DEALING WITH UNCERTAINTIES IN NUCLEAR SAFETY ANALYSES – PART I (PROBABILISTIC APPROACH)

Minodora APOSTOL[1], Aureliu LECA[2]

Lucrarea este structurată în două părţi: (I) tratarea incertitudinilor în abordarea probabilistă; (II) tratarea incertitudinilor în abordarea deterministă.

Partea I prezintă contribuţiile autorilor la dezvoltarea metodologiei analizei de incertitudine din studiul probabilist (PSA) de evaluare a securităţii nucleare. A fost realizată o evaluare a incertitudinii valorilor parametrilor de bază ai modelelor PSA de nivel 1 folosite în Institutul de Cercetări Nucleare Piteşti pentru evaluarea securităţii instalaţiilor nucleare şi a fost dezvoltat un modul pentru analizele de incertitudine.

Partea a II[a] va prezenta contribuţii originale în estimarea incertitudinilor şi influenţa lor în rezultatul final pentru abordarea deterministă, în cazul transportului produşilor de fisiune în circuitul primar CANDU, în timpul unui accident sever.

The paper is structured in two parts: (I) uncertainties treatment for probabilistic approach; (II) uncertainties treatment for deterministic approach.

Part I presents the contributions of the authors to the development of the methodology for uncertainty analysis in probabilistic safety assessment calculations. An uncertainty evaluation on the basic parameter values of the PSA level 1 models used in INR Pitesti for safety evaluation of nuclear facilities has been performed and a dedicated module for uncertainty analyses was developed.

Part II will present some original contributions for the uncertainties estimations and their influences in final results for deterministic approach in the case of fission products transport in CANDU primary circuit during a severe accident.

Keywords: aleatory, epistemic, PSA, error factor, uncertainty analysis

1. Introduction

Two classes of methods are used for the safety evaluation of a Nuclear Power Plant (NPP): deterministic and probabilistic. The methods are complementary and both are based on specific models. A model means the attempt of the analysts to represent the reality: NPP behavior, system behavior, initiating event occurrence, different phenomena involved in Severe Accident (SA) analyses, etc. In the physical sciences or engineering disciplines the models

[1] Scientific Researcher, Institute for Nuclear Research Pitesti, PhD Candidate, University POLITEHNICA of Bucharest, Romania, e-mail: minodora.apostol@nuclear.ro

[2] Prof., UNESCO chair on Engineering Sciences, University POLITEHNICA of Bucharest, Romania

are based on a mathematical structure leading to numerical results as a representation of the reality's behavior [1]. Since these models try to represent as accurate as possible the reality, it is impossible to capture all the aspects, so the uncertainties are inherently introduced by the modelling process. Also, many parameters used by the models are not exactly known, from different reasons: database limitations, limited applicability of the available data, different expert opinions, interpretation of the analysts, etc. The addressing of uncertainties is more and more used in the decision making process regarding nuclear safety. The results of any models do not offer enough credibility without an understanding of the uncertainties based on detailed studies [2].

Uncertainty analyses (UAs) are performed to determine the impact of the uncertainties on the results of the models. In Probabilistic Safety Assessment (PSA) studies, UAs represent a very important aspect. The objective of UAs in PSA is to assure the qualitative evaluation and quantitative assessment of the uncertainties in the final or intermediate results [3].

The purpose of this paper is to present some efforts performed in Institute for Nuclear Research (INR) Pitesti, in dealing with uncertainties in PSA field (Part I) and in the Fission Products (FPs) deposition and transport in the CANDU Primary Heat Transport (PHT) System, during a severe accident (Part II).

2. PSA and uncertainties

PSA study is applied to many areas, the most important quantity of resources being allocated for nuclear field. From the first modern PSA study (Reactor Safety Study WASH-1400) published in 1975 by the US Nuclear Regulatory Commission, PSA technique has become an important tool to assess the safety of NPP, especially after the accidents from Three Mile Island – 2 (1979) and Chernobyl (1986) [4].

Some of the advantages of PSA analyses are related to a good understanding of the plant design, of its performances, of the impact on the environment and regarding the identification of dominant risk contributors, offering a consistent and integrated framework for safety related decision making. Moreover, PSA is a conceptual and mathematical tool to obtain numerical estimates of risk for NPP and industrial facilities [3].

In international practice three levels of PSA are used [3], [5]:

- PSA Level 1 which involves the identification and quantification of the sequences of events leading to core damage;
- PSA Level 2 which uses the results from Level 1 and involves the evaluation and quantification of the mechanisms, amount and probabilities of subsequent radioactive material releases from the containment;

- PSA Level 3 which uses the results of Level 2 and involves the evaluation and quantification of the resulting consequences to both the public and the environment.

A major advantage of PSA consists of the possibility to quantify the uncertainties [3].

Two categories of uncertainties are recognized in the literature: aleatory and epistemic. The aleatory uncertainty gives PSA the probabilistic part of its name [6]. We are talking about aleatory uncertainty when the events or phenomena which are modeled are characterized as occurring in a "random" or "stochastic" manner (e.g. we do not know for sure when a pump or a motor will fail or when a reactor trip or a Loss of Coolant Accident - LOCA event will occur, or if a safety system will start successfully on demand, etc.). In order to describe their occurrences, probabilistic models are adopted (e.g. models for pump failure – on demand or during mission, models for Initiating Events (IEs) occurrence, models for probability of safety systems successfully start, models for human error probability, etc).

The epistemic (or state-of-knowledge) uncertainty is associated with the analyst's confidence in the prediction of the PSA model itself. It reflects the analyst's assessment of how well the PSA model represents the event or phenomena that have to be approached by the model [7].

By definition, the aleatory uncertainty is irreducible, whereas the epistemic uncertainty can be reduced by further study. Also the aleatory uncertainty is implemented in risk by the application of an appropriate probabilistic model, unlike epistemic uncertainty can be used to generate bound on the risk assessment, these bounds showing the range in which the risk distribution can vary [8].

Epistemic uncertainties influence and impact the results of PSA (core damage frequency for PSA level 1, frequencies of different classes of releases, which include large release, early release, large late release, small release – for PSA level 2 and frequencies of early fatalities and latent cancer fatalities for PSA level 3). This kind of uncertainty is divided into parameter uncertainty, model uncertainty and completeness uncertainty [1], [3], [6] and [9].

Parameter uncertainty is the most quantifiable among the three types of uncertainties and means the uncertainty in the values of the main parameters of the PSA model: equipment failure rates, IEs frequencies, human error probabilities, common cause failures. These parameters are used to quantify the core damage frequency (CDF), which is the risk metric generated by a PSA Level 1.

Model uncertainty is connected with the approximation of the reality. Since it is impossible to obtain an exact representation, certain limitations of any model occur. Model uncertainty could mean the uncertainty about the logic structure of the PSA model or in the choice to estimate the probabilities of basic events.

Completeness uncertainty refers to the problem that not all the possible scenarios (which can lead to undesirable consequences) are identified and assessed in the analysis. This lack of completeness introduces uncertainties in results and conclusions of PSA analyses, which can be difficulty evaluated.

2.1. Uncertainty analyses in PSA

In PSA studies, UAs play an important role, their objective being to provide quantitative measures and qualitative discussion of the uncertainties in the final results of the PSA (the risk to public health and safety) or intermediate results (the core damage frequency, the radionuclide release frequency or fatality frequency, etc.) [3], [9]. UAs can be performed qualitatively and/or quantitatively. A qualitative study consists of the identification of uncertainties and the qualitative evaluation of their importance [10], [11]. The aim of a quantitative study is to understand and to see the impact of the uncertainties associated with the values of PSA basic parameters in the final or intermediate results of a PSA study. The quantitative study may involve some or all of the following steps: evaluation of uncertainties in the input to each of the tasks of a PSA; propagation of input uncertainties through each task; combination of the uncertainties in the output from the various tasks and displaying and interpretation of the uncertainties in the PSA results.

There are many levels of uncertainty analysis: almost entirely consisting in a qualitative treatment of uncertainty; quantitative treatment of parameter uncertainty with a qualitative treatment of model and completeness uncertainty; quantitative treatment of both parameter and model uncertainties with a qualitative treatment of completeness uncertainties; and analysis of all these three types of uncertainty including a quantitative estimate of completeness uncertainty [10].

The simplest level of uncertainty analysis may be the qualitatively one with the scope to identify and ranking the most important uncertainties. To understand the impact on the PSA Level 2 results, an identification of the uncertainties in each task is necessary. PSA Level 2 involves the response of the containment and its safety systems to the loads attending core damage accidents. The results of PSA level 2 consists of containment failure states and their frequency of occurrence and the releases to the environment [12]. As has been mentioned earlier, PSA level 2 uses the results of PSA level 1. Starting from initiating events (LOCA, transients), using Event Tree (ET) and Fault Tree (FT) methods, accident scenarios could be obtained. The sum of all accident scenarios frequencies means Core Damage Frequency (CDF), which is the final result of PSA level 1 (Fig. 1.) The starting point for PSA level 2 represents the grouping of very large number of accident sequences from PSA level 1 into a smaller number of Plant Damage States (PDSs) that would be expected to have similar effects on containment response and fission product source terms. After the screening of low frequency PDSs, the accident progression and the impact on containment behavior are studied probabilistically by event trees. The different end states of the Containment Event Trees (CETs) are grouped into sets of release categories and distinct source terms are evaluated [13] (Fig. 1).

In [14] a preliminary qualitative uncertainty analysis is presented, by identification of major uncertainties in PSA level 1 – level 2 interface (PDSs grouping and screening for low frequency) and in the other two major procedural steps of a PSA level 2: analysis of accident progression and of containment and analysis of source term for severe accidents.

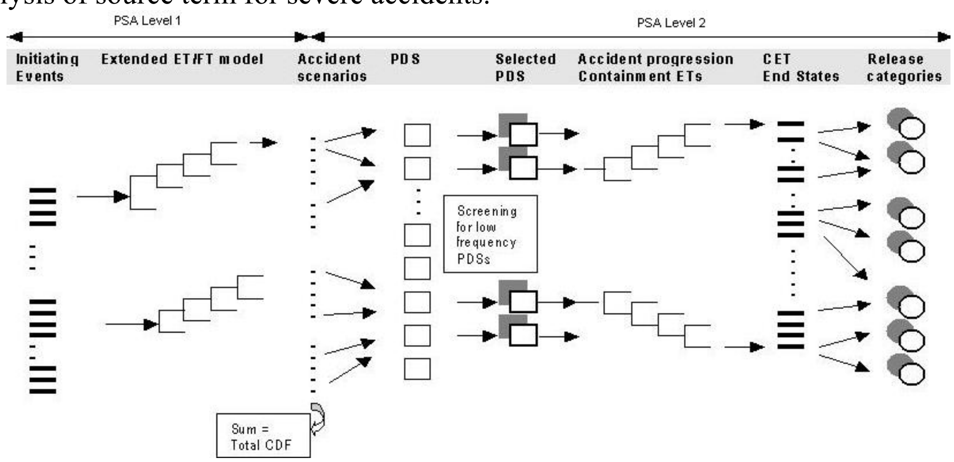

Fig.1. The major analytical elements of a PSA level 1 and level2

The identification of these uncertainties represents an important step in highlighting the points from PSA level 2, on which the precision of the study should be improved. These points can be [14]: the identification of the attributes (interface characteristics) so that all the accident sequences resulted from PSA Level 1 and grouped into PDSs to be identically treated in progression accident analysis; a clear formulation of the nodal questions used in development of CETs; modeling of physical phenomena; the parameters values of physical phenomena; the identification of appropriate attributes which affect radiological releases and consequences on the population and environment, etc.

2.1.1. Quantitative uncertainty analyses

A quantitative uncertainty analysis assesses the amount of the impact of uncertainties on final or intermediate results of a study. If we are talking about a PSA level 1, the final result supposes the core damage frequency (CDF) and the intermediate results could mean the system unavailability, the IE frequency or the accident sequence frequency. For the first time, the uncertainty analyses were introduced in WASH-1400 study, anticipating the idea that decision makers will not accept a single number as being the probability of a catastrophic accident coming from a nuclear reactor. So, starting from the estimation of uncertainties from PSA model input, a probability distribution of the final result has been obtained [4].

For a quantitative uncertainty analysis, which has to offer the credibility of a mathematical model result, the following steps are necessary: evaluation of uncertainties in input values; uncertainty propagation through the model and display and interpretation of results.

The first step is very important and for this reason this paper is mainly focused on the uncertainties evaluation for fundamental parameters of the PSA level 1 models used in INR for nuclear safety evaluation.

2.1.2. The uncertainty evaluation on the basic parameters values of the PSA level 1 models

In our methodology the uncertainty in the fundamental parameters values (e.g. failure rates, IEs frequencies, human errors probabilities, undeveloped events and common cause failures) used in PSA Level 1 models, is represented by using lognormal distributions. The lognormal distribution is broadly used in engineering area, in risk and reliability analyses and plays a very important role in uncertainty propagation, because the confidence interval of reliability parameters is expressed by means of error factor (*EF*). If x is a random variable, it falls in the confidence interval $\left[\frac{med(x)}{EF},\ med(x)\cdot EF\right]$ with a *0.90* probability (Fig.2), where $med(x)$ means the median of x [15] and [16].

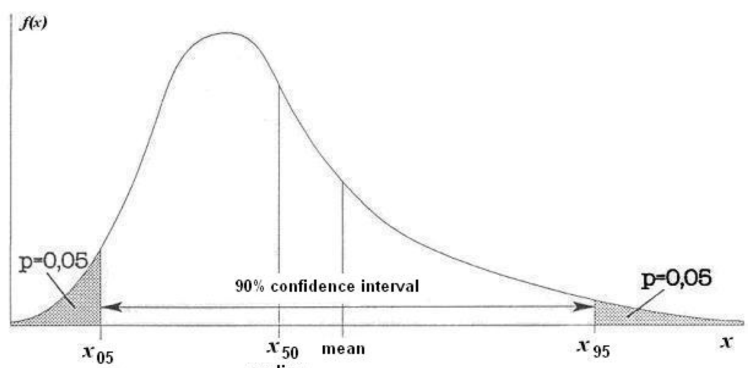

Fig. 2. Lognormal probability density function, its fractiles, mean and 90% uncertainty range

Considering the major parameters of the PSA models as random variables with a defined range, a probability density function (*pdf*) such as the lognormal, can be assigned in order to obtain the likelihood of occurrence of any particular value.

Uncertainty evaluation for failure rates

The reliability database used in INR Pitesti for safety evaluation of nuclear facilities is a generic, MsACCES database and it consists of 2087 records structured as: type, subtype and the class of component; failure mode; component boundary; mean value of failure rate (lambda), lambda 5%, lambda 50%, lambda 95%, mean time to repair (MTTR); data source used.

The error factors, compulsory necessary in uncertainty evaluation of failure rates and consequently in uncertainty analyses, were not specified. To convert the different types of generic data in lognormal distributions, 8 cases of error factors calculation have been identified [17] and 8 situations in this database have been found. By means of a program developed in Visual Basic, all the records from database have been completed with EFs for failure rates [18]. In Fig. 3, some records from reliability database are presented, after their completion with EFs.

Reliability Data					
DataID	**Mean λ**	**5% λ**	**50% λ**	**95% λ**	**ErrorFactor**
54	0.00013				10
596	0.00017	0.00001	0.00017		10
20123	0.00014	0.00005	0.00026	0.00068	3.69

Fig. 3. Records from reliability database, with EF

Uncertainty evaluation for IEs frequencies

An IE is an event that perturbs the steady state operation of an NPP, thereby initiating an abnormal event such as a transient or LOCA within a plant [16]. IEs are modeled as aleatory occurrences so the values of their frequencies are never precisely known. To justify this lack of knowledge of the IE frequency value, an EF (as a major characteristic of lognormal distribution) is introduced. The assignation of EFs depends on the IE frequencies derivation methods. A presentation of IEs frequencies derivation methods used in some available CANDU PSA analyses (Darlington Probabilistic Safety Evaluation - DPSE and in Bruce NGS B Risk Assessment - BBRA) and a description of the IEs frequencies derivation methods that can be used/were used in Romania for nuclear facilities (including Cernavoda NPP) can be found in [19]. The EFs can be calculated or assumed, taking into account the derivation methods used. By means of the following methods, the EF can be calculated [20]:

- *the fault tree method using specific/generic reliability data.* The sampling method used could be Monte Carlo and the final results may be expressed as *pdf*

or as *cdf* (cumulative distribution function). The EF for the resulted lognormal distribution is the ratio between 95% bound and the median value;
- *the Bayes method* that combines the station specific experience with the generic IEs frequencies. The EF is defined as the ratio of the 95th percentile of the lognormal posterior distribution to the median frequency.

The EF may also be calculated for a group of IEs, if for each IE of group the EF and the frequency of occurrence (the mean value) are known. This calculation is based on method of moments [10]. The method of moments approximates the probability density functions which characterize the uncertainty in the input parameters by their first two moments (the first moment is the mean - m and the second one is the variance - v) and it is used to estimate the moments of the output from the moments of the uncertain input parameters [21]. The algorithm presented in [20] is only for 2 IEs from the group and supposes that distribution of each IE of the group is lognormal.

The EF of a group that contains 2 IEs is calculated using the following formula:

$$EF_{group} = e^{1,645\sqrt{\ln\left[\frac{v_{group}}{m_{group}^2}+1\right]}} \quad (1)$$

where v_{group}, m_{group} and EF_{group} represents the variance, the mean and the error factor of the resulted lognormal distribution.

As it was mentioned above, the EF can be also assumed and the following methods are used for this purpose:
- *the method based on piping specific design data and generic data*. The following elements are necessary in using this method [19]: the examination of isometric schemes for the system whose frequency have to be calculated; information on weld locations, piping components (bends, elbows, pipes, tees); numbers of bends, elbows, tees, welds; the length of tees, elbows, bends, pipes; piping design; operating conditions; estimation of piping reliability parameters (rupture frequency for: pipes, tees/elbows/bends, bend/elbow welds, pipe to pipe welds); estimation of pipe rupture frequency. An EF of 5 is supposed to be in accordance with the uncertainty of the IE whose value can be calculated with this method [20];
- *non-informative prior distribution*. It is used when little or no generic prior information is available. An error factor of 10 has been assumed for IEs with zero occurrences. The value of 10 results from the chi-square distribution, which is the posterior distribution for this type of event [20].

Uncertainty evaluation for human error probabilities

The models used in INR Pitesti for pre-accident and post-accident human errors probabilities are based on decision event trees and Accident Sequence

Evaluation Program (ASEP) generic data [22]. To know the interval of possible values which the human error probabilities have to take it in order to propagate the uncertainty through the mathematic model (fault tree, accident sequence, etc), the lognormal *pdf* has been considered appropriately. For each sequence from the decision event tree, the human error probability and the error factor are given [22].

Uncertainty evaluation for common cause failures and undeveloped events

The undeveloped events are those which are not further developed in order to allocate them the failure rates. A direct unavailability assignation is done and the EF for this kind of events is 10 [18].

The Common Cause Failures (CCF) represents dependent events in which the failure states of two or more than two components appear in the same time or in a very short interval, due to a common cause. In PSA analyses performed for Cernavoda NPP unit 1, the CCFs have appeared in the fault tree models. The EF value for these events is 5 and has been adopted during discussion with the CCF analyst [18].

3. Some contributions to UAs software development

Another important step in uncertainty analysis is represented by the uncertainties propagation through the mathematical models. This means how the uncertainties of the input values are propagated to the final result of the model. In Fig. 4 a scheme of uncertainty propagation through a model **G** that depends on P_1, P_2 and P_3 parameters is shown, where f_{P_1}, f_{P_2} and f_{P_3} represent the parameters' uncertainties described by different *pdf*s and f_G means the uncertainty in the final result of **G**, characterized by its *pdf*.

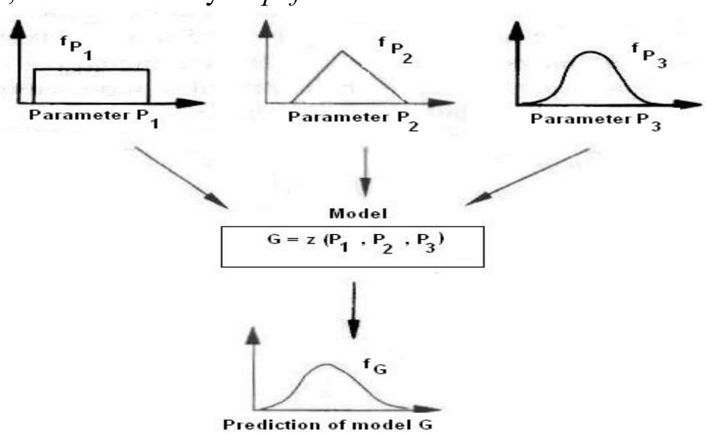

Fig.4. Uncertainties propagation through model G

In almost all cases, a quantitative uncertainty analysis is performed by numerical simulations that are dependent on the propagation methods [4]. These methods can fall into non probabilistic (interval analysis, uncertainty propagation using sensitivity derivatives, possibilistic methods – Fuzzy logic) and probabilistic methods [21]. The probabilistic methods can be divided in statistical methods (Monte Carlo technique as a classical example, probability bounds analysis [23]) and non statistical methods (the method of moments).

The computer programs package PSAMAN [24] was developed in INR Pitesti by the PSA team and it was used for Cernavoda unit 1 PSA level 1 analyses. That version did not include the uncertainty analyses. From this reason some efforts have been performed to develop computer applications able to accomplish integrated PSA analyses.

Based on the computer code MODULE [25], dedicated to fault trees analyses and which also performs uncertainty analyses by combining the Monte Carlo method with the method of moments, we have developed some mathematical and numerical models in order to obtain a complete uncertainty analysis for complex fault trees [26]. After that, a graphical interface to easy handle the modified MODULE was developed. The new application, called INCERT [27], allows the use of the same files types as application EDFT (included in PSAMAN computer programs package). INCERT has a single form and takes different aspects [26]. Related to the values of the EFs, a default value of 3 for basic event (other than human errors and undeveloped events) and a value of 10 for these have been used. The uncertainty analysis performed, was offering the uncertainty in the final results (the unavailability of the complex systems modeled for Cernavoda NPP unit 1), by propagation of uncertainty parameters through the fault tree model. The following information were available after performing UAs: the summarized results of the cumulative distribution function (median value, *EF*, mean value, standard deviation) and distribution confidence limits [26]. The INCERT module has been tested and a good agreement has been observed with EDFT program results, related to the point mean value [26].

In order to enhance the confidence in the results by reducing the possible errors introduced by the user in assigning the EF for the basic events and by using of correct data, the initial application INCERT has significantly been improved. It was also integrated into NewL [24] module (also part of PSAMAN package), a program that allocates the reliability data, quantifies the human errors and labels in a unique way the components. The new application called NewL&INCERT [28] represents an efficient tool for uncertainty analysis at fault tree and accident sequence level. The error factors for components failures and human errors probabilities can now be automatically assigned [29]. NewL&INCERT also recognizes the common cause failures and the undeveloped events and for these, the user must type an EF of *5*, respectively of *10*.

To propagate the parameters uncertainty into a probabilistic model representing the frequency of initiating events obtained by means of fault tree, INCERT_M application has been created [18]. The development of this new application has been necessary because the mathematical model for the *IE* frequency obtained using fault tree method is different from those of obtaining the top event unavailability, also by using the fault tree method. The equation (2) represents the unavailability of a top event - $\overline{A}_{TE}$, where N means the number of Minimal Cut Sets (MCS) and $\overline{A}_i$ represents the unavailability of an MCS. $\overline{A}_1, \overline{A}_2$, …. from (3) represents the unavailability of all components from a MCS and n_i means the number of components from MCS.

$$\overline{A}_{TE} = \sum_{i=1}^{N} \overline{A}_i \tag{2}$$

$$\overline{A}_i = \overline{A}_1 \cdot \overline{A}_2 \cdot \ldots \cdot \overline{A}_{n_i} \tag{3}$$

Equation (4) represents the frequency of occurrence of a top event - λ_{TE} where N means the number of MCS and λ_i represents the frequency of occurrence of an MCS. $\overline{A}_1$, $\overline{A}_2$, …. and λ_1, λ_2, … from (5) represents the unavailabilities respectively the failure rates of the components from MCS.

$$\lambda_{TE} = \sum_{i=1}^{N} \lambda_i \tag{4}$$

$$\lambda_i = \overline{A}_2 \cdot \overline{A}_3 \cdot \ldots \cdot \overline{A}_{n_i} \cdot \lambda_1 + \overline{A}_1 \cdot \overline{A}_3 \cdot \ldots \cdot \overline{A}_{n_i} \cdot \lambda_2 + \overline{A}_1 \cdot \overline{A}_2 \cdot \ldots \cdot \overline{A}_{n_i} \cdot \lambda_3 + \ldots\ldots + \overline{A}_1 \cdot \overline{A}_2 \cdot \ldots \cdot \overline{A}_{n_i - 1} \cdot \lambda_{n_i} \tag{5}$$

Therefore, starting from this requirement, the mathematical and numerical model has been developed in order to obtain the uncertainty in the initiating event frequency, propagated through the fault tree model.

3.1. Illustrative results of uncertainty propagation through the PSA level 1 models

Below, uncertainty analyses at fault tree level [28] and accident sequence level [30] are presented. Both analyses are performed using NewL&INCERT.

To evaluate the uncertainty in the unavailability of a system, the top event called **REC** (D_2O Storage, Transfer & Recovery System fails to supply feed pumps after very small LOCA; *Failure criterion*: 2/2 3333-P2, P3 fail to automatically start or to operate; *Mission time*: 24 hours) has been chosen [28]. The top event **REC** has been modeled for Cernavoda Unit 1 Probabilistic Safety Evaluation study and it is detailed presented in [31]. The summarized results of the cumulative distribution function (median value, EF, mean value, standard deviation) and distribution confidence limits are presented in Fig. 5.

DISTRIBUTION VALUES

NUMBER OF SAMPLE = 1200
POINT MEAN VALUE = 9.13080E-03
MEDIAN VALUE = 6.80672E-03
ERROR FACTOR = 3.34861E+00
MEAN VALUE = 9.08598E-03
STANDARD DEVIATION = 7.60295E-03
5% BOUND = 2.21524E-03
95% BOUND = 2.27930E-02

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERRCENT)	FUNCTION VALUE
.5	1.17483E-03
1.0	1.49765E-03
2.5	1.78162E-03
5.0	2.21524E-03
10.0	2.80202E-03
20.0	3.78359E-03
25.0	4.29373E-03
30.0	4.75546E-03
40.0	5.73812E-03
50.0	6.80672E-03
60.0	8.24881E-03
70.0	1.02151E-02
75.0	1.13434E-02
80.0	1.29075E-02
90.0	1.75665E-02
95.0	2.27930E-02
97.5	2.88812E-02
99.0	4.05536E-02
99.5	4.96753E-02

Fig. 5. Results for uncertainty analysis of REC top-event

The characterization of parameters uncertainty from the fault tree model is made by using lognormal distribution. The resulted *error factor EF = 3.34* shows the probability that the unavailability of the top event to fall into *[2.0327E-03, 2.28E-02]* interval is *0.9*.

To see the influence of parameters uncertainty on the frequency of an accident sequence expressed as occurrences/year, the DCCF9 sequence from "Dual Computer Failure" Event Tree [32] has been selected. 50 MCS have been obtained after the integration of the representative systems of this sequence (Recirculated Cooling Water, Moderator System, electrical power supply systems, etc), modelled by using the fault tree method, in the event tree used to construct the sequence. The value of an MCS is the product between the probabilities of all elements from MCS (component failures - BE_{i1}; human errors - BE_{i2}) and the frequency of initiating event $frequency\left(IE_j\right)$(6) and the final value of accident sequence frequency is the sum of all MCS frequencies.

$$Frequency\left(MCS_i\right) = frequency\left(IE_j\right)BE_{i1}BE_{i2}... \tag{6}$$

The uncertainty analysis made for DCCF9 sequence, has shown an error factor of *14.3* resulted from the cumulative distribution function of accident sequence frequency, this meaning a low confidence in the resulted value [30]. An important contributor to DCCF9 sequence occurrence is the event called "*The operator does not succeed to mitigate the accident after dual computer failure*" and because the choosing of probability for this human error was not very clear from the available documentation, an EF of *10* has been applied. A new uncertainty analysis, by adopting an EF of *5* for the same human error probability has been made and the resulted EF was *7.6* This new value shows that it is very important to know the derivation method of every parameter that enters the model of accident sequence.

Regarding the point mean value of IEs frequency, firstly a comparison between INCERT_M program and EDFT program has been made for some initiating events, as can be seen in table 1 [18], [33], where:

- CSDVO: Condenser steam discharge valves spurious opening (one or more valves);
- HTPT: Spurious PHT pump trip when reactor is operating in 2/2 mode;
- LIAF: Loss of instrument air flow;
- LIAI: loss of instrument air inventory;
- LRSWI: Loss of raw service water inventory;
- LOCGDC: Loss of cover gas deuterium control;
- PLMC: Partial loss of moderator cooling.

Table 1

Initiating Event frequency comparison

INITIATING EVENT			INITIATING EVENT FREQUENCY (POINT MEAN VALUE)	
Top Events	*Max Size*	*Max Value*	*INCERT_M*	*EDFT*
CSDVSO	200	1.E-07	1.21740E+00/1000 years	1.2173999994 over 1000 years
HTPT	200	1.E-07	2.01120E+02/1000 years	201.11959406 over 1000 years
LIAF	200	1.E-07	1.53234E+02/1000 years	153.23390264 over 1000 years
LIAI	200	1.E-07	8.52042E+01/1000 years	85.204199571 over 1000 years
LRSWI	200	1.E-09	2.35908E+01/1000 years	23.5908002 over 1000 years
LOCGDC	200	0	1.67695E+01/1000 years	16.7695216 over 1000 years
PLMC	200	0	3.08226E+01/1000 years	30.8226 over 1000 years

Results of uncertainty analysis for IE LIAF are presented in Fig. 6.

DISTRIBUTION VALUES

NUMBER OF SAMPLE = 1200
POINT MEAN VALUE = 1.53234E+02/1000 years
MEDIAN VALUE = 1.52308E+02/1000 years
ERROR FACTOR = 1.18734E+00
MEAN VALUE = 1.60213E+02/1000 years
STANDARD DEVIATION = 2.04974E+02/1000 years
5% BOUND = 1.28276E+02/1000 years
95% BOUND = 1.79310E+02/1000 years

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERRCENT)	FUNCTION VALUE
0.5	1.08957E-01/1000 years
1.0	1.14349E+02/1000 years
2.5	1.23360E+02/1000 years
5.0	1.28276E+02/1000 years
10.0	1.33228E+02/1000 years
20.0	1.39766E+02/1000 years
25.0	1.42356E+02/1000 years
30.0	1.44305E+02/1000 years
40.0	1.48339E+02/1000 years
50.0	1.52308E+02/1000 years
60.0	1.56159E+02/1000 years
70.0	1.60308E+02/1000 years
75.0	1.62863E+02/1000 years
80.0	1.65682E+02/1000 years
90.0	1.73067E+02/1000 years
95.0	1.79310E+02/1000 years
97.5	1.84258E+02/1000 years
99.0	1.92213E+02/1000 years
99.5	1.97574E+02/1000 years

Fig. 6. Results for uncertainty analysis of LIAF initiating event

The characterization of parameters uncertainty is made by using lognormal distribution, the uncertainty propagation technique is Monte Carlo, simple random sampling method and the number of samplings for all aleatory variables from the model is *1200*.

From this cumulative distribution function that characterizes the uncertainty in the frequency of “Loss of instrument air flow” initiating event we can read the median value, the error factor, the mean value, which are important parameters of an uncertainty analysis. The EF resulted (*1.18)* offers a good credibility of the frequency of this IE.

4. Conclusions

(1) UAs play an important role in PSA studies aiming to provide quantitative measures and/or qualitative discussion of the uncertainties in the final results (the risk to public health and safety) or intermediate results (the core damage frequency, the radionuclide release frequency or fatality frequency, etc.). The preliminary qualitative uncertainty analysis performed, provides the points on which the precision of the PSA level 2 study should be improved.

(2) The reliability database used in INR Pitesti for safety evaluation of nuclear facilities was completed with error factors for failure rates. For other parameters of the PSA level 1 models the assignment of error factor was based on different methods, literature and expert judgment. In this manner a useful tool was obtained.

(3) An important effort was performed in order to develop methods and tools to perform UAs. In INR Pitesti the PSA computer code package used to perform safety evaluation of nuclear facilities, was completed with a dedicated module to perform UAs. An integrated system able to compute PSA parameters together with the associated confidence limits was obtained.

(4) The results obtained on different test cases show a good agreement with the results of other codes.

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