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Human reliability analysis for probabilistic safety assessment of a nuclear power plant

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Despite the high reliability of safety systems of nuclear power plants (NPP), human actions still play an important role in NPP safety. Evaluation of human reliability is therefore important for a full-scope probabilistic safety assessment (PSA) and risk analysis.

In PSA, which includes modelling of interaction of safety systems during an accident sequence, human errors are modelled together with hardware failures. Methods for the modelling of human errors and evaluation of probabilities of such errors are different compared with methods used for modelling and probability estimation of equipment failures.

The paper describes the modelling of actions that were to be performed by the Ignalina NPP operators during an accident sequence. Such modelling was applied for the PSA of the Ignalina NPP. A combination of ASEP (Accident Sequence Evaluating Procedure) and THERP (Technique for Human Error Rate Prediction) was applied. Such methodology allows evaluating the operators' error probability at different phases of action (identification, decision-making and implementation) and enables to properly account for different factors that impact human performance (interface, alarm, indications, procedures, training, stress, time, etc.).

The human reliability analysis (HRA) presented in the paper allowed to refine the Ignalina NPP PSA model. Application of such methodology is possible in areas where operators play an important role in ensuring safety, e. g., in a new NPP or in the present and future industry of oil, gas, electricity, and transport.

Key words: operators' actions, human reliability analysis, probabilistic safety assessment, nuclear power plant

1. INTRODUCTION

An integral part of a qualitatively performed probabilistic safety analysis (PSA) [1, 2] for a nuclear power plant (NPP) is a human reliability analysis (HRA) which identifies the possible human actions that could affect the safety of a facility. Human actions influence safety in different ways, such as making safety equipment unavailable due to errors during repair or maintenance, or initiating an abnormal event, or making errors during accident mitigation. The analyst's task is to identify human actions vital for the plant safety, adequately evaluate the factors that have the highest impact on the performance of plant operators, to evaluate human error probability (HEP) for each action and include the actions in the PSA model. Like the possibility of equipment failure, the possibility of human error is characterized by probability. However, human errors are rarer than equipment failures, and humans cannot be simply "tested" to get such statistics. In addition, the reasons for human errors and the factors that influence them are more numerous and differ from those for equipment failures. There are many HRA methods used for different stages of analysis. HRA specialists can apply different methods taking into account their advantages and disadvantages, also combining and developing their own method for a specific study. Despite the methods used, every HRA should have at least the following attributes:

- important performance-shaping factors (PSF) that affect human actions, are expressed clearly enough to understand and document;
- the dependencies are identified and accounted for;
- probabilities of human errors are consistent internally and with the plant experience and other evidence;

- uncertainties are identified, quantified and displayed;
- the whole analysis is well documented.

The PSA model of the Ignalina Nuclear Power Plant (INPP) was being developed and updated for more than ten years. Initially, post-initiator human actions in the PSA of the INPP had been modelled using a simplified time-dependence model in which HEP was dependent only on the time window available to perform a corresponding human action. During the further development of the INPP PSA, a need for more accurate HRA was identified. However, no such analysis has ever been performed for the INPP or for other industrial objects in Lithuania.

This paper presents a short overview of the methodology and an analysis performed for post-initiator human actions.

2. HUMAN RELIABILITY ANALYSIS

2.1. Post-initiator actions

Post-initiator human actions are the actions performed by an operator after an initiating event, i. e. when an emergency situation occurs. After the initiating event, the plant operators must take actions for the manual activation, control and alignment of the plant systems that are required to ensure the plant's safety and avoid an accident. These tasks are an integral part of the plant's response to the initiating events; they are well defined and described in plant emergency procedures. In most cases, the plant safety systems are activated automatically after an initiating event, and the operators' role is to align and control these systems. The importance of the operators' actions becomes much higher if the emergency sequence of the events does not correspond to the expected scenario; e. g., the safety system does not start automatically or some safety systems fail, or additional failures occur. In such cases, operators must backup the start-up signals, initiate redundant systems or equipment instead of failed ones and take additional measures to keep the plant safety parameters within safety limits. In practice, operators' actions are affected by many different factors. Most important of them are as follows:

- time;
- stress;
- experience and training;
- availability of written procedures;
- recognition of the event and plant status.

These factors are scenario-specific, i. e. their effect may differ for the same operators' actions performed in different circumstances depending on the initiating event and accident scenario. A valid HRA should account for such differences and provide consistent HEPs.

For incorporation into the PSA, the post-initiator actions can be differentiated into three different types:

- *type 1 – procedural safety actions*. These actions involve success or failure in the subsequent procedures or rules in response to an accident sequence;
- *type 2 – aggravating actions / errors*. These actions are a special set of commission errors that significantly aggravate the accident progression. Such errors occur when the operator's mental image of the plant differs from the actual state. In this case, the wrong diagnosis of the situation leads to performing the right actions to a wrong event. Another form of such errors occurs when the right diagnosis is made, but a non-optimal strategy for dealing with the event is chosen;
- *type 3 – improvising recovery / repair actions*. These actions are non-standard and consist in the recovery of unavailable equipment or the use of non-standard procedures.

2.2. Main stages of analysis

The main stages of human reliability analysis are:

1. To identify the operators' actions. For instance, the key actions important for the INPP safety are identified in the PSA model. These actions are verified using the INPP operating and emergency procedures, interviews with operators.

2. For each action to identify:

a) success criteria – what action shall be performed; how many operators are required to perform the action; what level of operator's skill or training is required; where the action must be performed;
b) boundary conditions – what is the initiating event, what equipment failures or human errors preceded the action;
c) timing – when the action shall be performed, what indications are a signal to start the action; how much time is available to the operator to perform the action; how much time is required to perform the action;
d) dependencies – what relations between the actions affect the same function; what are the possible options and which option has a priority.

Different sources of information are analysed at this stage: the plant operating and emergency procedures, safety reports, the PSA model and documentation, the checklist of interviews with operators, etc.

3. To perform a quantitative analysis of the operator's errors.

Approximately 75% of human reliability analysis work takes to identify actions and include them into the PSA model, and 25% takes to estimate human error probabilities.

In general, PSA uses event tree method for modelling the possible accident scenarios and fault tree method for modelling the plant safety systems' failures related to different scenarios. Human actions and the corresponding errors are included in both event trees and fault trees.

2.3. Overview of HRA methods

There are a number of HRA methods, which have their own advantages and disadvantages, differ in the levels of details and highlight different aspects of human actions. The most commonly used HRA methods are THERP (Technique for Human Error Rate Prediction) and ASEP (Accident Sequence Evaluation Procedure) described below.

2.3.1. Technique for human error rate prediction (THERP)

As described in [3], THERP is a method for identifying, modelling, and quantifying human failure events (HFEs) in a PSA. It is a reasonably complete approach to HRA and has probably been used more often than any other HRA technique. Beside its application to NPPs, THERP has recently been used in the maritime affairs and in other industries. It has also been applied worldwide since its publication, and a sizeable knowledge base now exists on THERP application.

However, with respect to modelling, THERP does not provide an explicit guidance on how to model a human failure event in a PSA. Nonetheless, its qualitative guidance can be useful in performing it. The THERP decomposes non-diagnosis HFEs into lower-level errors and identifies important performance shaping factors (PSFs) via task analysis (one of the principal features of a THERP analysis). This decomposition is graphically represented as HRA event trees. THERP also contains a database of nominal HEPs, a few of which have some basis in empirical evidence, but also involves adaptation / extrapolation by the authors to fit the NPP domain. The rest of the database represents an expert judgment of the THERP authors, which is based on knowledge and data gathered over decades of research and practice based on human–machine interactions in industrial and military facilities, including NPPs.

The resource-intensive nature of THERP limits its application in full-scale PRAs to the extent intended by the method (e. g., to perform task analyses, to use HRA event trees), but it can be supplemented with a screening procedure (e. g., ASEP, see below) to quantify the majority of HFEs in the analysis. The full THERP task analysis can then be focused for a subset of the HFEs, which represent the dominant contributors to the risk [3].

2.3.2. The accident sequence evaluation program HRA procedure (ASEP)

As described in [4], ASEP is a less-resource-intensive HRA method. In contrast to THERP, ASEP is intended to be able to be implemented by systems analysts who are not HRA specialists. Given the "short-cuts" in the method (compared to THERP), the ASEP quantification approach is purposely intended to provide conservative estimates. ASEP addresses the quantification of both pre-accident and post-accident HFEs and provides a specific guidance for deriving both the screening and the nominal values for both types of HFEs. It is based on THERP, but purposely simplifies parts of THERP, such as the model for dependency. In addition, ASEP is almost entirely self-contained; the users need not be familiar with THERP and are not required to use any of the THERP models or data.

However, ASEP does not address most of activities related to the HRA process, such as identification of HFEs, and does not provide a detailed guidance on how to model the HFEs. Thus, in using ASEP, it is assumed that the HFEs have already been identified and modelled and only the quantification of the associated HEPs is required [5].

2.4. Quantification using ASEP and THERP methods

A combination of the THERP and ASEP methods was used for quantification of human error probabilities in a PSA study for the Ignalina NPP.

Each post-initiator dynamic action includes two stages:

1. Cognitive stage, i. e. an effort to notice and recognize the situation that requires operator's intervention, to think it over and to take a decision.

2. Implementation stage, when the action itself is performed.

Each stage is affected by a different set of factors. The cognitive, or decision-making, stage is affected mostly by availability of information about the plant status (e. g., alarms, indications), availability and quality of written procedures and the operator's skills and training and his ability to recognize the situation and make a decision. The implementation stage is mostly affected by a physical possibility to perform the action, i. e. the number of personnel, the number of operations, access to the equipment, procedures, equipment labelling. Both stages are affected by the stress level of operators and the time available. Therefore, HEPs at each stage are estimated separately. Knowing these HEPs at each stage, the total probability of human error P_e is simply calculated as

$$P_e = P_d + P_a, \tag{1}$$

where P_d is the probability of failure to correctly diagnose the required response and make a correct decision, and P_a is the probability to perform the required action.

The main steps of P_d and P_a evaluation are related to the timing analysis. Initially, for each action it is necessary to identify the maximal time window T_m available to perform the action, after which it is actually too late to take any actions. Examples of time window are the time that the reactor can survive without cooling before its overheating, or time a pump can run without cooling the bearings until its failure. The time interval T_m can be evaluated using results of deterministic analysis. Such time window includes the time interval T_d to diagnose and make a decision, as well as the time interval T_a to perform an action (Fig. 1).

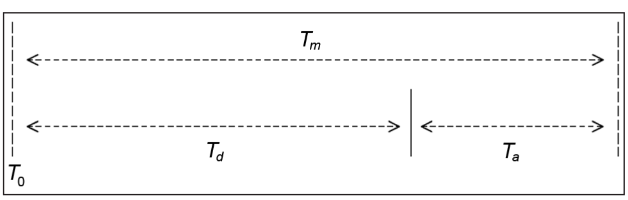

Fig. 1. Time to diagnose and perform an action

T_0 – time moment of annuncing (or receiving some other compelling signal) of an abnormal event; T_m – estimated maximum allowable time interval to complete the diagnosis and the required post-diagnosis actions to satisfy PSA success criteria ("available time window"); T_d – the estimated allowable time interval for a correct diagnosis, that permits sufficient time to accomplish post-diagnosis actions before T_m; T_a – estimated time interval for performing the required actions after the correct diagnosis.

Knowing T_m, the time interval T_a, which is sufficient to perform the action after a decision has been made, is estimated next. This time interval T_a could be very short, e. g., the time interval necessary for pressing the button on the control panel, or much longer, e. g., the time needed to manually open a valve located in another unit. Then, the time interval T_d, which is available to recognize (diagnose) the situation and make a decision on what actions shall be performed is simply calculated as follows:

$$T_d = T_m - T_a. \quad (2)$$

Application of the ASEP method is based on the assumption that the available time interval is the key factor to the cognitive and decision-making stage and that an error in the diagnosis and decision-making means the failure of the entire action. Thus, the nominal P_d probability of diagnosis error depends on time T_d and is evaluated using the ASEP time-dependency curve and tables [4] (Fig. 2). The uncertainty limits of this type HEP are assigned using the same curve and tables.

The ASEP method contains the guidance and tables that allow to adjust the nominal HEP in order to account for different factors such as alarms, stress, procedures, training, etc., and to estimate the probability of human error during several decisions that shall be taken in a compressed time.

Finally, the P_a probability of failure to perform an action is similarly estimated by applying THERP tabulated data [3] depending on action complexity, procedures, skills and training. Such estimation represents a very simplified application of the THERP method.

2.5. Uncertainty estimate for human error probabilities

The nominal value of HEP estimates (P_d and P_a) and the error factor E_f for each HEP are provided in THERP and ASEP tables. The nominal HEP is treated as "the best estimate" and is taken to be a median $p_{0.5}$ (not mean μ) value. It is assumed

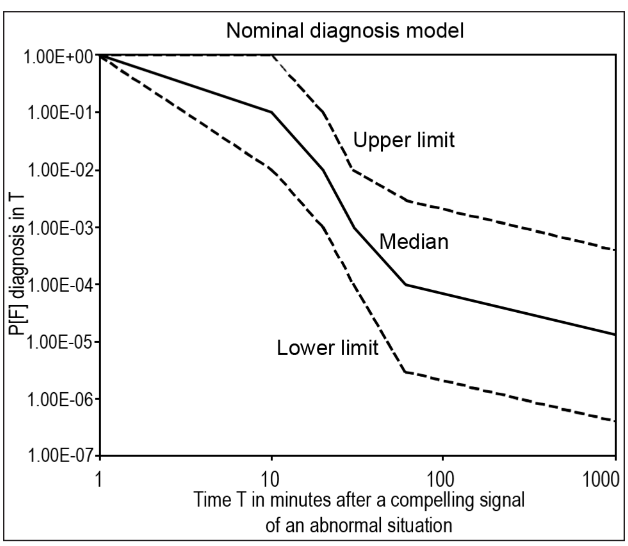

Fig. 2. Error probability during diagnostics and decision-making

that the distribution of HEP estimate is lognormal, and this is taken into account for estimating the mean value and uncertainty measures.

According to statistical theory, the uncertainty of the corresponding HEP estimates may be expressed in percentiles p_i (usually $p_{0.95}$, $p_{0.05}$) or using the so-called error factor of estimation, i. e. E_f, which for lognormal distribution is expressed by the following equations:

$$E_f = p_{0.95} / p_{0.5} = p_{0.5} / p_{0.05} = (p_{0.95} / p_{0.05})^{0.5}. \quad (3)$$

From the above equations, it is possible to derive the formulas for calculating the following percentiles:

$$p_{0.95} = p_{0.5} \cdot E_f;\ p_{0.05} = p_{0.5} / E_f;\ p_{0.5} = (p_{0.95} \cdot p_{0.05})^{0.5}. \quad (4)$$

Thus, knowing the HEP estimate (P_d or P_a) expressed as the median $p_{0.5}$ and having the corresponding error factor E_f, the above formulas trivially enable calculating the percentiles $p_{0.95}$ and $p_{0.05}$ as the uncertainty measures for this estimate.

The final task is to evaluate uncertainty bounds (UCB) for the total human error probability P_e which is a sum of probabilities P_d and P_a. Due to the statistical features of the sum of lognormally distributed values, the procedure of estimating the uncertainty of P_e is not so trivial. Thus, for the practical calculations, a procedure of UCB propagation from the THERP method is adapted.

Based on the lognormal assumption of HEP, both $\ln(P_d)$ and $\ln(P_a)$ values are distributed normally, and therefore the mean values $\mu_N(P_d)$, $\mu_N(P_a)$ and standard deviations $\sigma_N(P_d)$, $\sigma_N(P_a)$ of such normal distributions can be calculated from the known values of $p_{0.5}$ and E_f:

$$\mu_N = \ln(p_{0.5});\ \sigma_N = \ln(E_f^2) / 3.29. \quad (5)$$

The mean μ and variance σ^2 of lognormally distributed values (i. e. $\mu(P_d)$, $\mu(P_a)$ and $\sigma^2(P_d)$, $\sigma^2(P_a)$) can be determined from the mean μ_N and standard deviation σ_N of normally distributed values using the following equations:

$$\mu = \exp(\mu_N + \sigma_N^2 / 2), \quad (6)$$

$$\sigma^2 = \exp(\sigma_N^2 + 2\mu_N) \cdot (\exp(\sigma_N^2) - 1). \quad (7)$$

In general, the THERP method is based on several studies which show that the sum of the lognormal variables can be adequately approximated by the lognormal distribution. This is why the mean and variance of P_e are expressed using the sum of lognormally distributed values:

$$\mu(P_e) = \mu(P_a) + \mu(P_s);\ \sigma^2(P_e) = \sigma^2(P_a) + \sigma^2(P_d). \quad (8)$$

Then, based on the lognormal distribution of P_e, the mean and the standard deviation for a normal distribution of $\ln(P_e)$ values can be calculated using the following equations:

$$\mu_N(P_e) = \ln\left(\mu(P_e) / \sqrt{1 + (\sigma^2(P_e) / \mu^2(P_e))}\right), \tag{9}$$

$$\sigma_N(P_e) = \sqrt{\ln(1 + (\sigma^2(P_e) / \mu^2(P_e)))} \,. \tag{10}$$

Finally, for P_e, the median $p_{0.5}$ and UCB (i. e. $p_{0.05}$ and $p_{0.95}$) are:

$$p_{0.50}(P_e) = \exp(\mu_N(P_e)), \tag{11}$$

$$p_{0.05}(P_e) = \exp(\mu_N(P_e) - 1.645 \cdot \sigma_N(P_e)), \tag{12}$$

$$p_{0.95}(P_e) = \exp(\mu_N(P_e) + 1.645 \cdot \sigma_N(P_e)). \tag{13}$$

2.6. Human reliability analysis for PSA of the INPP

Initially, probabilities of human errors for PSA of the INPP were estimated using a simplified time window model. The total HEP (i. e. P_e) was conservatively assumed to equal 1.0 for actions with the time window less than 10 minutes, $P_e = 0.1$ for the time window between 10 minutes and 1 hour, and $P_e = 0.01$ for the time window more than 1 hour. Such a simplified model was only acceptable for a rough estimate at the initial stage of PSA when less information was available. The main drawback of the simplified approach is that HEPs are the same for different accident scenarios having the same time windows.

Later, when the PSA model and data become more accurate, a more precise analysis and estimation of human errors were required. The need for a new HRA was recognized by the PSA team and recommended by expert missions, and a more detailed HRA was performed within the scope of the INPP PSA update [6] (the so-called "living PSA"). During PSA updating, various HRA methods used worldwide were analysed, the methodology for the PSA of the INPP was developed, and the analysis was carried out both for pre- and post-initiator human actions.

For instance, the following seven post-initiator actions important for the reactor cooling were analysed:

1. Water supply from the emergency core cooling system (ECCS) to the group distribution header (GDH) in case of a loss-of-coolant accident (LOCA).
2. Primary circuit (PC) makeup in case of feedwater loss.
3. Transition to PC water supply from long-term ECCS instead of main feedwater pumps.
4. Closure of valves of the main circulation pumps (MCP) in case of LOCA.
5. Service water from Unit 1 supply to the intermediate cooling circuit #2 (ICC-2) and diesel generators (DG) of Unit 2.
6. Alternative cooling of ECCS pumps in case of the ICC-2 failure.
7. Alternative cooling of auxiliary feedwater pumps (AFWP) in case of the ICC-2 failure.

Each action was analysed in the context of different initiating events and accident scenarios. The number of such scenarios in different groups of initiating events varied from 5 to 18. As an example, an analysis of one of such actions is presented below.

2.6.1. Example of operators' action analysis

Action name: Water supply from ECCS to GDH in case of LOCA.

Description: In case of a loss-of-coolant accident (LOCA), an automatic start-up of the emergency core cooling system (ECCS) with water supply to the group distribution header (GDH) is provided in accordance with ECCS algorithms #1–4 [7]. If the ECCS automatic start-up fails, the operator shall manually initiate the necessary equipment and ensure water supply to the primary circuit (PC).

Functional success criteria:

- plant conditions to start the action: presence of ECCS start-up conditions and absence of ECCS operation indications (pumps work, valves are open, water flows from water pressurized tanks and pumps);
- time window end conditions: reactor core overheat;
- action goal: to prevent reactor core overheat;
- actions to be taken: recognize LOCA; recognize ECCS start-up failure; start water supply from short-term ECCS (pressurized tanks and MFWP); start water supply from long-term ECCS (ECCS and AFW pumps).

Physical success criteria:

- equipment to be initiated: for short-term ECCS – open valves; for long-term ECCS – start the pumps and open valves;
- who performs the action: the leading engineer for unit control (LEUC) under supervision of a deputy shift supervisor;
- where the action is performed: the main control room, workplace of the LEUC;
- procedures to be used: symptom-oriented emergency procedure, accident mitigation procedure.

Time-based success criteria. Time is a scenario-context factor that depends on the initiating events. Therefore, an action has to be analysed for two groups of initiating events – Large LOCA and Medium LOCA. Time window is defined in accordance with the results of a thermal-hydraulic analysis. According to the performed analysis [7], in case of a large LOCA, the reactor core temperature will exceed the acceptability criteria after 600 seconds. For a medium LOCA, this time is 1500 seconds. Therefore, time windows T_m to start short-term ECCS for these groups of initiating events are 10 minutes and 25 minutes, respectively.

The operation of at least one of the three trains of the short-term ECCS adds two minutes to the reactor overheat time. Thus, the time window T_m to initiate the long-term ECCS in case of the short-term ECCS success is 12 minutes for a large LOCA and 27 minutes for a medium LOCA.

Table 1. **Estimates of timing and HEP for the action "Water supply from ECCS to GDH"**

Initiating event	Operator's action	T_m, min	T_a, min	T_d, min	$\mu(P_a)$	$\mu(P_d)$	$\mu(P_e)$	$p_{0.50}(P_e)$
Large LOCA	Start of short-term ECCS	10	1	9	1.57E-01	3.75E-03	1.61E-01	1.30E-01
	Start of long-term ECCS	12	1	11	9.93E-02	3.75E-03	1.03E-01	8.35E-02
Medium LOCA	Start of short-term ECCS	25	1	24	6.43E-03	3.75E-03	1.02E-02	7.77E-03
	Start of long-term ECCS	27	1	26	4.05E-03	3.75E-03	7.80E-03	6.24E-03

This is a good example showing that PSF could be different for the same action under different scenarios. Here, the time window for a medium LOCA is almost twice longer, allowing more time to diagnose, make a decision and implement the required action.

Boundary conditions. Analysis of boundary conditions is the most difficult and important part of the analysis. All factors that have an impact on the operator during the action have to be identified, analysed and documented. For the case of the above example, findings of the analysis of boundary conditions are briefly presented below.

The time required to perform the action is assumed to be one minute. Since during a large and a medium LOCA the plant parameters change rapidly, a high stress level is anticipated. Operators are monitoring the main plant parameters and start taking actions in accordance with their experience and emergency procedures. Operators in the Main Control Room have a good picture of the main parameters and processes. The operation of ECCS is one of the important parameters to follow, and a failure of ECCS is recognized very soon. The recognition and decision-making are affected by competing actions and the priority of actions. At the same time, the operator shall monitor the PC parameters, feedwater flow, pressure in DS, the operation of protections. Also, operators are monitoring the start-up of Accident Localization System (ALS) and Emergency Deaerators Makeup (EDM) pumps. However, monitoring of ECCS start-up is of the highest priority, and it is assumed that the competing actions will not disturb the action to start the ECCS manually.

Quantification. In the previous version of the PSA model, HEP for this action was conservatively assumed to equal 1.0, i. e. that the action was impossible due to a very short time window.

In the current analysis, P_d is estimated using the median ASEP curve (Fig. 2) [4] and, depending on the time window T_d, the probability estimate P_a is calculated applying the THERP [3].

The results of calculation are presented in Table 1.

As mentioned above, initially, in the PSA of the INPP, the operator's action "Manual start of ECCS" was conservatively

Table 2. **Estimates of HEP and its uncertainty for operator's actions**

Scenario-specific action	Mean value	Error factor	Uncertainty measures – percentiles p_i (including median)		
	$\mu(P_e)$	E_f	$i = 0.95$	$i = 0.50$	$i = 0.05$
1. Start of short-term ECCS, Large LOCA	1.61E-01	2.94	3.82E-01	1.30E-01	4.43E-02
2. Start of long-term ECCS, Large LOCA	1.03E-01	2.91	2.43E-01	8.35E-02	2.87E-02
3. Start of short-term ECCS, Medium LOCA	1.02E-02	3.35	2.60E-02	7.77E-03	2.32E-03
4. Start of long-term ECCS, Medium LOCA	7.80E-03	3.00	1.87E-02	6.24E-03	2.08E-03
5. Drum-separator makeup	3.96E-03	2.88	9.28E-03	3.22E-03	1.12E-03
6. Drum-separator makeup after MFWP trip	3.96E-03	2.90	9.32E-03	3.21E-03	1.11E-03
7. Closure of MCP valves	5.78E-03	2.69	1.30E-02	4.83E-03	1.80E-03
8. Service water (SW) supply to ICC-2	4.01E-03	2.89	9.41E-03	3.25E-03	1.12E-03
9. SW supply to DG	1.46E-03	2.81	3.37E-03	1.20E-03	4.27E-04
10. SW supply to ICC-2, Large LOCA	4.33E-02	2.78	9.93E-02	3.57E-02	1.28E-02
11. SW supply to DG, Large LOCA	1.00E+00	–	–	–	–
12. SW supply to ICC-2, Medium LOCA	1.26E-02	3.61	3.35E-02	9.28E-03	2.57E-03
13. SW supply to DG, Medium LOCA	1.26E-01	2.97	3.01E-01	1.01E-01	3.41E-02
14. SW supply to ICC-2, Transients	8.44E-03	3.10	2.07E-02	6.66E-03	2.15E-03
15. SW supply to DG, Transients	6.09E-03	4.01	1.71E-02	4.27E-03	1.07E-03
16. ECCS pumps – alternative cooling	1.50E-03	3.00	3.61E-03	1.20E-03	4.01E-04
17. ECCS pumps – alternative cooling, Large LOCA	1.45E-02	4.56	4.31E-02	9.44E-03	2.07E-03
18. ECCS pumps – alternative cooling, Medium LOCA	7.17E-03	4.14	2.04E-02	4.93E-03	1.19E-03
19. ECCS pumps – alternative cooling, Transients	5.86E-03	3.97	1.64E-02	4.13E-03	1.04E-03
20. AFW pumps – alternative cooling	1.50E-03	3.00	3.61E-03	1.20E-03	4.01E-04
21. AFW pumps – alternative cooling, Large LOCA	1.37E-02	2.77	3.15E-02	1.13E-02	4.09E-03
22. AFW pumps alternative – cooling, Medium LOCA	8.48E-03	4.27	2.45E-02	5.75E-03	1.35E-03
23. AFW pumps – alternative cooling, Transients	5.86E-03	3.97	1.64E-02	4.13E-03	1.04E-03

assumed impossible. However, during the systematic analysis, this action was analysed against different initiating events and, as a result, four scenario-specific actions were identified and the corresponding HEP values were calculated.

For each action, it is necessary to perform specific assessments. As an example, for performing the first considered action, the nominal HEP is expressed as a median value of the lognormal distribution $p_{0.5}(P_a)$ = 1.26E-01 with E_f = 3, whereas for decisions in relation to the first action, the nominal HEP is a median value of the lognormal distribution $p_{0.5}(P_d)$ = 3E-03 with E_f = 3. Thus, according to equation the corresponding HEP mean estimates for the first considered action are $\mu(P_a)$ = 1.57E-01 and $\mu(P_d)$ = 3.75E-03. The calculated mean estimates $\mu(P_e)$ (see equations (1)–(8) and Table 1) of the total HEP are more realistic as compared with the initial conservative assumption of P_e = 1.0 for these actions.

2.6.2. Overview of HRA for PSA of the INPP

The example provided in the previous subsection shows how during a HRA different factors were identified for the actions under different scenarios. The result of a single action analysis is expressed as four different scenario-specific actions with their own probabilities (see Table 1). After the analysis of seven actions, in total 23 scenario-specific actions were identified (Table 2). The estimates of HEP in PSA are presented as mean values (calculated using equation (8). The corresponding median estimates and uncertainty measures are calculated using equations (11)–(13).

The above analysis was focused on reactor cooling actions, but due to the lack of resources it does not cover all post-initiator actions. The following areas were not considered: reactor shutdown; recovery of systems, manual start-up of redundant equipment, manual opening of failed valves. For the analysed events, the values that are more reasonable were obtained. The use of such values makes the PSA model more realistic and consistent and allows using PSA for practical applications.

Another important output of such HRA is a thoroughly documented process of analysis, similar to the one described in the previous subsection. The use of such records enables to identify the main factors that affect operators' performance during abnormal events and accident mitigation. Such findings can be used by an NPP to improve the reliability of operators' actions.

3. CONCLUSIONS

A combination of the ASEP (Accident Sequence Evaluating Procedure) and THERP (Technique for Human Error Rate Prediction) methods was successfully applied for human reliability analysis. This methodology enables to assess the operators' error probability at different phases of action: identification, decision-making and implementation, also allowing to properly account for different factors that impact the human actions that are to be performed by NPP operators during postulated accident sequence.

The presented results and calculation description is the part of performed probabilistic safety analysis and the first experience of systematic and comprehensive HRA for NPP in Lithuania. A similar approach and methodology as used in HRA for PSA of INPP could be applied in other areas where operators play an important role in ensuring safety, e. g., in a new NPP or the present and future industry of oil, gas, electricity, and transport.

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ŽMOGAUS PATIKIMUMO ANALIZĖ ATOMINĖS ELEKTRINĖS TIKIMYBINIAM SAUGOS VERTINIMUI

Santrauka

Nepaisant to, kad atominės elektrinės (AE) saugos sistemos yra labai patikimos, žmogaus veiksmams vis dar tenka svarbus vaidmuo užtikrinant AE saugą. Todėl žmogaus patikimumo įvertinimas yra svarbus vykdant visapusišką tikimybinį saugos vertinimą (TSV) bei rizikos analizę.

Tikimybiniame saugos vertinime, kuris apima saugos sistemų sąveikos avarinės situacijos metu modeliavimą, žmonių klaidos yra modeliuojamos kartu su įrangos gedimais. Žmonių klaidų modeliavimo ir jų tikimybių įvertinimo metodai skiriasi nuo metodų, taikomų techninės įrangos gedimams modeliuoti bei jų tikimybėms įvertinti.

Šiame straipsnyje aprašomas veiksmų, kuriuos turi atlikti Ignalinos AE operatoriai avarinės situacijos metu, modeliavimas.

Toks modeliavimas buvo pritaikytas Ignalinos AE TSV. Buvo panaudotas ASEP (angl. *Accident Sequence Evaluating Procedure*) ir THERP (angl. *Technique for Human Error Rate Prediction*) metodų derinys. Tokia metodika leidžia įvertinti operatorių klaidų tikimybę skirtingose veiksmo atlikimo stadijose: identifikavimo, sprendimo priėmimo ir įgyvendinimo, be to, leidžia teisingai įvertinti įvairius žmogaus veiksmus, turinčius įtakos faktorius, veiksmų vykdymą sąlygojančiai sąsajai, avarinė signalizacijai, prietaisų duomenims, procedūroms, apmokymams, stresui, laikui ir pan.

Straipsnyje pateiktas žmogaus patikimumo įvertinimas leidžia patikslinti Ignalinos AE TSV modelį. Ši metodika gali būti taikoma ir srityse, kuriose operatorius atlieka svarbų vaidmenį užtikrinant saugą, pavyzdžiui, naujojoje atominėje elektrinėje, naftos, dujų, elektros pramonėje ir transporto sektoriuje.

Raktažodžiai: operatoriaus veiksmai, žmogaus patikimumo analizė, tikimybinis saugos vertinimas, atominė elektrinė

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АНАЛИЗ ЧЕЛОВЕЧЕСКОЙ НАДЕЖНОСТИ ДЛЯ ВЕРОЯТНОСТНОЙ ОЦЕНКИ БЕЗОПАСНОСТИ АТОМНОЙ ЭЛЕКТРОСТАНЦИИ

Резюме

Несмотря на то, что системы безопасности атомной электростанции (АЭС) имеют высокую надежность, действия человека все еще играют важную роль в безопасности АЭС. По этой причине оценка надежности человека важна для всесторонней вероятностной оценки и анализа безопасности (ВАБ) и анализа риска.

В ВАБ, который включает моделирование взаимодействия систем безопасности во время аварийной ситуации, ошибки человека моделируются вместе с отказами оборудования. Методы моделирования ошибок персонала и оценки вероятностей таких ошибок отличаются от методов, применяемых для моделирования и оценки вероятностей отказов оборудования.

В настоящей статье описывается моделирование действий, которые должны выполняться операторами Игналинской АЭС во время аварийной ситуации. Данное моделирование использовалось во ВАБ Игналинской АЭС. Была применена комбинация методов ASEP (англ. *Accident Sequence Evaluating Procedure*) и THERP (англ. *Technique for Human Error Rate Prediction*). Такая методология позволяет оценить вероятность ошибки операторов на разных стадиях действия: идентификация, принятие решения и выполнение, а также позволяет правильно учесть различные факторы, влияющие на действия человека: интерфейс, аварийная сигнализация, показания приборов, процедуры, подготовка, стресс, время и т. д.

Анализ надежности человека, представленный в данной работе, позволяет уточнить модель ВАБ Игналинской АЭС. Также применение данной методологии возможно в областях, где оператор играет важную роль в обеспечении безопасности, например, на новой атомной электростанции, в существующей и будущей нефтяной, газовой, электрической промышленности и транспортном секторе.

Ключевые слова: действия оператора, анализ надежности человека, вероятностная оценка безопасности, атомная электростанция