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S.V.Petrin, V.Y.Yuferev, A.M.Zlobin

STUDY OF POSSIBILITY USING LANL PSA-METHODOLOGY
FOR ACCIDENT PROBABILITY RBMK RESEARCHES

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LANL Technical Contact

_____ F.Motley

_____ 1995r.

VNIIEF Principal Investigator

_____ A.V.Pevnitsky
22-04 _____ 1995r.

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ABSTRACT

The reactor facility probabilistic safety analysis methodologies are considered which are used at U.S. LANL and RF NIKIET. The methodologies are compared in order to reveal their similarity and differences, determine possibilities of using the LANL technique for RBMK type reactor safety analysis. It is found that at the PSA-1 level the methodologies practically do not differ. At LANL the PHA, HAZOP hazards analysis methods are used for more complete specification of the accounted initial event list which can be also useful at performance of PSA for RBMK. Exchange of information regarding the methodology of detection of dependent faults and consideration of human factor impact on reactor safety is reasonable. It is accepted as useful to make a comparative study result analysis for test problems or PSA fragments using various computer programs employed at NIKIET and LANL.

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INTRODUCTION

Currently probabilistic safety analysis (PSA) is widely used in atomic-energy industry. It is an accepted and highly probably the only means quantitatively estimate sufficiency of taken anti-accident measures.

PSA advantages are as follows [1]:

- PSA enables to secure balanced approach to solving reactor safety problems, purposely and effectively spend funds to enhance the safety level of operating facilities allowing to detect "bottle necks" in safety systems;
- PSA enables to assess expediency of existing structure and technological process modification from the safety standpoint;
- PSA secures assistance for facility leadership in selection of an effective procedure of technological process control;
- PSA secures personnel training at a level adequate to the problem of complex technical system safing;
- PSA secures assessment of the facility control strategy in emergency modes;
- PSA allows to develop manuals and requirements for equipment, procedures providing regular (and even continuous) maintenance of the safety level;
- PSA allows to set priorities of future researches in the safety scope;
- PSA allows to elaborate approaches to development of new concepts of reactor facilities.

Alongside with the advantages, PSA has some specific features ("disadvantages") which give rise to scepticism regarding obtained quantitative analysis results.

It is related to essential effect of subjective factors on the analysis results. One can always doubt in completeness of detected accident models, means of their implementation, effect of emergency probability estimate dispersion (affected by difference in investigators' experience,

completeness of initial data, the used methodology of accident model detection). It impedes probabilistic parameter rate fixing, appropriate safety criteria setting.

There are different ways to avoid the above PSA "disadvantages". One of such ways is information exchange among research centers on methodological issues, gained research experience. Such is the goal pursued by the U.S.LANL-RFNC-VNIIEF contract aimed at studying the possibility to use western (in particular, employed at LANL) codes and techniques for RBMK type reactor safety. The specific set of the contract tasks is specified in the protocol of the Russia-American workshop which was held in December, 1994 in Los Alamos [2]. One of these tasks (task 4, Appendix 3) consists in studying the possibility to use the LANL PSA technique to study RBMK accident probability. It is the solution of this problem that this report is devoted to.

1. PROBLEM FORMULATION

PSA pursues the following goals [1]:

- identification of event sequences (chains) leading to heavy radiation accidents;
- estimation of expected probability of corresponding event chains and possible result indefiniteness degree;
- assessment of radiation accident effects.

Achievement of these goals is secured at different PSA levels. Most frequently the following three levels are distinguished:

PSA-1: detection of radiation accident development ways and estimation of expected accident probability, the indefiniteness degree of this estimate;

PSA-2: detection of the scales of radiation-hazardous material venting to reactor rooms and environment;

PSA-3: assessment of radiation accident effects (degree of environment contamination, personnel and population radiation commitments).

In addition to the above levels, the PSA-0 level is often mentioned (initial data acquisition for estimations) which is often combined with PSA-1, as well as the PSA-4 level (estimation of the risk of human perishing caused by irradiation from cancer diseases or acute radiation disease).

The level of analysis result indefiniteness increases with transition to a higher PSA level. At the PSA-1 level the indefiniteness is usually estimated as one order of magnitude, at the PSA-2 level up to two orders of magnitude, and at the PSA-3 level up to several orders of magnitude (depending on the character of meteorological conditions, terrain features and other factors). Therefore, in many cases the PSA-2 and PSA-3 estimations are performed by deterministic techniques using conservative assumptions over-estimating accident danger. It should be noted that PSA-3 analysis results are not directly related to features of a specific reactor facility but are estimated with account of hazardous material

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aerosol transport processes in the atmosphere. Essentially, all the worldwide techniques describing this transport rely on common relations given, for example, in the reference book [3] and manuals [4,5 and others]. In this sense the PSA-3 level study methodology is universal in its nature and presents no interest in the context of the problem of studying possibility to use the PSA LANL methodology for RBMK safety analysis.

The PSA-2 level analysis is based on neutron physics, thermohydraulics and dynamics deterministic problem solution where reactor facility features can count which determine the character of severe radiation accident development and generation of hazardous venting sources. Possibility to employ western codes, in particular, techniques for solving corresponding problems at RBMK safety analysis is the research subject under the afore-mentioned LANL-VNIIEF contract (tasks 1-3, Appendix 3 [2]), but is beyond the scope of this study.

Thus, specifying the study subject in the context of this report, one can say that the problem reduces to comparison between the reactor safety analysis methodologies at the PSA-1 level used at LANL and NIKIET. In so doing no goal is posed to compare computer codes, since LANL codes were not presented to the report authors by American side (they are commercial codes at DOE disposal).

When the methodologies were compared at the PSA-1 level, a special attention was paid to the following key problems which essentially effect on reactor safety analysis results:

- detection of dependent faults;
- modeling of human behavior (account of human factor);
- detection of final radiation accident events.

These aspects are focused on in Sections 2 and 3.

2. REVIEW OF PSA-1 METHODOLOGY USED AT LANL

2.1. General description of PSA-1 level procedures

At U.S.LANL manuals [6-8] are used at PSA-1 level reactor safety analysis. Thus, the PSA-1 methodology used is not a LANL development and does not contain any specific elements as compared to commonly accepted procedures described earlier in [4,5]. In recent manuals already in early 1980s attention was focused on the afore-mentioned key analysis points, that is the techniques for accounting uncertainties of quantitative estimates of accident probabilities, detection of dependent faults, accounting the human factor effect, external rare effects (natural phenomena, technogeneous accidents, transport accidents). The experience of interaction with staff members of Technology Safety Analysis Department servicing at LANL for over 5 years showed that they take the use of the PSA-1 methodology creatively, suggest their own formats of analysis result representation, participate in search for and selection of acceptable initial data for the analysis (see the reports on the analysis of hazard sources and PSA for Russian fissile material storage facility [9,10]. At the same time the general PSA-1 methodology in their interpretation remains invariable. It gives the right to rely on the classic description of the PSA-1 methodology given in [4,5] when reviewing it.

PSA-1 is usually performed in several stages:

- preliminary analysis at the stage of facility concept elaboration (still prior to facility designing) which enables to determine principal hazards and suggest conceptual solutions taken into consideration at designing to reduce the facility hazard category;
- summary analysis at the stage of facility designing to determine sufficiency of taken safety measures and efficiency of corresponding systems;
- final analysis (after construction, but prior to putting the facility into service) to account the effect on safety of the design deviations which occurred in the progress of facility construction.

The PSA procedure involves:

- initial information acquisition (description of a facility design, of interrelations among facility elements, technical regulations, peculiarities of production site location relative to potential sources of external effects, characteristics of natural phenomena), documenting the acquired data, study of equipment faults probabilities, human errors, frequency of beyond-design external effects (earthquakes, tornados, floods, etc.);
- system analysis (determination of safety functions and the sets of safety systems and safety related systems; modeling of these systems; determination of success effects; construction of "event trees" with account of system response to operation conditions provided by initial events; construction of "fault trees" with estimation of system reliability and uncertainty of this estimation, with account of human behavior models if human actions are needed to achieve a success in safety function performance);
- quantitative results interpretation to determine critical element groups (sets of faults making the highest possible contribution to the emergency probability value).

Uncertainties in the accident probability estimation results are estimated by special codes using the Monte Carlo method and accounting the type of random variable distribution, variation factor values. Should the type of distribution be unknown, for example, due to insufficient statistic data on observation results, estimations by the Bayes theorem [4] are used.

At selection of initial events one leans upon the results of system safety function study which determines the technique of initial event grouping in certain categories, as well as facility operation experience. For PWR, BWR type reactors there is a set of initial events involving about 40 their types. The corresponding list appears even in IAEA recommendations and reflects PSA performance experience for many NPP

with corresponding reactors. However, this practice (the technique of initial event selection "by list") can not be accepted as quite satisfactory. Paper [11] describes some procedures used in the USA at analysis of chemical plants, non-reactor nuclear-hazardous facilities which enable to more completely detect hazards and, hence, are of interest for initial event analysis. As it became known from presentation made by M.K.Sasser at the afore - mentioned December workshop in Los Alamos [12], the PHA, HAZOP, FMEA methods might also find use in the reactor technology. VNIIEF research practice at studying pulsed research reactor safety serves the confirmation of it.

The essence of the PHA method consists in the fact that for a facility being analyzed all possible energy sources and hazardous materials are listed. Then the list of models of possible accidents and their initial (initializing) events is derived by search through the set of specific energy sources and hazardous materials (under assumption of existence of an initializing event which releases energy affecting a hazardous material carrier). Give the following example to illustrate this method as applied to RBMK type reactors:

- the energy source - energy of neutrons transferred to graphite in the course of their slowing-down. This energy can accumulate in graphite if its temperature is lower than 200°C, i.e. graphite annealing will not take place at prolonged reactor operation in a corresponding mode;
- hazardous materials - fuel in fuel elements with radioactive fission products accumulated in it and activated structure materials;
- initializing event - transition to the mode of operation with higher graphite temperature with autocatalytic process of energy release up to 700 cal./g (graphite can heat up to 1500-1600°C, numerous fuel channel damages are possible with steam burst of the reactor)[13].

In the considered example the initial event is prolonged operation at low temperature of the neutron moderator. With PSA-1 it is necessary to determine systems preventing such an event or attenuating its effects.

The essence of the HAZOP method consists in the fact that through technological process analysis a set of key words are selected which characterize hazard sources for a given technology, and of key words which characterize the rate of deviation from technological process requirements (more than, less than, missed, other than set, etc). Then combinations of key words from these two groups are looked over thereby describing the situations among which the risk models calling for further analysis can occur. In so doing logics prompts initial events realizing corresponding risks. Thus, for the afore-considered example the combination of the key words "graphite temperature" and "less than" prompts the danger risk of operation at low temperatures of moderator masonry. The risk, naturally, is determined by a trained analyst.

The essence of the FMEA method consists in studying effects (either direct or indirect) of individual facility, aggregate, system failure. Here failures play the role of initial events initializing hazardous material getting out of hand. This method is convenient at checking whether the "single failure principle" is met and detection of cascade dependent failures.

2.2. Account of dependent equipment faults

At PSA-1 of great importance is accounting and modeling dependent equipment faults. A dependent fault constitutes a multiple failure of several elements whose probability does not reduce to the product of probabilities of individual element failures. Dependent faults involve failures by a common cause (simultaneous or almost simultaneous failures by a single cause) and cascade failures (multiple failures where failure of one element results in failure of another)[14].

The importance to account dependent faults is related to the fact that non-detected dependences between safety system structural elements, external effects (for example, at fires, equipment submergence, etc.), errors with equipment design and manufacture, mounting and

launching-adjusting operations, improper actions of personnel can lead to multiple failures of reserved equipment, failures to execute safety functions at accidents. Non-account of dependent faults can lead to improper conclusions in the progress of PSA (not only from the standpoint of quantitative results, namely, an error of several orders of magnitude, but also from the standpoint of determination of most vulnerable combinations of safety system elements which are minimal cutsets of "fault trees"). it is agreed that dependent faults can make a predominant impact on safety system reliability. Thus, dependent faults, including failures by a common cause, are the principal failure source for massively reserved systems, for example, NPP safety systems [15,16].

To the present day the problem of dependent faults determination can not be considered as completely resolved to a full extent in terms of methodology, since the results of research performed by different groups of experts often disagree. This problem is treated in review [17]. It classifies dependent faults and describes methods for their detection, as well as states that to avoid erroneous conclusions regarding facility safety at high reliability level through equipment reservation, the dependent faults analysis has to be carried out still at early stages of safety system reliability study.

The dependent faults analysis is performed at qualitative and quantitative levels. At the qualitative analysis:

- mechanisms and factors are studied which determine dependence among system elements (either functional or provided by change in environment status);
- potential dependent failures are identified (by system modeling using logic diagrams, namely, "fault trees" or "event trees");
- groups of elements which can be subject to common effects are identified;
- efficiencies of provided protection means to prevent dependent faults or restrict their scale are estimated;

- extremely low-probable events (among identified dependence causes) are classified and rejected;
- adequate reliability models are selected for further quantitative analysis.

The initial information for the qualitative analysis is the same as with ordinary analysis of system reliability. Attention is paid to possible appearance of dependent faults both at facility operations and facility mounting, repair, modernization or removal from operation. In these terms it is interesting to determine impact of people on system reliability, since dependent faults can result from errors at design implementation (non-detected defects of equipment manufacture and mounting, defects related to incomplete and low-quality control over manufacture and mounting operations, equipment repair, poor equipment adjustment and low quality of launching-adjusting operations, improper performance of operations at equipment maintenance and repair, errors at equipment switching or failure to switch equipment at its normal operation or in emergencies). Dependent faults can result from conceptual project-design errors (impossibility or inefficiency of safety function execution in emergencies, functional dependences not detected at projecting, non-adequate computer software setting an improper algorithm for equipment control in an emergency). Such errors can be exemplified by using protective devices whose activation time is more than the reactor runaway period in any emergency.

The dependent event qualitative analysis methods can be divided into two groups:

- analysis of element fault cause impact on spread of the failure to other elements (the FMEA method and others);
- analysis of fault cause matrices for various system elements with detection of common fault causes for various elements. In this case attention is paid to using groups of identical elements (even with account of time and place of their manufacture), the degree of their

functional and design conceptual variety, physical separation, liability to various loadings at external effects, personnel errors at facility designing, manufacture, construction and operation.

The quantitative analysis of dependent faults is performed with account of their classification. Dependence among different elements can either result from deterministic nature functional causes or be of stochastic character. Cascade failures provided by functional causes can have non-unity conditional probabilities in the general case.

At common-cause failure quantitative analysis various parametric models are used (the β -factor model, the principal parameter model, the three-particle model - the so-called Greek - letter model and its analog - the α -factor model and others).

The essence of these models consists in introduction of same measure at accident probability estimation which characterizes the conditional probability of dependent fault occurrence and corrects (basing on statistical observation data) intensities of individual element, system failures. Thus, for example, in the expert assessment method used in U.K. the scale is used showing system reliability variation depending on a reservation technique. If, for example, system non-readiness is characterized with the probability 10^{-1} - 10^{-2} , then at availability of a reserve system with partial difference in operation principles the non-readiness probability for a set of these systems is characterized by the value 10^{-2} - 10^{-4} , at full difference in system operation principles by the value 10^{-3} - 10^{-5} and at availability of another similar channel from two systems with different operation principle by the value 10^{-4} - 10^{-6} .

Inconvenience of the α , β -factor model versions consists in some subjectivism in evaluation of this factor. In doubt is the possibility of transfer of the factor values obtained from some sources to analyze other systems where there are data of independent fault probability. It is related to difference in human factor impact on reliability indications of different systems. Therefore operational statistics data on frequency of

simultaneous failures of similar systems, pieces of equipment are often used for quantitative estimations. It is a so-called principal parameter method. It is convenient should there be needed statistic stock.

2.3. Human factor modeling

At the PSA-1 level one of principal sources of result uncertainties is the human factor [1]. As the practice of nuclear power facility operation shows, accidents are often caused by a complex combination of conceptual project-design errors, structure defects, violation of operation regulations and human errors [18]. At PSA account of human factor impact is attributed to most complicated problems owing to the following human factor peculiarities [19]:

- human performance defies description by simple methods developed to calculate element and system reliability;
- human actions are poorly described by binary logic (effects of stress and alternative responses depending on time, situation, character features, qualification and other factors have to be taken into consideration).

Three principal groups of improper human operator actions are distinguished:

- actions based on improper intentions and errors because of improper event interpretation;
- errors of button and key manipulation;
- performance of initially correct actions with further doubt and transition to an improper way.

At human factor modeling in PSA one usually uses techniques based on employment of operation action tree (OAT), on uncertainty matrix, the Technique for Human Error Rate Prediction (THERP)[19], as well as the expert assessment method. Unfortunately, because of complexity of the problem, human reliability analysis methods have not been sufficient by developed by now, which impedes and restricts their

application in practice. In particular, such a situation leads to the fact that analysis results can be strongly dependent on techniques used. Currently the THERP method is most commonly used abroad (including at LANL).

For better agreement among results obtained by different methods, U.S. EPRI systematized algorithms of human operator error rate prediction (the SHARP-EPRI procedure) [20]. A correlation model of operator reliability (HCR) was developed based on the concept of normalized reliability-temporal curves which enabled to reduce a great number of operator responses to three behaviour mechanism types: skill (capability), rules and knowledge. The normalized curves constitute non-response probabilities $P(t)$ as functions of normalized time t and can be approximated by the three-parameter Weibull function. It is interesting to note that the values of these parameters are highest for the curve corresponding to "skills" and lowest for the "knowledge" curve. It leads to the fact that at $t > 1$ the "skills" normalized curve decreases most rapidly with time allocated for an operator, i.e. provides the lowest non-response probability at a given time.

The HCR correlation model is essentially integral. In order to account the effect of various factors on operator behaviour in more detail a behaviour subelement model (the CSE model) was developed. It is believed that it may turn out to be useful not only to estimate human operator reliability but also to assess the risk from unauthorized (subversive) acts which constitutes a problem closely related with human factor consideration [19].

In studies carried out in the USA on NRC initiative on the problem of NPP resistance against unauthorized personnel actions 25 alternatives were determined on the technology of NPP vulnerability reduction with respect to these acts (STAT) [19,21]. Corresponding actions are determined as "subversive" if they directly initialize the initial event (accident) and as "interference" (sabotage) if they lead to equipment failures without accident initialization. The studies performed showed that

interference is most probable among all considered similar personnel actions and most probable interference forms are vandalizm and arson. Principal causes of vandalizm acts are taken to be loss of jobs, political considerations and psychological anomalies.

Any unauthorized personnel action probability estimate accuracy, naturally, is not high, and under conditions of criminality and terrorism growth quantitative estimation results can severely change. The following is suggested to be introduced as most effective technical solutions which enhance safety system stability:

- passive protection means (constructionally built-in a reactor) with higher resistance to unauthorized actions;
- reservation of system of vital importance for control and protection; introduction of cross connections;
- strengthening control over access to safety systems.

Studies on human factor effect account continue. This problem can not be conceived as resolved nowadays.

3. REVIEW OF PSA-1 METHODOLOGY USED AT NIKIET FOR RBMK SAFETY ANALYSIS

The PSA-1 methodology used at NIKIET when studying NPP safety with RBMK type reactors was analyzed basing on review of specific reports [22,23].

These reports describe PSA principles for the second block of Ignalina NPP with RBMK-1500, give concepts of model construction for PSA-1, present quantitative estimates of probabilities for various emergency types, discuss the statistic data acquisition and processing system needed for development of domestic data base and further specification of estimates obtained on Sweden data base.

The whole PSA-1 methodology is borrowed from foreign sources and agrees with that described in Section 2. In particular, PSA-1 is realized through the following stages:

- description of object, system;
- determination safety functions and systems;
- selection of initial events;
- study of response to the initial events with construction of "event trees";
- study of system serviceability under arising emergencies with account of environment status variation;
- determination of criteria of system operation success;
- identification of events which are nodes of "fault trees";
- construction of system models and "fault trees", calculation of reliability indices.

The analysis employs the Sweden program complex RISK SPECTRUM - PSA [24]. It is implemented on personal computers, covers all of the PSA-1 problems, found its application in 18 countries (as of the year 1992) and can be considered as a tried analysis tool. The complex performs system modeling with construction of "event trees" and "fault trees", assesment of effects, determination of base events and their

grouping, analysis of system functions and determination of minimal cutsets (critical element groups), result uncertainties, sensitivity to initial events and time dependences with in various formats. The complex can distinguish similar failures (by request) from different "fault trees" which allows to do search for dependent failures. At construction of "event trees" binary or other (as desired) principle of event branching may be used. If operators (NPP personnel) are involved in the system models, the program complex takes into account also human factor impact. The data base on human reliability from the reference book [19] is used.

In paper [22] when selecting initial events the authors leaned on the typical list of initial events for RBMK and NPP with RBMK-1500, statistic data about events at NPP with RBMK type reactors, initial event lists for other (PWR, BWR, CANDU type) reactors.

Initial events are categorized into three groups:

- transitional events;
- blocking of a passage section in a coolant circuit;
- loss-of-coolant accidents (formation of various intensity leaks in various technological zones).

As a primary risk the accidents related to disruption of a reactor core of its part. The following is taken for safety functions:

- reactivity control;
- securing integrity of reactor space and rooms;
- coolant circuit pressure maintenance;
- provision of reactor core cooling (including in emergencies).

The methods of risk analysis (PHA, HAZOP) to detect potential risks and initial events are not used. The FMEA method is used in the program complex [24].

Thus, at the PSA-1 stage NIKIET uses the methodology borrowed from foreign nuclear center practice, as well as the Sweeden program complex.

CONCLUSION

1. The PSA-1 methodology used to analyze RBMK type reactor safety does not differ from that used by other foreign design institutions, as well as U.S.LANL.

2. It is reasonable to try the risk analysis methods used at U.S.LANL (PHA, HAZOP) for more complete determination of the initial event list. These analysis methods are formalized and, hence, can be useful when using PSA-1 for any reactor and off-reactor facilities including RBMK. Their mastering does not call for development of any special codes (computer programs) and, therefore, does not call for joint (with LANL) research, comparative analysis through test problem solution.

3. Existing PSA-1 level methodologies common for all countries designing NPP regarding human factor accounting, determination of dependent faults make useful the information exchange about gained experience of probabilistic safety analysis performance. It is reasonable to call on different expert groups to analyze some facilities with comparison of study results which allows, on the one hand, to enhance confidence in overcoming of subjective factors in the process of PSA-1, and, on the other hand, to reveal some specific features of reactor facilities advancing skill of researchers. From this standpoint additional contacts, for example, among LANL, NIKIET and VNIIEF, are possible at the stage of PSA-1 performance for specific reactor facilities. In particular, it is interesting to compare the codes used at LANL and NIKIET through comparative reliability analysis for selected systems (for example, RBMK control and protection systems and their auxiliary subsystems) under conditions of agreed initial data and failure criteria. Another example of such comparison might be performance of separate PSA fragment RBMK to the full extent, i.e. analysis of a set of accident series related to a single initial event, accident. Here the result might be compared obtained from computations by a whole set of codes and characterizing probability of various accident effects.

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