Methodological Principles and Results of Probabilistic Safety Analysis for RBMK-1000 NPP (Second Generation Reactors)

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1 INTRODUCTION

The aim of the present investigation is to apply the current methodology for the NPP probabilistic safety analysis (PSA) to the RBMK-type reactors. At this stage, the PSA is limited by the zero level (PSA-0) that includes the inherent probabilistic reliability and different events realization analysis, accident sequences and scenarios. The special investigations of accidents referring to PSA-1 and 2 levels were carried for restricted number of events. Due to the significant differences between the RBMK and other reactor types (PWR, NHR) and the impossibility of using the PSA study experience gained both in the USSR and abroad, completely, a number of principle problems concerning the RBMK PSA were required to be solved. Primarily they are as follows: a) development of general concept to perform the RBMK PSA-0(1); b) choice of methods for investigations; c) solution of the information support problem (data base development). Such aspects effecting the safety as IAEA initial events list, dependent failure, personnel errors, reliability indices of different types, etc. were taken into account. The second generation of the RBMK-1000 NPP type units was chosen as an analysis object [1, 2].

2 METHODOLOGY

2.1 General concept and core failure criteria

In developing the PSA-0 concept for RBMK the following main problems are assumed to be solved:

- identification of the reactor failure modes during severe accidents and establishment of the nomenclature for the probabilistic safety indices (PSI);

- classification of initial (accident) events (IE) being analysed according to groups and subgroups;

- working out the principles for the determination of accident consequences and for the solution of other general PSA problems. The carried out detailed investigations on choosing the RBMK core accidental states permit to single out four core failure categories (and corresponding PSIs):

1) Standard heat removal conditions (S) when the safe operating levels are not violated.

2) Violation of the reactor heat removal conditions (V). These conditions mean an achievement of such a temperature of fuel elements and their claddings at which the cladding failure and fission product release into the coolant are possible (for single or multiple fuel channels). At the same time there is no transient to catastrofic accidental processes, the geometry of the reactor components and the core is preserved.

3) Reactor core damage (D). These conditions are characterized by the severe accidental processes caused by their significant deviation from the design scenario. The finite reactor states lead to partial fuel melting, multiple pressure tube rupture (MPTR) at low pressure.

4) Reactor core severe accident (A) accomponied by MPTR at high pressure and the upper plate uprise in the reactor space or core fuel melting.

Correspondene between RBMK core failure categories and International Nuclear Event Scale Levels [5] can be established as follows: category V can be referred to Levels 1-3; category D corresponds to level 4; and category A - to Levels from 5 to 7. More precise breaking down RBMK core failure criterias in accordance with Event Scale Levels will be possible after detailed investigations in PSA - 1,2.

The radiation consequences should be estimated with regard to localization system (LS) functioning under the accidental conditions of V and D type. The efficiency of LS operation during the accident mitigation is assumed to be considered at the subsequent stages of accident analysis. The core state due to the development of accident event sequences resulting in conditions of V, D and C type, was determined by numerical analysis (within the framework of design safety analysis and separate PSA-1 RBMK investigations) including the extrapolation of numerical relationships as well as by expert estimation method with a required margin.

2.2 Techniques and codes for reliability and safety analysis

The event-tree and fault-tree approaches are used as a base procedure for analysis of accident seqences and PSI assessment. It is necessary to take into account three main factors determining the values of safety systems failure probabilities: a) independent failures of components and their simulteneons occurence; b) dependent failures and common cause failures; c) human errors. The account for each of these factors requires the appropriate methodical, software and information support. The methodical base for assessment of safety system unavailability and initial event probabilities is a "Methodology for analysis of safety related system (SRS) reliability" [3] permitting to perform analyses with account for three abovementioned factors. For these purposes the fault-tree method and the SHARM-2 [4] program package realizing this method was used to assess the reliability of complex systems. The common cause failures (CCF) were accounted for by means of so-called Greek Letter Method (GLM) permitting to calculate the probabilisties of joint dependent failures of two (beta - factor), three (gamma-factor) and four or more (delta-factor) identical elements. The human-errors during accident mitigation were accounted for on the bases of an approach that is based on human-error probability estimation (in decision making) versus the available time.

The uncertainties of the PSA-0 results were analysed versus the initial data ones (input data scattering) with the use of the SHARM-2 computer code. According to these methods that are consistent with the approaches accepted in the International practice it is possible to evaluate uncertainty of reliability and safety indices involving Bayesian approach as well. As a result of SHARM-2 calculations for the system reliability indices (failure probability to demand), the point value and its 90% boundaries are determined. The error factors for components were determined by the statistics operation data and were chosen from the International data banks. The above-mentioned range of problems defines the adopted concept of the RBMK probabilistic safety analysis of zero level. As a whole, it covers all PSA-0 aspects adopted at present in the International practice for such investigation type.

2.3 Initial event (IE) groups

The drawing up of IE list was studied up to now on the basis of the recommended IE list given in special "Guidance", in Engineering Safety Study for the RBMK NPP project (second generation), as well as on preliminary investigations of this problem [1]. Combining the initial events into groups with the identical functional event trees permitted to distinguish seven groups:

- primary pipeline rupture (large size of D=800 mm, medium - 300 mm and small - 80 mm);

- small leak in the primary circuit (partial rupture of the specific size) in the region from the discharge header (DH) of the main circulation pumps (MCP) to the lower water lines (LWL) which leds to stagnation of circulation in reactor;

- steamline rupture; tripping and unfitting of main relief valves (MRVs);

- feed pipeline rupture, feedwater supply failure;

- power unit blackout;

- control rod extraction (single or multiple);

- loss of water in the reactor shut-down system cooling circuit.

The above-mentioned list covers the main design basis initial events as well as a number of events that are not available in the "Recommended List". The hypothetical events as the initial ones were not considered in real form, however, the undesign proceeding of a number of initial events can be related to the hypothetical ones by consequences.

2.4 Primary piping break and leak probability assessment

The probability of rupture or large leak of the primary components being considered was assessed as an initial event probability. Probabilistic fracture mechanics approach was used for this purpose. Input data for computer codes were presented by statistical mechanical properties and crack growth parameters, flaw size distributions and inspection reliability. Some results of probabilistic assessments are presented in Table 1.

Table 1 Probabilities of RBMK-1000 Primary Components Rupture and Leak

Primary piping components	IE group code	Probability of rupture, 1/year	Probability of leak, 1/year
Group Discharge Header	LL1	1.5E-10	1.0E-6
MCP discharge headers	LL3	0.5E-6	1.3E-6
MCP suction headers	LL3	0.2E-6	1.1E-6
MCP Dy 800 pipelines	LL3	1.0E-9	1E-5
Separator connecting pipelines and downcomers	LL4	1.0E-6	1E-4
Lower water lines	SL3	7.2E-8	7E-7

3 RESULTS

Table 2 presents the results of the probabilistic safety indices (PSI) assessment. Annual probabilities (point estimates and 90% upper bounds) are listed separately for each of three core failure categories (and corresponding International Nuclear Event Scale Levels). As follows from the Table 2 the integral annual probabilities of accident appearance with the consequences of category V (heat removal condition violation) have the following values: 2.59E-4 1/year (point estimate); 5.70E-3 1/year (upper bound);

for the consequences of category D: 1.68E-5 1/year (point estimate); 3.28E-4 1/year (upper bound);

for the consequences of category A: 3.24E-8 1/year (point estimate); 1.41E-6 1/year (upper bound).

The power unit blackout accident of 43% and 47%, respectively, makes the most contribution to the integral PSI both for the category of V and for that of D. The dominant accident scenario is blackou t of 0,5 hour or more in failure of the accumulator system or when water is not supplied from the ECCS water tank by personnel as well as blackout of more than 1,5 hour in failure of the accumulator system.

The conditions providing the authorized personnel actions to bring the safety systems into operation, in partucular, RCSS manual reset by the operator in case of feedwater failure at power level of less than 60%, have the most effect.

The dominant contribution to PSI for the steam pipeline rupture was made by false tripping and unfitting of two or more MRVs with the imposition of dependent external blackout, steam pipeline rupture wit h the imposition of long term ECCS failure (including, due to dependent external and internal blackout).

The effect of loss of water accident in the RCSS cooling circuit on the integral PSI is practically absent as the probability of accident with the consequences of category A has the value of the order of 1.E-15 1/year. This conclusion is supposed to be of prime importance as it permits to place this accident into the category of "hypothetical" one.

Table 2. Results of Probabilistic Safety Indices Analysis for RBMK-1000 NPP (probability in 1/reactor-year)

Initial event group and code	Point Estimate for the Failure Category (Int. Nucl.Event Scale Levels)			Upper 90% Bound for the Failure Category (Int. Nucl.Event Scale Levels)		
	V(1-3)	D(4)	A(5-7)	V(1-3)	D(4)	A(5-7)
1. Primary pipeline rupture (LL)	4.2E-7	—	3.1E-8	8.9E-6	—	1.4E-6
2. Small leak in primary piping (SL)	5.1E-7	8.2E-9	—	1.0E-5	3.6E-7	—
3. Steam pipeline accident (PT)	6.8E-6	1.8E-6	—	1.1E-4	3.4E-5	—
4. Feedwater pipeline accident (FW)	7.5E-5	7.6E-6	—	9.3E-4	9.8E-5	—
5. Loss of water in RSS cooling circuit (KS)	—	—	2.0E-15	—	—	1.8E-14
6. Control rod extraction:						
- single (SE)	6.6E-5	—	—	2.4E-4	—	—
- group (GE)	—	2.2E-10	1.9E-9	—	9.8E-10	7.6E-9
7. Station blackout	1.1E-4	7.5E-6	—	4.4E-3	2.0E-4	—
(LP)						
Total:	2.6E-4	1.7E-5	3.2E-8	5.7E-3	3.2E-4	1.4E-6

The zero-level probabilistic safety analysis model of the RBMK-1000 NPP of the second generation, developed in this report, may serve as a basis for performing the PSA of concrete power units of the given type as well as of the reactors of other generation including the RBMK-1500. The results obtained should be considered as preliminary ones and investigations in this area will be continued.

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