The data of (Samoylov et al. 1982, Samoylov et al. 1982a, Samoylov et al. 1996, Technical Reports Series No. 388, STI/DOC/010/388 1998, IAEA-TECDOC-1345 2003, IAEA-TECDOC-1454 2005, NF-T-2.1, STI/PUB/1445 2010), requirements (NP-082-07 2007) and results of experimental studies of behavior of reactor fuel of EGP-6 reactors implemented in 1990 – 1994 by the SSC RF-IPPE and published in (Baranaev et al. 1993, Baranaev et al. 1993a, Dolgov et al. 1995, 1996, Suvorov et al. 1996, 1998) were used in the investigation of processes of fuel destruction and release of radioactive fission products (RFP) from damaged fuel during the accident. They are represented in the form of three sequential stages:

Growth of power leads to growth of temperature and coolant pressure, with subsequent heat exchange crisis in some fuel assemblies. It results in the partial destruction of internal and external fuel cladding in those fuel assemblies where local temperature reaches ~ 900°C. At the momentum of fuel cladding destruction it were occurred instant yield of inert radioactive gases (IRG), iodine and cesium isotopes. Radioactive fission products (RFPs) are released with steam-water mixture, which serves as the carrying medium for subsequent arrival of RPFs to the reactor stack, space and later up to central hall and atmosphere.

After cessation of coolant outflow further partial destruction of fuel claddings can proceed only in those fuel assemblies (FA) where local temperature in the process of the accident could reach magnesium boiling point (~1100°C). Instant yield of IRG, iodine and cesium iso-
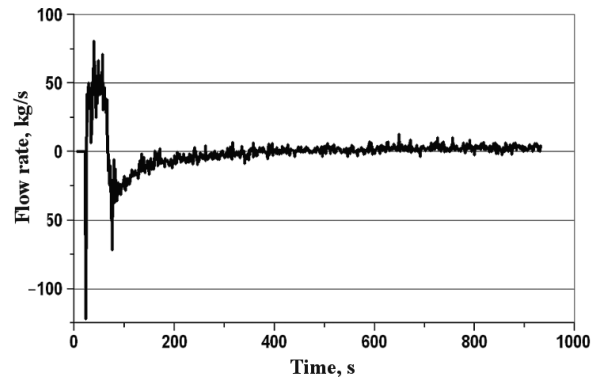
topes from contacting material of fuel matrix occurs at the moment of rupture of fuel cladding. Carrying medium for direct release of radioactive products from the reactor stack and space (and later) is not available during this phase.

Diffusive release of RFPs from heated fuel occurs in accordance with subsequent dynamic of temperature growth in those parts of destroyed fuel elements, where temperature increased above ~1100°C. Carrying medium for direct release of RFPs from reactor stack, space ans later is missing during this stage.

According to conservative estimation based on the results of experimental studies (Baranaev et al. 1993, Baranaev et al. 1993a, Dolgov et al. 1995, 1996, Suvorov et al. 1996, 1998) values of instant yield of volatile RFPs from the damaged parts of fuel during the first and second stages consists of 0.5% total activity per iodine, 0.07% total activity per cesium and 9% total activity per IRG.

Gradual increase of the yield of main RFPs from the damaged parts of fuel occurs in the process of heating uranium dioxide grit in magnesium matrix above 1100°C, which can be described based on (Baranaev et al. 1993, Baranaev et al. 1993a, Dolgov et al. 1995, 1996, Suvorov et al. 1996, 1998) in the form of Arrhenius equation as follows:

$$K = K_0 \exp(-Q/RT),$$

where K is the release rate, s^{-1}; R is the absolute gas constant; T is the fuel temperature, K.

Constants Q and K_0 presented in Table 1 were obtained by processing of experimental data.

Assessment of fuel damage during the accident

Dynamics of accident was described in (Bilibino NPP. Unit 4 2016). On the first stage of the accident within 30 secs from the initial events (displacement of automatic control rods) growth of neutron flux power to ~ 450% of the rated value and pressure in the MC up to ~10.0

Table 1. Constants for equation, which describes yields for radionuclides from fuel under hitting after cladding rupture

Nuclide	K_0, c^{-1}	Q, J/mole	Medium	Comment
I	0.30	187.7×10^3	Steam	1100 < T < 1350ºC
	5.09×10^{17}	766.9×10^3		T > 1350ºC
Cs	7.13×10^{-4}	111.8×10^3	Steam	1100 < T < 1350ºC
	1.60×10^{14}	649.5×10^3		T > 1350ºC
Kr, Xe	5.5×10^1	242.0×10^3	Steam	T > 1100ºC

MPa occur. Multiple ruptures of fuel accompanied with blow down of coolant into the graphite stack take place between the 30-th and 65-th secs. Complete drainage of the MC occurs by the 100-th second of accident. Further accident is accompanied with slow heating of FAs and graphite stack, followed with their subsequent cooling. Temperature dynamics for variation of max temperatures of FAs and graphite stack is presented in Figs. 5 and 6.

Dynamics for amount of damaged FAs is provided in (Bilibino NPP. Unit 4 2016) and was estimated on the basis of data of thermal engineering calculations for variations of temperatures of fuel elements and graphite (see Figs. 5 and 6). The following dynamics of FAs damages was obtained in accordance with results of calculations:

– Stage I. Partial damage of fuel cladding in 16 FAs with maximum power (eight FAs in the central part of the reactor core along the length equal to 1.45 m and eight FAs in the middle part along the length of 1.2 m) occurs by the 35-th second when temperature equal to 900° is reached.
– Stage II. Partial damage of fuel cladding in 110 FAs with medium power (along the length of 1 m when temperature equal to 900°C is reached), occurs by the 100-th second. Yield of RFPs from fuel are removed with coolant outflow through graphite stack into the reactor space.
– Stage III. Criterion of partial destruction of fuel rod cladding without coolant pressure, corresponded to tempera-

ture above 1100°C is reached after the 100-th sec only in FAs with maximum power, which takes place during the time interval between 100 s and 4000 s. Zone of fuel elements destruction increases within this interval as follows: for eight FAs with maximum power in the central part of reactor core length of damaged sections increases from 1.45 to 1.85 m while for eight FAs with maximum power of peripheral part of the core it increases from 1.2 to 1.55 m. This partial destruction is accompanied with additional yield of RFPs. Diffusive yield of RFPs occurs as well due to the heating of fuel grit above 1100°C. However, carrying medium for intensive release of RFPs from the reactor space is not present during this stage.

Estimated fraction of damaged fuel is present in Table 2 for different time moments. During estimation of fuel destruction it was conservatively accepted that temperature of all six fuel elements of FAs is equal and corresponds to the dependence for FAs with maximum and medium power presented in Figs. 5 and 6.

Damaged FAs activity inventory

Information on FA burnup is needed for estimation of source term of the accident. Distribution of FAs according to their power and burnup by the end of the time interval between fuel loading and time moment when accumulati-

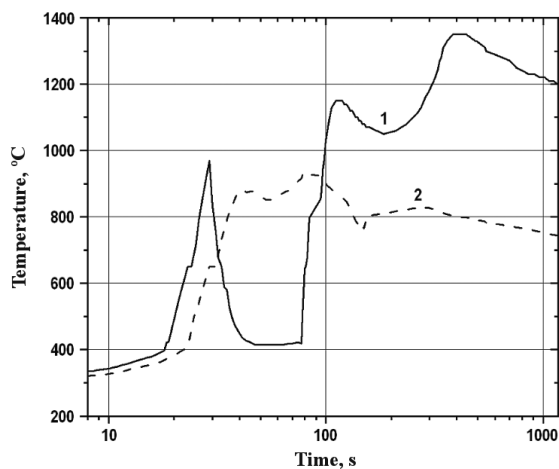**Figure 5.** Dynamics of variation of maximum temperature of FAs (ºC) during beyond design basis accident: 1 – temperature of fuel rod for FA with maximum power; 2 – temperature of fuel rod for FA with medium power
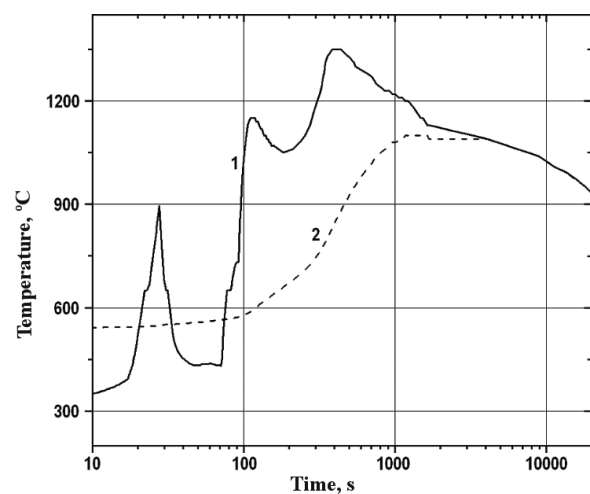

Figure 6. Time evolution of maximum temperatures of fuel rods (1) and graphite (2) for FAs with maximum power during beyond design basis accident

Table 2. Average damaged parts of FAs during different stages of the accident

Group of FAs	Stage I	Stage II	Stage III
16 FAs with maximum power	0.40	0.40	0.57
110 FAs with medium power	0.00	0.33	0.33

on of RFPs in fuel is maximum is presented in Figs 7 and 8. It was conservatively assumed that the accident takes place specifically at that time moment.

All damaged FAs were divided into three groups according to their power and burnup. Among 16 FAs with maximum power (those to be damaged the first) 12 FAs have average burnup ~ 60 MW×day/FA with power equal to 300 kW/FA and four FAs have average burnup ~ 157 MW×day/FA with power equal to 290 kW/FA. 110 FAs of medium power which are also destroyed during the initial phases of the accident correspond to the group with average level of burnup ~ 219 MW×day/FA with power equal to 227 kW/FA. Calculations were performed with code ORIGEN-2.1 (Croff 1983) taking into account calculated data from (Deen et al. 1995, Sharapov 1997) for quantitative determination of accumulation of fission products in the above three groups of FAs for different time moments after the initial event of the accident allowing correctly determining quantities of released short-lived nuclides.
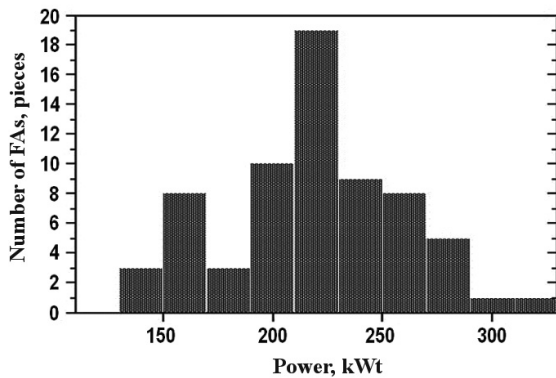

Figure 7. Histogram of FA distribution according to power in EGP-6 reactor installation
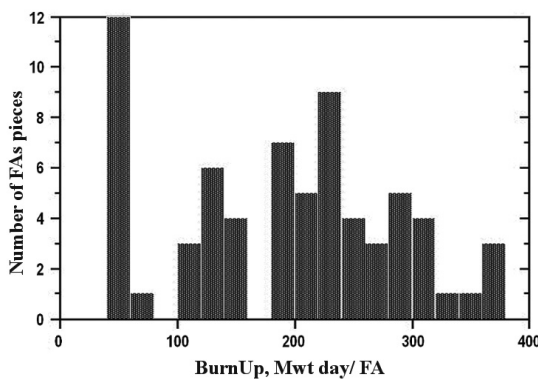

Figure 8. Histogram of FA distribution according to burnup in EGP-6 reactor installation

Fission products transport through graphite stack

Transport of RFPs through the graphite stack is accompanied with their partial sorption and secondary desorption. Coefficients of transport for RFP through the graphite stack are presented in Table 3 for the stage of coolant outflow.

Table 3. Coefficients of transport for RFP through graphite stack, %

Graphite stack, ºC	Radionuclides		
	IRG	I	Cs
400	0.2	0.05	0.02
500	1.7	0.59	0.25
600	2.3	0.9	0.4
700	3.7	1.9	1.2
800	5.0	2.8	1.9
900	7.7	5.8	5.2
1000	10.5	8.8	6.6
1100	16.0	13.2	14.0
1200	25.0	20.0	22.0

Coefficients of trapping RFPs with graphite stack (Table 4) were used for quantitative estimation of activities of different nuclides on the basis of the data of calculations for graphite stack temperature dynamic.

Table 4. Coefficients of RFP trapping with graphite stack during different time intervals, %

Time interval, s	0 – 250	250 – 4000
Cs	0.34	0.37
I	0.54	0.58
IRG	1.19	1.21

Severe beyond design basis accident radioactive source term

Release of nuclides into the reactor space (RS) was formed taking into account all the above presented data on the activity inventory for damaged fuel, yield and transfer of RFPs. Consist and activity for RFPs release into the RS and the dynamics of its time evolution are presented in Table 5.

The computational analysis of the accident dynamic demonstrates that approximately 7.3% of total mass of steam-water mixture, which amounts to ~1215 kg, which outflow from the RS during stages I and II, are immediately released into atmosphere through the destroyed window of the CH. Remaining mass of steam-water mixture (16285 kg or 92.7% of total mass) is dynamically redistributed after outflow from the RS between the volumes of rooms of the plant in concordance with technological communication lines and available leaks in the structures. Following this, transported activity can be released in atmosphere through ventilation systems and later in the ventilation stack.

Table 5. Consist and activity for RFPs release into the RS, Bq

Nuclide	Release into the RS from reactor stack for different stages of the accident, Bq		
	Stage I (loss of sealing)	Stage II (loss of sealing)	Stage III (heating)
Kr-85	2.2E11	4.1E09	8.5E12
Kr-85M	5.4E12	3.7E11	1.7E13
Kr-87	1.1E13	7.2E11	2.4E13
Kr-88	1.5E13	1.0E12	1.3E13
Kr-89	1.4E13	5.0E11	4.6E03
I-131	3.2E11	2.3E10	2.2E11
I-132	4.8E11	3.4E10	1.2E12
I-133	7.4E11	5.3E10	4.6E13
I-134	8.3E11	5.9E10	2.7E9
I-135	6.9E11	4.9E10	2.0E13
Xe-131M	1.4E11	9.5E9	7.8E11
Xe-133	2.9E13	2.0E12	1.3E12
Xe-133M	8.7E11	5.8E10	1.4E12
Xe-135	1.4E13	8.6E11	7.6E12
Xe-135M	5.0E12	3.3E11	2.4E9
Xe-137	2.0E13	8.3E11	9.7E9
Cs-134	2.7E9	2.3E7	1.1E12
Cs-136	8.3E8	2.0E7	2.1E13
Cs-137	4.3E9	8.1E7	5.5E11
Cs-138	6.5E10	4.6E9	1.1E12

Radiation consequences of the first two stages of the analyzed accident are determined by two releases with different physical parameters:

– Through windows of the CH in the aerodynamic shadow of the building with low height of the release (14 m), low velocity (~ 0.1 m/s) and with elevated average temperature ~55ºC of released matter;
– Through ventilation stack along the systems of ventilation.

Significant part of steam is condensed and precipitates in the process of transport of steam-water mixture on the surfaces together with non-gaseous radionuclides contained in them. Coefficient of activity transport through the ventilation system was introduced for accounting for this fact and, as well, for the fact that ventilation system is equipped with filters. The value of this coefficient was

taken to be equal about 0.1 for cesium and iodine nuclides and to 1 for IRG as the conservative estimation.

Estimated results of radioactive source term for the first two stages of the accident through the windows of the CH and through the ventilation stack, obtained in the assumption that activity of RFPs is uniformly spread within the coolant, are presented in Table 6.

Steam-water mixture as a carrying medium for direct transport of activity from the RS in the rooms of the plant building missed on the III stage of the accident. Nevertheless the nuclides could be transported from RS to CH with convective air flows. After that they can be released into environment with one of the following two paths – through the destroyed windows of the CH and through the ventilation stack after transport through ventilation systems. The following two ultimate cases of formation for source term were examined to analyze radiation consequences for the III stage of the accident:

Table 6. Estimated results of radioactive source term for stages I and II of the beyond design basis accident, Bq

Nuclide	Release through CH windows	Release through the stack	Nuclide	Release through CH windows	Release through the stack
Kr-85	1.6E10	2.0E12	Xe-131m	1.1E10	1.3E11
Kr-85m	3.9E11	5.0E12	Xe-133	2.2E12	2.7E13
Kr-87	7.8E11	1.0E13	Xe-133M	6.3E10	8.1E11
Kr-88	1.1E12	1.4E13	Xe-135	9.9E11	1.3E13
Kr-89	1.0E12	1.3E13	Xe-135m	3.6E11	4.6E12
I-131	2.4E10	3.0E10	Xe-137	1.5E12	1.9E13
I-132	3.6E10	4.4E10	Cs-134	4.0E8	2.3E8
I-133	5.4E10	6.9E10	Cs-136	6.0E7	7.7E7
I-134	6.0E10	7.7E10	Cs-137	3.3E8	4.0E8
I-135	5.1E10	6.4E10	Cs-138	4.8E9	6.0E9

Table 7. Estimated radioactive source term on the III stage of the accident, Bq

Nuclide	Release through CH windows	Release in the stack	Nuclide	Release through CH windows	Release in the stack
Kr-85	8.5E12	8.5E12	Xe-131m	7.8E11	7.8E11
Kr-85m	1.7E13	1.7E13	Xe-133	1.3E12	1.3E12
Kr-87	2.4E13	2.4E13	Xe-133m	1.4E12	1.4E12
Kr-88	1.3E13	1.3E13	Xe-135	7.6E12	7.6E12
Kr-89	4.6E3	4.6E3	Xe-135m	2.4E9	2.4E9
I-131	2.2E11	2.2E10	Xe-137	9.7E9	9.7E9
I-132	1.2E12	1.2E11	Cs-134	1.1E12	1.1E11
I-133	4.6E13	4.6E12	Cs-136	2.1E13	2.1E12
I-134	2.7E9	2.7E8	Cs-137	5.5E11	5.5E10
I-135	2.0E13	2.0E12	Cs-138	1.1E12	1.1E11

1. Release of all activity, which outflow from RS on the III stage of the accident (Table 7), through windows of the CH. This assumption provides max conservative estimation for consequences of the radiation accident and is realized in the case of loss of power supply for the ventilation system of space above the reactor. However, release through the windows is not large even during the dynamic phase.
2. Release of all activity from the RS through ventilation ducts in the ventilation stack. Total coefficient of activity transport through the ventilation system is determined only with filters efficiency, and for stage III this coefficient of activity transport for Cs and I nuclides was accepted to be equal to 0.1, and for IRG was taken to be equal to 1.

Release in the ventilation stack for the stage III is the most probable, because flow rate for the ventilation system from the space above the reactor is ~ 12000 m³/h.

Estimated radioactive source term on the III stage of the accident are presented in Table 7.

Thus, to perform analysis of consequences of sever beyond design basis accident for public it is necessary takes into account the dynamics and scale of fuel destruction, specific features of transport for radionuclides and paths of release into the environment for different phases of the accident.

Conclusion

It war performed computational analysis for phenomenology of severe beyond design basis accident due to uncontrolled positive reactivity insertion initiated by withdrawal of two pare of automatic control rods with followed by full failure of reactor emergency protection system.

An assessment of the dynamics of fuel assemblies damage and the impact of possible ways of transporting radionuclides through the NPP premises on the formation of an accidental release was made.

Radioactive source term of the accident into the environment have been estimated on the basis of a conservative approach and for the most probable scenario of the accident.

It was demonstrated the features of the self-defense of EGP-6 reactor (trapping of RFPs with reactor stack, the operation of the ventilation system of space above the reactor) allow significantly reduce the release of activity into the environment, even in the case of a postulated sever beyond design basis accident with fuel element damage in 126 fuel assemblies.

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