PROBABILISTIC SAFETY ANALYSIS OF NPP WITH THE BREST-OD-300 REACTOR

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Abstract. BREST-OD-300, an innovative inherent safety fast reactor, is being developed as a pilot and demonstration prototype for the basic commercial reactor facilities of future nuclear power with a closed nuclear fuel cycle [1].
As part of the PSA level 1 with the BREST-OD-300 reactor the following main tasks have been solved:

– selection and grouping of initial events;
– identification of success criteria (SC) for the modeled functions;
– analysis of emergency sequences;
– analysis of systems;
– data analysis;
– analysis of the PSA results.

The obtained CDF value ($9.0 \cdot 10^{-9}$ 1/year) satisfies to the target values in NP-001-15 [2] both for the total probability of severe accidents (10^{-5}) in an interval of 1 year and the total probability of a major emergency release (10^{-7}) in an interval of 1 year.

Key Words: fast reactor, nuclear safety, probability safety analysis.

1. Introduction

The objectives of a probabilistic safety analysis (PSA) for a nuclear power plant (NPP) with BREST-OD-300 reactor [1] are to:

– assess the probabilistic safety indicators for the BREST-OD-300 NPP (risk indices);

– identify the major risk contributors, carry out the importance, sensitivity and uncertainty analyses to enable identification of the dominant risk factors; and develop measures for reducing the risk, and improving the safety and reliability of the power unit and its systems.

Due to the fact that protection against all types of accidents in the BREST-OD-300 reactor design is based predominantly on passive principles and safety systems, one of the PSA goals was to assess the efficiency of these means and demonstrate them to be advantageous over active safety systems. A much smaller magnitude than specified in NP-001-15 [2] was proposed to be used as the probabilistic criteria (target) for the total probability of severe accidents for each NPP unit in an interval of one year and equal to 10^{-5}. Given the BREST-OD-300 design features, the practicable risk target is the total probability of a major emergency release equal to 10^{-7} 1/year which is also specified in NP-001-15.

The analysis objects were modeled and the model's characteristics were calculated using the RiskSpectrum code.

2. Contributions of initiating events and results of uncertainty analysis

As part of the BREST-OD-300 NPP PSA model analysis, the probability of severe accidents (core damage risk hereinafter) was determined for each initiating event (IE), and core damage was defined as the value of the total risk of Category A [3-4].

Reliability indicators were calculated for the IE components and frequencies based on generalized statistical data for NPPs and research reactors [5].

The parametric uncertainty of the core damage probability for Category A was analyzed using a statistical test method (Monte Carlo method) implemented in the RiskSpectrum code.

Table 1 presents the results of the core damage frequency assessment and the uncertainty analysis for each IE and the total risk magnitude.

Table 1: Summary of the core damage frequency assessment and uncertainty analysis results.

IE group code	**Description of IE group**	**IE group frequency, 1/year**	**Pointwise estimate**	**5 %**	**50 %**	**95 %**
1 Unauthorized change of reactivity – UCR						
UCR-CR	1.1 Unauthorized control rod (CR) group for automatic control or shim rod (SR) withdrawal from the core during reactor operation at different power levels (30÷100% of the rated power and the minimum controlled reactor power, MCRP) as the result of a CPS equipment failure or an operator error	7.4E-8	8.0E-16	2.0E-17	2.4E-16	3.0E-15
UCR-MCP	1.2 Erroneous startup of MCP to the rated flow rate at a low power level	8.2E-4	8.8E-12	1.2E-12	5.7E-12	2.6E-11
UCR-RF	1.3 Unauthorized CPS CR floating	3.7E-2	4.0E-10	5.4E-11	2.6E-10	1.2E-09
UCR-CRM	1.4 Postulated insertion of the total positive reactivity margin at a maximum design rate (withdrawal of as many CPS CR from the core as possible)	7.1E-8	7.6E-16	1.0E-16	5.0E-16	2.3E-15
2 Violation of heat sink from the core – VHS						
VHS-BLFA	2.1 Blocking of the coolant flow rate at the FA inlet	5.0E-4	3.1E-11	9.1E-13	1.0E-11	1.2E-10
VHS-MCP	2.2 Disconnection or malfunction of main circulation pumps (MCP)	1.0E-2	1.1E-10	2.5E-12	3.2E-11	4.2E-10

IE group code	Description of IE group	IE group frequency, 1/year	Pointwise estimate	5 %	50 %	95 %
VHS-SGF	2.3 Reduction in the lead flow area due to lead freezing on the SG tubes	1.0E-3	1.1E-11	2.4E-13	3.1E-12	4.0E-11
VHS-CF	2.4 Floating of the core or reflector components during power operation	1.0E-3	1.1E-11	2.6E-13	3.3E-12	4.1E-11
VHS-LTSS	2.5 Leakage through the cold and hot lead separating shell	6.7E-8	7.2E-16	1.7E-17	2.2E-16	2.7E-15
VHS-LOOP	2.6 Postulated loss of the unit system and emergency power supply	1.0E-1	1.1E-09	2.6E-11	3.2E-10	4.0E-09
3 Pressure variation in the gas chamber of fast reactor – PVGC						
PVGC-SGL	3.1 Loss of integrity by the SG tubes	1.2E-2	1.5E-09	4.8E-11	5.3E-10	5.3E-09
4 Deterioration of heat removal to the secondary circuit – DHSC						
DHSC-FWP	4.1 Malfunction of the secondary circuit condensate and feedwater pumps	1.0E-2	1.1E-10	2.6E-12	3.3E-11	4.2E-10
DHSC-FVFC	4.2 False actuation of the feedwater isolation valves, closure of the control valves at SG inlet	2.4E-4	2.6E-12	4.6E-13	1.9E-12	7.4E-12
DHSC-FWPR	4.3 Feedwater piping rupture	5.0E-3	5.4E-11	1.3E-12	1.6E-11	2.1E-10
PCSC-TC	4.4 Turbine condenser failure	3.0E-1	3.2E-09	7.6E-11	9.5E-10	1.2E-08
5 Excess heat removal to the secondary circuit – EHSC (potential lead coolant freezing in the SG)						
EHSC-SLSV	5.1 Loss of the steam line integrity upstream of the isolation steam valve (ISV) of one SG module	1.0E-3	1.1E-11	2.6E-13	3.4E-12	4.0E-11
EHSC-SLSH	5.2 Loss of integrity by the HP steam header downstream of the ISV	1.0E-3	1.1E-11	2.6E-13	3.3E-12	4.2E-11
EHSC-FAEP	5.3 False actuation of emergency protection during rated power operation	1.0E-2	1.1E-10	2.6E-12	3.3E-11	4.1E-10
EHSC-FWSS	5.4 Closure of the FWSS heating steam supply line valve	8.7E-5	9.5E-13	1.3E-13	6.1E-13	2.9E-12
6 IEs leading directly to core damage						
CF	6.1 Loss of integrity by the reactor vessel liner with a lead coolant leakage	9.8E-10	9.8E-10	3.5E-11	3.6E-10	3.6E-09

IE group code	**Description of IE group**	**IE group frequency, 1/year**	**Pointwise estimate**	**5 %**	**50 %**	**95 %**
MSGTR	6.2 Multiple SG tube rupture	1.0E-9	1.0E-09	3.7E-11	3.8E-10	3.9E-09
7 IEs caused by indoor fires						
F_ECCS	7.1 ECCS room fire	9.5E-5	5.9E-12	1.0E-12	4.1E-12	1.7E-11
F_ICPS	7.2 ICPS room fire	2.6E-3	2.8E-11	3.8E-12	1.8E-11	8.4E-11
F_FWP	7.3 Fire in the feedwater pumps and support systems room	1.3E-2	1.4E-10	2.0E-11	9.1E-11	4.2E-10
F_MSV	7.4 Fire in the MSV control equipment room	4.2E-3	4.6E-11	6.5E-12	3.0E-11	1.4E-10
F_TG	7.5 TG hall fire	1.3E-2	1.4E-10	1.9E-11	9.0E-11	4.3E-10
	Total:	5.0E-1	9.0E-9	1.9E-09	6.0E-09	2.5E-08

Besides, results have been obtained (Fig. 1), which characterize the level of the BREST-OD-300 unit immunity to the occurrence of an IE at the power unit. The immunity level is represented by the risk barrier showing the extent to which the core damage frequency decreases as compared to the IE frequency thanks to the operation of the unit safety systems. Graphically, the risk from each IE is presented by a two-component column in which the first component shows the IE frequency and the second one shows the probability of the safety systems failure to perform their functions. Such form of representation of results makes it possible to visualize the level of the contribution to the risk both from the IE and from the safety systems.

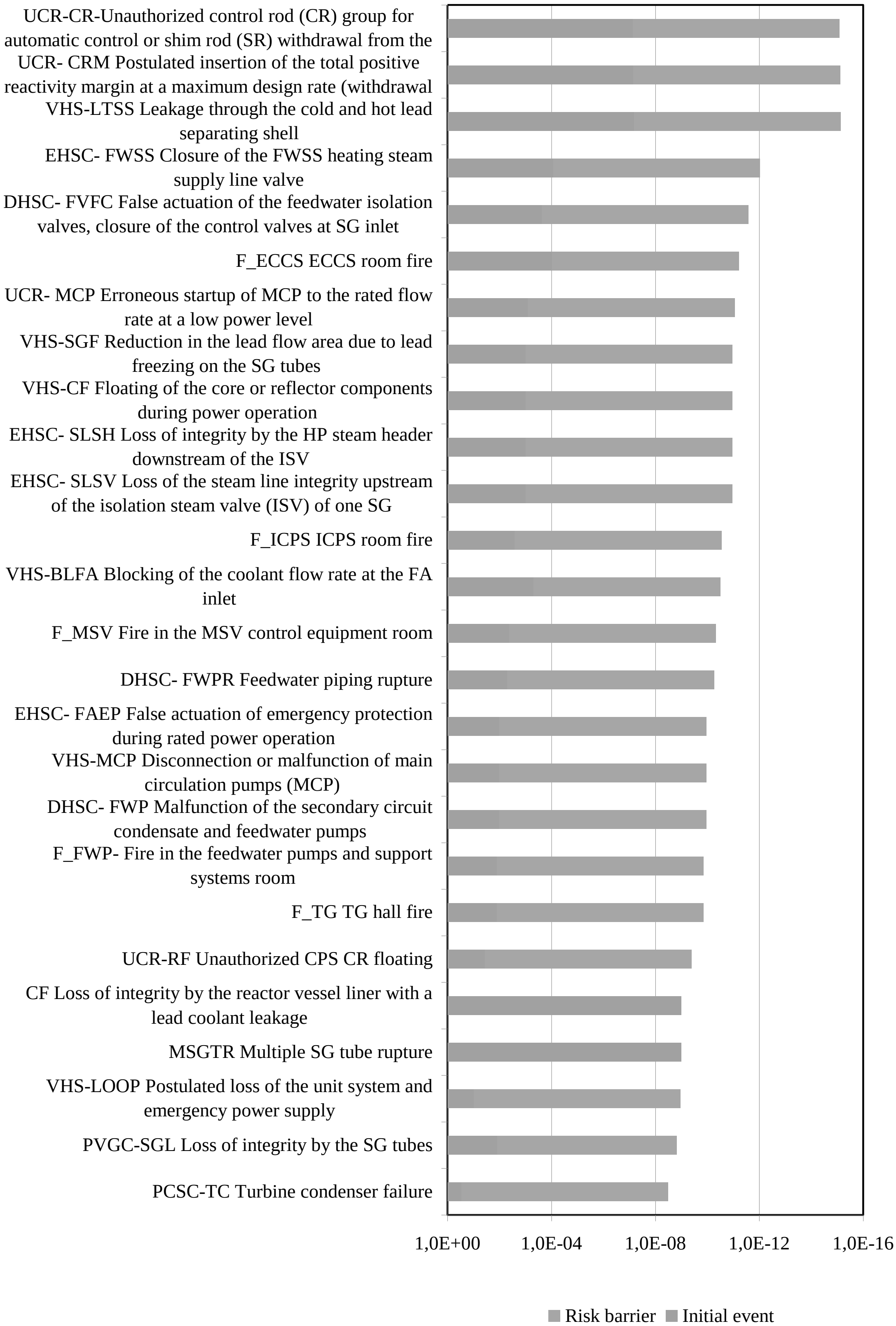

FIG. 1. Risk topography for the BREST-OD-300 NPP UNIT.

The obtained integral risk value for Category A ($9.0 \cdot 10^{-9}$ 1/year) satisfies to the target values in NP-001-15 both with respect to the total probability of severe accidents (10^{-5}) in an interval of 1 year and the total probability of a major emergency release (10^{-7}) in an interval of 1 year.

3. Contributions of minimal cut sets, importance and sensitivity analysis results

Minimal cut sets (MCS) is a unique combination of events leading to the core damage. These events may include failures of components, operator errors or special base events with designated conditional probabilities. The dominant MCS are ranked in the order of decrease in their contribution to the total reactor core damage frequency.

The most important MCS obtained in the analysis of the BREST-OD-300 NPP unit PSA model for the core damage risk assessment are listed in Table 2.

Table 2: MCS for the core damage magnitude of Category A with a contribution of over 1%.

MCS number	Frequency, 1/year	Contri-bution, %	MCS	Description of events
1	3.2E-09	35.7	IE_PCSC-TC SAOR/SNR-AT-ALL	Turbine disconnection (with regard for the condenser failure) CCF – Field tube break (failure of all 4 channels)
2	1.1E-09	11.9	IE_VHS-LOOP SAOR/SNR-AT-ALL	Loss of auxiliary power CCF – Field tube break (failure of all 4 channels)
3	1.0E-09	11.1	IE_MSGTR	Multiple rupture of the SG tubes (over 8)
4	9.8E-10	10.8	IE_CF	Loss of integrity by the reactor vessel liner with a lead coolant leakage
5	4. 0E-10	4.4	IE_UCR-RF SAOR/SNR-AT-ALL	Spontaneous floating of CPS CR CCF – Field tube break (failure of all 4 channels)
6	1.4E-10	1.6	IE_F_TG SAOR/SNR-AT-ALL	TG hall fire CCF – Field tube break (failure of all 4 channels)
7	1.4E-10	1.5	IE_F_FWP SAOR/SNR-AT-ALL	Fire in the feedwater pumps and support systems room CCF – Field tube break (failure of all 4 channels)
8	1.3E-10	1.4	IE_PVGC- SGL SAOR/SNR-AT-ALL	Loss of integrity by the SG tubes CCF – Field tube break (failure of all 4 channels)
9-11	1.1E-10	1.2	IE_VHS-MCP; IE_EHSC-FAEP; IE_DHSC-FWP; SAOR/SNR-AT-ALL	Disconnection of 4 MCPs False actuation of emergency protection during rated power operation Disconnection of two PN-2 pumps (with no standby pump startup)CCF – Field tube break (failure of all 4 channels)

A number of dominant emergency sequences leading ultimately to the core damage were identified as the result of computational studies. The major contributors to the total risk magnitude are:

– failure of the NCS/ECCS Field tubes;

– multiple rupture of the SG tubes;

– reactor vessel leak.

Recommendations were worked out based on an analysis of the dominant risk contributors for increasing the BREST-OD-300 unit safety:

– for the SLLS – increased requirements to personnel in establishing the water level in the pressure suppression pool;

– for the ECCS – continuous monitoring as well as status inspection and maintenance of the Field tubes;

– detailed study of the IEs leading directly to the core damage.

The safety improvement measures proposed as part of the analysis make it possible to eliminate the said dominant risk factors and to improve the safety accordingly.

Importance and sensitivity were analyzed with respect to the following factors affecting the core damage risk assessment (Table 3):

– initiating events;

– common cause failures;

– personnel actions;

– basic events.

Contributions of events to the core damage frequency are presented in Fig. 2.

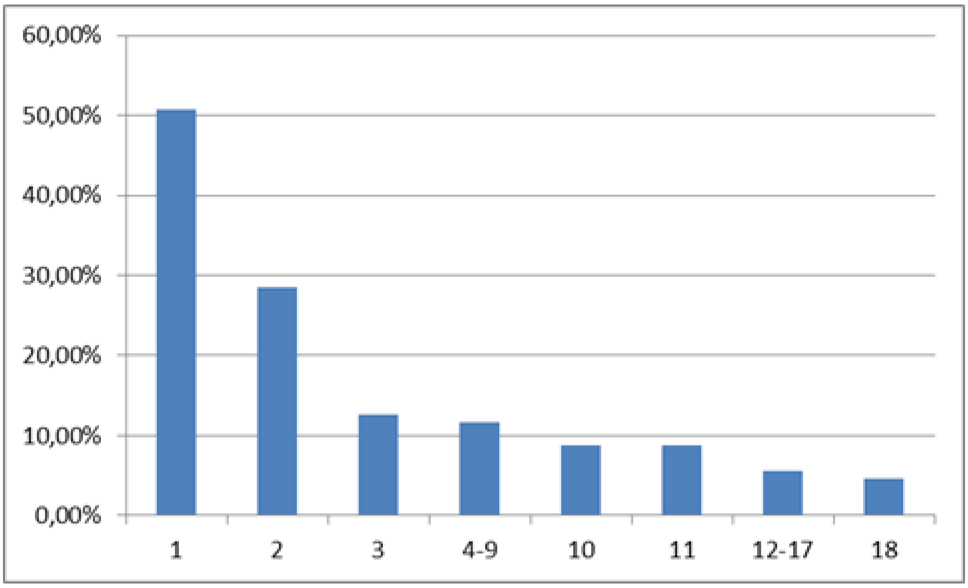

Fig. 2. Contributions of the BREST-OD-300 unit model events to the core damage frequency (the events numbers are in accordance with Table 3).

Table 3: Importance and sensitivity analysis results for all events considered in the BREST-OD-300 unit probabilistic model.

No.	**Event code**	**Event type**	**Event description**	**NOM**	**FV**	**FC**	**RDF**	**SENS**	**Contri-bution, %**
1	IE_F_FWP	SLI	Feedwater pump room fire	1.3E-02	5.1E-01	5.1E-01	2.0E+00	1.0E+01	50.8
2	IE_F_TG	SLI	TG hall fire	1.3E-02	2.9E-01	2.9E-01	1.4E+00	4.8E+00	28.5
3	IE_F_MSV	SLI	MSV control equipment room fire	4.2E-03	1.3E-01	1.3E-01	1.1E+00	2.4E+00	12.6
4-9	TSOU-AZ__C-13	CCF	Detected failure of the PLC (in 2 safety channels)	1.1E-06	1.2E-01	1.1E-01	1.1E+00	2.2E+00	11.7
10	2TSOU-AZ__C-ALL	CCF	Detected failure of the PLC (in 3 safety channels)	8.2E-07	8.8E-02	8.0E-02	1.1E+00	1.9E+00	8.8
11	1TSOU-AZ__C-ALL	CCF	Detected failures of the PLC (in 3 safety channels)	8.2E-07	8.8E-02	8.0E-02	1.1E+00	1.9E+00	8.8
12-17	TSOU-AZT__C-ALL	CCF	Detected failure of the PLC (in 3 safety channels)	6.2E-07	5.5E-02	5.5E-02	1.1E+00	1.6E+00	5.5
18	SAOR/SNR-AT-ALL	CCF	ECCS Field tube break	1.1E-08	4.7E-02	4.6E-02	1.1E+00	1.5E+00	4.6
Event type: IE – initiating event; BE – base event; CCF – common cause failure; PA – personnel action; SLI – special local impact. Column designations: NOM – nominal frequency value for an IE or the probability for other events; FV – Fussel-Vesely importance; FC – importance according to the fractional components method; RDF – risk decrease factor; RIF – risk increase factor; SENS – sensitivity factor.									

So, Table 3 contains a set of the unit risk dominant events, a deterministic and a probabilistic analysis of which will make it possible to develop measures to reduce the total risk of the core damage from internal initiating events.

4. Conclusions

The following major PSA goals were achieved as part of the study on the PSA of level 1:

– selection and grouping of initial events – a list of accident IEs was formed based on the analysis results (including SLIs);

– identification of success criteria (SC) for the modeled functions of systems – SCs were identified for the modeled functions of systems as part of this goal based on deterministic calculations for each IE selected for further analysis;

– analysis of emergency sequences – models of emergency sequences were developed and trees of events were built for each of the IE groups. The developed trees of events for all IE groups were included in the logical-probabilistic unit model to assess the implementation probabilities for end states;

– analysis of systems – logical-probabilistic models (failure trees) were developed as part of this goal for the following unit systems:

a) emergency core cooling system (ECCS);
b) steam generator leak localization system (SLLS);
c) integrated control and protection system (ICPS);
d) process safety control system (PSCS);
e) normal cooldown system (NCS);
f) reactor unit gas system (RUGS);
g) reactor vessel cooling system (RVCS);
h) reactor vessel heat-up system (RVHS);
i) coolant quality control and maintenance system (CQCMS);
j) power supply system;
k) SG protection system;

– data analysis – frequencies were assessed for the IE groups based on generalized statistical processing of information from different NPPs and research reactors;

– analysis of the PSA results – the obtained value of the Category A risk ($9.0\cdot10^{-9}$ 1/year) satisfies to the target values in NP-001-15 [2] both for the total probability of severe accidents (10^{-5}) in an interval of 1 year and the total probability of a major emergency release (10^{-7}) in an interval of 1 year.

The following conclusions have been made as the PSA result:

– the greatest contributors to the core damage magnitude are IEs caused by a turbine disconnection (with regard for the condenser failure) and a loss of auxiliary power involving, additionally, an ECCS failure. The implementation frequency of such scenarios is $\sim10^{-9}$ 1/year, which is due to the highly reliable ECCS with natural air circulation. Additionally, high heat capacity of the circuit makes it possible for the personnel to keep the ECCS serviceable by opening the flow control valves manually, while there is also a possibility to start the normal cooldown system;

– high reliability of the IPCS ($\sim1\cdot10^{-8}$ 1/demand) is achieved through the physical separation of sets which prevents their cross-effects and excludes the potentiality of a CCF. When the ICPS fails, negative power and temperature feedbacks and the passive feedback system (PFS) operation lead to the core not being damaged in a number of accidents;

– use of a once-through steam generator with a coolant-submerged bundle of heat-exchange tubes allows minimizing the probability of an intense steam pressure pulse in the primary circuit in the event of a wall break between the primary and the secondary circuits, which, in turn, simplifies the leak localization system. With a multiple break of the SG tubes (more than 8), the measures to reduce the break effects include the presence of a large compensation gas space in the BREST vessel for reducing the vessel pressure pulse, and, in case the SLLS is overfilled, use of rupture disks in the system for discharge of the steam that has passed through the water to the selected unit rooms. The core damage frequency in this case is $1\cdot10^{-9}$ 1/year;

– use of an integral layout for the primary circuit and the fast reactor's metal concrete vessel makes it possible to improve considerably the vessel reliability, since there are several barriers to separate the radioactive coolant from the environment, including steel cavity shells together with penetrations forming the primary circuit, heat-resistant concrete, an intermediate steel barrel, heat-insulating concrete, and an envelope with heat-resistant high-strength concrete. The frequency of such IE as loss of integrity by the reactor vessel liner with lead coolant leakage is $9.8 \cdot 10^{-10}$ 1/year;

– use of passive systems that ensure functional redundancy provides for a high level of the BREST-OD-300 unit safety with a core damage frequency of $9.0 \cdot 10^{-9}$ 1/year, a value comparable to similar values for other NPP projects (Table 4) [6].

Table 4: Core damage frequency values for different NPP projects.

Core frequency damage, 1/year						
VVER-TOI	**AP-1000**	**Kudankulam**	**Leningrad-2**	**Novovoronezh-2**	**BREST (reactor PSA)**	**BREST (NPP PSA)**
$1.3 \cdot 10^{-7}$	$2.4 \cdot 10^{-7}$	$2.7 \cdot 10^{-7}$	$2.2 \cdot 10^{-7}$	$1.7 \cdot 10^{-7}$	$8.7 \cdot 10^{-9}$	$9.0 \cdot 10^{-9}$

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