# Probabilistic Safety Analysis Results for BN Reactor Power Units

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**Abstract**. Probabilistic safety analysis is a constituent part of range of works aimed at BN power units safety assessment. At present time JSC “Afrikantov OKBM” has several PSA studies developed and it is continuing to improve all PSA models. Also, complex of activities on Probabilistic safety analysis Level 2 is being performed.

**Key Words**: Probabilistic safety analysis, initiating event, severe accident probability.

## 1. Introduction

Probabilistic safety analysis (PSA) is a constituent part of range of works aimed at BN power units safety assessment during operation (BN-600 and BN-800 reactors power units), lifetime extension (BN-600 reactor power unit), designing (BN-1200 reactor power unit). PSA reports are part of the document sets required to obtain appropriate Russian regulator (Rostechnadzor) licenses.

## 2. Software and methods used

All of PSA studies were performed using probabilistic safety analysis software system CRISS. This is the domestic software have been put into practice of probabilistic safety analysis for nuclear plants. CRISS software system is developing and improving by JSC “Afrikantov OKBM” during more than 20 years. It is the “client-server” architecture software allowing to resolve all tasks of full-scale PSA. CRISS software system has been licensed by Rostechnadzor.

CRISS software system allows to:

- accumulate in database information about safety systems, initiating events, human errors, components reliability data including common cause failures (CCF) models parameters, operability tests scheduling for safety systems components;
- create and edit fault trees;
- create and edit event trees;
- perform fault trees and event trees qualitative and quantitative analysis with automated CCF modelling;
- perform importance analysis, sensitivity analysis and uncertainty analysis;
- edit minimal cutsets;

- print and save fault trees and event trees graphic images, qualitative and quantitative analysis results, importance analysis, sensitivity analysis and uncertainty analysis results in Microsoft Office formats;
- implement HRA procedures using SHARP, THERP and HCR methodics;
- import databases and logic models (fault trees and event trees) from other PSA software.

For each PSA study appropriate probabilistic model of reactor power unit has been developed. To develop PSA probabilistic model event trees and fault trees methodology has been used.

Consequences of accident sequences in the general case are characterized by two states: an unsafe state (CD) and a state when a specified criteria are not violated owing to initiation of appropriate technical features provided for safety assurance – safe state (OK).

As a specified criteria of severe accident (severe reactor core damage) were considered excess of fuel elements damage limits for design basis accident and excess of reactor vessel temperature over allowable value.

Modelling of accident sequences was carried out within 24-hour interval after beginning of initiating event. Recovery of failed equipment within the modelling interval was not take into account. For separate passive sodium equipment (with long recovery period) modelling interval has been extended to 10 days.

Types of failures taken into account in system modelling:

- latent failures of system elements within interval between tests of their functionality;
- failures on demand during a system actuation;
- failures in operating modes;
- unavailability of elements due to their outage for repair.

In assessment of the safety systems reliability and determination of a probability of the accident sequences all types of potentially possible dependencies between elements, system trains and systems are analyzed, as follows:

- dependence on effect of an initiating event;
- structural-functional dependence conditioned by a presence of common structural elements or auxiliary systems;
- dependence conditioned by changes of systems equipment operating conditions due to accident sequence parameters changes;
- dependence conditioned by similarity of equipment design (common cause failure).

To treat CCF the binomial failure rate (BFR) model has been used.

For preliminary conservative estimates (screening analysis) are used recommendations from the SHARP (Systematic Human Actions Reliability Procedure) guide on analysis of personnel reliability. For comprehensive quantification of the most important erroneous actions by personnel are used THERP (Technique for Human Error Rate Prediction) guide and HCR (Human Cognitive Reliability) guide. Dependence analysis between different personal erroneous actions in each minimal cutest containing more than one erroneous actions has been performed.

Importance analysis (by Fussell-Vesely and Risk Achievement Worth coefficients) have been performed for different probabilistic models elements (for example initial events groups, accident sequences, minimal cut sets, systems elements, personnel errors). Uncertainty analysis has been done by the use of Monte-Carlo method.

## 3. PSA Development

General PSA goals are the following ones:
– power unit safety level assessment;
– recommendations development for power unit safety measures improvement.

First of all, PSA Level 1 (PSA-1) for internal initiating events for reactor full power operation mode was developed. All following PSA-1 studies are based on probabilistic model prepared within that PSA-1.

As initiating events for PSA were considered potentially dangerous events which could lead to core damage under certain conditions (additional failures of elements, systems, personnel errors).

The following types of initiating events were considered in PSA-1 for BN reactors:
- initiating events associated with failures of unit systems (elements) or personnel errors (internal initiating events);
- initiating events, caused by internal hazards (fires and floods);
- initiating events, caused by natural and man-induced external hazards.

Within each of the PSA-1 studies, system reliability analysis was performed, accident sequences were developed, human reliability analysis was implemented, a database on initiating event frequencies and system component reliability indices was developed, an integral probabilistic model of nuclear power unit was formed, and quantitative analysis was performed including importance, sensitivity and uncertainty analysis.

Obtained PSA results have been issued in the form of PSA reports and delivered to customers for the subsequent Rostechnadzor representation. The part of reports has successfully passed Rostechnadzor expertise, the others are in process preparation of transfer.

The PSA database on initiating event frequencies and component reliability indices is developed and updated based on the analysis of BN-600 power unit operating experience. For BN-800 and BN-1200 reactors power units PSA data analysis takes into account design distinctions between power units.

Main results of all performed PSAs (probabilities of severe accidents per one year) for all power units with BN reactors are given in Table I. Importance of different types of accidents for the specified units is given in Table II.

TABLE I: PROBABILITIES OF SEVERE ACCIDENTS PER ONE YEAR FOR POWER UNITS WITH BN REACTORS.

| **Power unit / Reactor** | **Internal initiating events for full power operation mode** | **Internal initiating events for low power and shutdown reactor modes** | **Internal hazards (fires, floods)** | **External hazards** | **Total for all initiating event types** |
|---|---|---|---|---|---|
| Beloyarsk power Unit 3 / BN-600 | 2.0E-6 | 3.9E-6* | 3.0E-6 | 1.2E-6 | 1.0E-5 |
| Beloyarsk power Unit 4 / BN-800 | 6.0E-7 | 2.3E-7 | 3.3E-7 | ~1E-8 | 1.2E-6 |
| Beloyarsk power Unit 5 / BN-1200 | 5.0E-7** | – | – | – | 5.0E-7 |
| * Results of the 2009 Analysis.<br>** Results of Preliminary PSA | | | | | |

Correlations of severe core damage frequency (CDF) for different types of initiating events for Beloyarsk power Units 3, 4 are shown in Figures 1 and 2, respectively.

TABLE II: IMPORTANCE OF DIFFERENT TYPES OF SEVERE ACCIDENTS FOR POWER UNITS WITH BN REACTORS.

| **Accident type** | **Probability of severe accidents per one year for power unit** | | |
|---|---|---|---|
| | **BN-600** | **BN-800** | **BN-1200** |
| Contribution of accidents with loss of heat removal | 94 % | 91 % | 91 % |
| Contribution of accidents with loss of primary coolant | 1 % (~1E-7) | 1 % (~1E-8) | 0 % |
| Contribution of accidents with emergency reactor shutdown systems failure to operate | 5 % (~5E-7) | 8 % (~1E-7) | 9 % (~4E-8) |
| Total probability | 1.0E-5 | 1.2E-6 | 5.0E-7 |

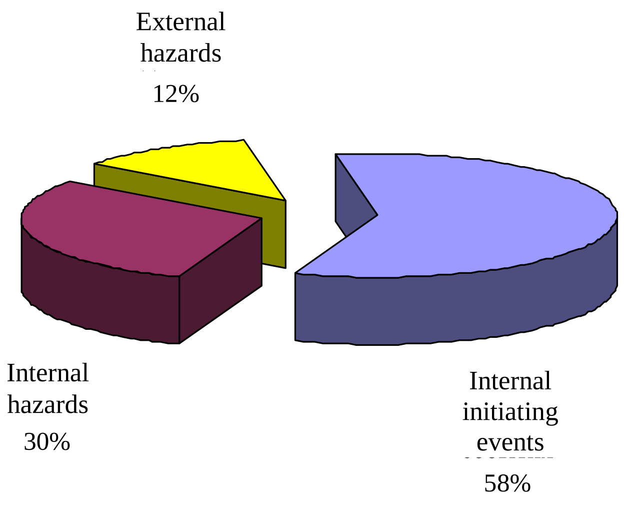


*FIG. 1. Correlation of CDF for different types of initiating events for Beloyarsk power Unit 3 with the BN-600 reactor.*

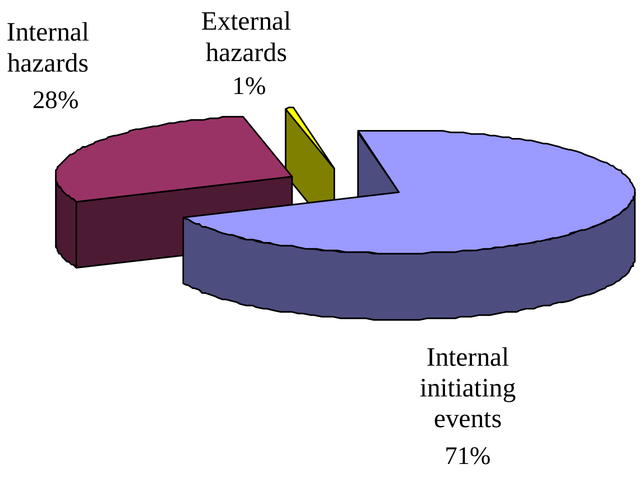


*FIG. 2. Correlation of CDF for different types of initiating events for Beloyarsk power Unit 4 with the BN-800 reactor.*

Emergency sequences with loss of heat removal contribute determinatively in severe accident probability for all power units with BN reactors. The probability of their realization is decreased in BN-800 and BN-1200 designs by applying design decisions with consistent increase of reliability of emergency heat removal systems – increased number of heat-removal channels and enlarged application of passive principles of their functioning. Safety improvement of emergency heat removal systems are shown at Table III.

TABLE III: FAILURE PROBABILITY OF EMERGENCY HEAT REMOVAL SYSTEMS

| **Reactor power unit** | **BN-600** | **BN-800** | **BN-1200** |
|---|---|---|---|
| Structure | 3 trains on third contour + 1 air heat exchanger connected to the second contour | 3 trains (with 2 air heat exchangers) connected to the second contour | 4 trains connected to the first contour |
| Principle of functioning | Active | Active/Passive | Passive |
| System efficiency | 3*100% + 1*100% | 3*(2*50%) | 4*100% |
| Failure probability | ~4E-5* | ~1E-5 | ~1E-6 |
| * With additional emergency heat removal system (EHRS) via air heat exchanger | | | |

The probability of a severe loss-of-coolant accident is decreased in the BN-800 reactor as compared with the BN-600 reactor due to implementation of a syphon break device the BN-800 reactor. In the BN-1200 reactor severe accidents caused by primary coolant leakage are fully prevented because the entire primary equipment is located inside of the reactor vessel.

In BN-800 and BN-1200 reactors, the probability of severe accidents associated with reactor emergency shutdown systems failure to operate is decreased due to application of passive control rods.

## 4. Tasks of Future Studies

At the present time JSC “Afrikantov OKBM” is continuing to improve all PSA models based on additional operation experience and calculations of accident modes for units with BN-600 and BN-800 reactors. The PSA models also take into account safety improvement measures based on Fukushima Daichi accident lessons learned for power plants with BN-600 and BN-800 reactors. Complex of activities on Probabilistic safety analysis Level 2 (PSA-2) is being performed.

The PSA models for the power unit with the BN-1200 reactor are improved in the unit design development process.