UPGRADE OF THE SPENT FUEL POOL PSA FOR NPP PAKS MOTIVATED BY THE CHANGE IN EMERGENCY OPERATING PROCEDURES

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Level 1 PSA models of the spent fuel pool (SFP) are available for the Paks Nuclear Power Plant in Hungary since 2002. Prior to implementing symptom-oriented emergency operating procedures (EOPs) for accident sequences related to the SFP in 2011, post-initiator operator actions (so-called Type C human interactions) had been considered in PSA on the basis of event-based EOPs. Due to the change in the procedures, the whole area of human reliability analysis (HRA) for Type C actions had to be reconsidered and renewed. PSA model development and re-quantification in view of EOP improvement were completed in 2015. Motivated by the implementation of symptom-oriented EOP for SFP accidents, a thorough review of the accident sequence models was performed in 2015 to identify the necessary model modifications. These modifications and several other model upgrades reflecting findings from deterministic safety analyses, PSA model upgrades for the reactor, detailed system analyses, as well as the update of input reliability data were subsequently performed. This reassessment led to a significant change in the results of the SFP PSA with respect to the annual fuel damage probability (FDP) as well as to the contributions of the different plant operational states (POSs) and initiating events to the overall risk. This paper presents an overview of the SFP PSA upgrade for the Paks NPP performed in response to the implementation of the new EOPs. Important methodological aspects are summarized. Model upgrades made within and beyond HRA relevance are highlighted. Key analysis findings are discussed as well.

I. BACKGROUND

A level 1 PSA model of the SFP is available for the Paks Nuclear Power Plant in Hungary since 2002. Initially, the assessment was limited to internal events and internal hazards, and unit 1 was selected out of the 4 Paks units as a reference unit for the analysis. In 2006 the assessment of the SFP PSA was extended to all units. Consequently, unit specific, stand-alone level 1 PSA models and results are now available for the four SFPs of the VVER-440/213 type units of the Paks NPP. Originally, 4 different POSs were defined for the SFP representing a typical annual fuel cycle. These POSs are analyzed separately by calculating the fuel damage risk for each of them. Several model updates have been performed since 2002 to reduce uncertainties, remove some conservatism assumed in the initial assessment and reflect safety impacts from plant modifications. Until 2011, post-initiator operator actions (so-called Type C human interactions) had been considered on the basis of event-based EOPs relevant to the SFP. The general philosophy of this type of EOPs is to identify precisely the transient prior to any operator interventions. Human failure events were modelled by a holistic decision tree (HDT) reflecting the role of underlying performance influences in failure likelihood in an environment when the event-based approach was in use. In 2011 new, stand-alone symptom-oriented EOPs were elaborated for low power and shut down conditions (hereafter: L-SOEOPs) to replace the earlier event-based procedures. A separate EOP of the L-SOEOPs deals exclusively with the operator actions that should be carried out after the occurrence of a SFP accident (hereafter: SFP-SOEOP). When symptom oriented EOPs are in use, operator actions are taken on the basis of different measurable parameter values or changes of thereof without requiring the operator to identify unambiguously the transient itself. Implementation of the SFP-SOEOP affects the definition of post-initiator human failure events in HRA (identification) because it involves substantial changes in the strategy of emergency responses and in the actual instructions in the procedures. Also, the probability of human failure events (quantification) may alter considerably due to changes in the ways of procedure usage and in procedure quality. Consequently, the whole area of HRA for Type C actions had to be reconsidered and renewed due to the change in EOPs.

II. OBJECTIVES

In compliance with the abovementioned effects of the EOP changes, PSA model development and re-quantification in view of EOP improvement has been performed for the SFP PSA of the Paks NPP. Among others, the objectives of the assessment were to quantify the risk implications of this procedural change, to assess the level of risk in each POS defined for the SFP PSA and to identify the main risk contributors. Another important objective was to identify and also to quantify the effects of further possible developments or refinements of the L-SOEOP as safety enhancement proposals for the nuclear facility. Besides modifications induced by EOP changes, additional analysis tasks were also necessary to thoroughly review the accident sequence models to identify the need for other model upgrades and perform these upgrades reflecting findings from deterministic safety analyses, PSA model upgrades for the reactor, detailed system analyses, as well as the update of input reliability data.

III. METHODOLOGICAL ASPECTS OF MODELLING SYMPTOM-ORIENTED EOP IMPLEMENTATION

Use was made of the methodology elaborated earlier in the reactor (full power) PSA of the Paks plant for the purposes of the PSA model development and re-quantification to reflect the effects of EOP improvement for the SFP. The methodology on human failure event identification and quantification relevant to the reactor PSA was found applicable to the SFP PSA too. The upgrade of the PSA included review and modification of the accident sequence models in accordance with the new procedural requirements for emergency operations, re-definition of human failure events for Type C errors in the PSA model, quantification of operator reliability for the re-defined failure events, and re-quantification of accident sequences. Hereby important methodological aspects are summarized by giving an overview of the major analysis steps on modelling operator errors while using symptom oriented EOPs.

III.A. Changes in the Definition of Human Failure Events

The same basic modelling principle was applied in the SFP PSA model upgrade as in the original SFP PSA to identify human failure events: operator interventions, as an action or series of actions needed to ensure the fulfilment of a safety function, were defined during event sequence delineation. The level of decomposition for a PSA event depends on the contents of the procedures. So the interventions were mostly described at system or subsystem level unless the operation of multiple systems/pieces of equipment was necessary in a common task. A human failure event was described as a failure to bring an intervention to success. Errors of commission were taken into account to the extent they contributed to the failure of a required action but aggravating actions were not modelled.

Credit was given to proceduralized actions only, with the exception of the following two actions: (a) in case of loss of SFP heat removal system as well as loss of off-site power (LOOP) operators may identify the temperature increase in the SFP prior to 60 °C and (re)start the corresponding heat removal train, (b) operators has sufficient time to perform corrective actions in case of failure to enter or erroneously terminate the application of the SFP-SOEOP. Putting emphasis on proceduralized actions mostly is a significant deviation from the original PSA model, where several such actions were also considered that were not described in the procedures explicitly but were covered by training/knowledge. The decision to model proceduralized actions almost exclusively reflects the shift in the way control room crews are supposed to work in an emergency situation.

Descriptions were developed for each operator intervention to show and maintain the relationship with the corresponding EOP entries for the purpose of subsequent quantification. 7 operator interventions were defined for the PSA model, all of them covered by the SFP-SOEOP.

III.B. Optimal Pathways and Accident Sequence Models

First we reviewed the initiating events in all POSs one by one. Based on the available thermohydraulic analyses, plant documents on automatic plant responses to transients and the original PSA event trees, an optimal (or nominal) pathway of emergency responses was determined for each initiating event (more specifically for each sequence) as required by the SFP-SOEOP. The nominal pathway for an initiating event (e.g. loss of SFP heat removal system) changes in some POSs. Extracts were made from the EOP indicating all the procedural steps of an optimal pathway to help the analysts follow the operator responses determined by the EOP. The analysis was focused primarily on the possible deviations from the optimal pathway. All possible deviations were identified by the associated steps in the procedures. These procedural steps were highlighted in the extracts and were further analyzed.

The accident sequence models in the level 1 SFP PSA for Paks were also reviewed to identify the changes that were seen necessary on the basis of the new EOP. Although automatic plant responses did not change due to the introduction of new procedures in itself, substantial changes were made in the accident sequence models (including event trees, fault trees and basic event definitions) to adequately represent the modified emergency response strategies in view of the new procedural requirements. The event trees were updated in the PSA model, and the description of thereof was also refreshed in the documentation.

III.C. Simulator Observations

In the early and mid-nineties numerous observations were made at the training simulator center of the Paks NPP to support HRA in the plant PSA (Ref. 1). Use was made of the data and insights obtained from about 200 PSA-oriented simulator exercises to develop a HDT based model of operator reliability (Ref. 1). In this model human error probabilities are determined as a function of relevant and ranked influences on performance (e.g. task complexity, training and experience, time window, etc.) so that the probability values are decreasing progressively as multiple performance influences change unfavorably and vice versa. Details are given in Ref. 2.

The HDT model developed by incorporating simulator insights from the Paks plant is specific in the sense that it reflects the characteristics of control room crew operation and the role of underlying performance influences in an environment when the earlier, event-based EOPs were in use. Symptom-oriented procedures have changed the strategy towards emergency responses and it has, at the same time, resulted in significant changes in the rules and actual methods of teamwork for the Paks control room crews. Thus a decision was made to perform new simulator observations with the new EOPs being in use for full power operation (Ref. 3). The observations were made more than a year after the official introduction (2003) of the symptom-oriented procedures and after more than two re-fresher training periods on the simulator for each crew to reduce biases due to the effects of transition from the old to the new strategies of emergency operation for full power operation.

Using an initial advice from PSA and HRA analysts, four scenarios were designed for the purposes of the simulator studies as a result of collaboration between PSA experts and training instructors:

- Medium loss of coolant accident with no high-pressure emergency cooling pumps,
- Main steam line break with loss of feedwater supply due to adverse effects of high energy line break,
- Multiple ruptures of steam generator tubes with stuck open steam generator safety relief valve,
- Loss of normal power supply from safety buses with degraded diesel generators.

All the 24 control room crews were tested for each scenario. So the observations covered 96 exercises providing a substantial database for use in support of HRA. The observer team was made up of 6 members on the average, 2 from the developers of the Paks PSA/HRA models, 2 from the training department and 2 from the plant PSA team. The observer team adopted an approach of consensus. The individual observations were compared and discussed after each exercise and a summary data sheet was completed as a final resolution on the observables. Open questions were left for subsequent review based on the electronic data records. One observer used an electronic EOP-tracking system (a computerized version of the procedures) to produce a copy of the pathways the operators took during their responses.

III.D. Simulator Data Analysis and Quantification for HRA

Analysis of simulator data can support, among others, the objectives of HRA. As given in Figure 1, a simple high-level breakdown was used to address the major failure classes related to a human failure event when using the new EOP. Failure to reach the appropriate EOP entry can be considered as a decision error. Various lower level decisions can contribute to this failure event. Failure to perform the required action is basically an execution error made on the condition that the diagnosis was correct, i.e. the right EOP entry was identified and used during response.

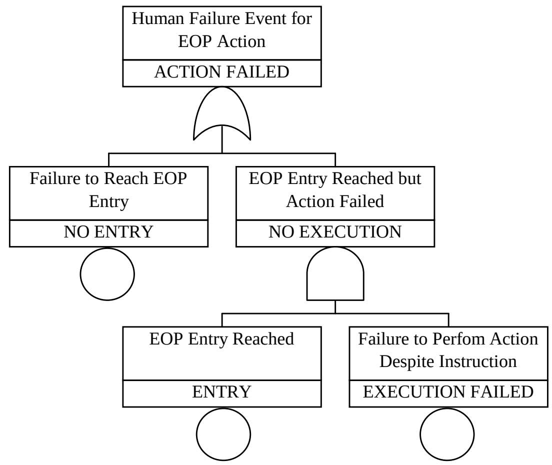

Fig. 1. Failure of an EOP Action

The data from simulator observations could be applied in support of modelling and quantifying the diagnosis part of a failure event. All the procedure entries in the optimal pathways for the four types of observed scenarios were treated as low-level decision points. These decision points were categorized on the basis of 3 factors each having 3 quality attributes as follows:

- Procedure quality
  This factor was used to describe how well or otherwise the given procedure entry was formulated in terms of layout and wording to support decision-making (1 – poor, 2 – fair, 3 – good).
- Quality of human-machine interface (HMI)
  This factor was meant to describe quality features of the control room environment concerning diagnosis required in relation to the given procedure entry (1 – poor, 2 – fair, 3 – good).
- Other effects on performance
  These were introduced as a surrogate to characterize whether there were additional situational features that hindered decision (1 – yes, strong effect, 2 – yes, moderate effect, 3 – no).

Table I summarizes the results of the quantification process for the theoretically possible 27 combinations of the quality attributes used to characterize the above 3 factors for a decision point. For more details on the assessment methodology of the simulator observations and the quantification process of the decision point level relative failure frequencies see Refs. 3 and 4.

The above process yielded information from the simulator observations that could be integrated into the framework depicted in Figure 1 for quantifying human error probabilities. For this a lower level decomposition was developed for the human failure events in the PSA model. This was achieved by identifying the potential pathways and the associated decision points that the operators can take so that they finally fail to successfully perform the required intervention as assumed in the PSA. The decomposition is represented by a fault tree for each human failure event with basic events characterizing the decision point level successes and failures in the various pathways. The fault trees include both decision errors (as the diagnosis part of a failure event) and execution errors. Decision errors cover: decision point level errors, skipping of a procedure entry, procedure selection errors, errors in using warnings and fold-out pages, and failures in built-in recovery processes (an effect that could not be observed extensively on the simulator) in the EOPs. The advantage of this approach is that it directly shows dependencies between actions belonging to a pathway.

With respect to the execution part of a human failure event a modified version of the earlier HDT model was developed. Based on the fact that manual actions and the associated error mechanisms are not related closely to the strategy of emergency operation (as manifested by the new EOPs), a simplified HDT model was considered appropriate. Details about the simplification process are beyond the scope of this paper.

In summary, a fault tree representation has been developed for the human failure events related to the use of the new SOEOPs in the updated PSA model:

- Most basic events representing diagnosis (decision related) failures have been quantified using relative frequencies from simulator observations.
- Expert judgment has also been used to estimate probabilities for certain basic events related to diagnosis in the fault trees (e.g. recovery potential).

- A simplified version of the earlier HDT model has been developed and used to assess basic event probabilities for execution failures in the fault trees.
- The error probability for a human failure event can be obtained by solving the associated fault tree.

Although the original objective of the methodology elaborated earlier was to model mitigation of accident sequences for the reactor at power operation, the methodology was found appropriate for analyzing SFP transients in the different POSs, thanks to the similarities in the structure of symptom-oriented EOPs for the SFP and for the reactor. On the other hand the same control room staff is responsible for coping with reactor and SFP accidents, hence we applied the same estimates for decision point level relative failure frequencies for human error quantification.

TABLE I. Estimates for Decision Point Level Relative Failure Frequencies

Entry Level Quality Attributes for			Relative Frequency of Decision Error	
Procedures	HMI	Other, hindering	Observed	Approximated
1	1	1		1.00E+00
		2		6.30E-01
		3		2.75E-01
	2	1		9.43E-01
		2		3.24E-01
		3	1.42E-01	1.42E-01
	3	1		5.35E-01
		2		1.84E-01
		3	1.25E-01	8.04E-02
2	1	1		8.31E-01
		2		2.86E-01
		3		1.25E-01
	2	1	4.17E-01	4.27E-01
		2		1.47E-01
		3	6.62E-02	6.42E-02
	3	1	2.50E-01	2.43E-01
		2	8.33E-02	8.33E-02
		3	3.57E-02	3.64E-02
3	1	1		2.77E-01
		2		9.53E-02
		3	4.17E-02	4.17E-02
	2	1		1.43E-01
		2		4.90E-02
		3	2.14E-02	2.14E-02
	3	1		8.10E-02
		2		2.78E-02
		3	7.54E-03	1.00E-02

IV. MODEL UPGRADES BEYOND HRA RELEVANCE

Besides modifications induced by EOP changes, the review of the accident sequence models revealed the need for several other model upgrades. The most important group of these changes is related to refinements in the definition of adequate emergency responses to SFP transients but other kinds of modifications were also seen necessary. Some of the important model upgrades made beyond strictly HRA relevance are presented in the following.

IV.A. Redefinition of SFP POSs

Originally, 4 different POSs were defined for the SFP representing a typical annual fuel cycle. The different pool states addressed in the SFP-SOEOP, as well as the need for a refined POS definition initiated by the review, led to the redefinition of the POSs. As a result, 6 POSs were distinguished for the SFP in the PSA. On one hand, the POS definitions and characteristics were based on the different pool states addressed in the relevant EOPs. On the other hand, the amount of the water in the SFP, as well as the residual heat of the fuel assemblies were also taken into consideration in the definition and characterization of the POSs. The 6 POSs were described by the number of fuel assemblies and the associated heat decay heat in the SFP, volume of water above the fuel rods, volume of water in the SFP and average annual POS duration. Table II summarizes the characteristics of the newly defined POSs.

TABLE II. Characteristics of redefined POSs for the SFP

No.	Description	Type of fuel outage	Heat production (kW)	Volume of water above the fuel rods (m^3)	Total volume of water (m^3)	Average annual POS duration (h)
01	Entire core is in the SFP	Long	3600	156	293	160,4
02	Loading and unloading	Long	2300	356	493	52,2
03	Refueling	Short	900	441	493	103,1
04	Fuel transport by means of a C30 container	any	600	306	358	552,0
05	Conditioning	any	600	94	146	4839,2
06			400	94	146	3053,1
					Total	8760,0

IV.B. Changes in the Scope of SFP Initiating Events

LOOP was initially screened out from the internal events PSA for the SFP, since full credit was given to successful recovery from LOOP within 24 hours, and no fuel damage occurs in the SFP in any POS in this time frame. A recent re-analysis of risk from a LOOP event concluded that the earlier exponential approximation of time to recovery had resulted in an optimistic estimation of recovery potential for longer time windows, and LOOP recovery could not be fully credited within 24 hours if a more realistic approximation was applied. Consequently, LOOP had to be analyzed in detail. With this modification the following events were analyzed as internal initiating events in each POS:

- loss of SFP heat removal system (taking into account the failure of both trains of the normal heat removal system, i.e. the heat removal train in operation as well as the standby heat removal train; recovery from loss of the SFP heat removal system and water supply from alternative water sources were taken into consideration as mitigation actions),
- loss of coolant accidents (separate assessments were performed for isolable and for non-isolable pipe sections of the normal heat removal system for the SFP; recovery of pipe ruptures and isolation if applicable as well as water supply to extend time to recovery were taken into consideration in modelling mitigating actions),
- loss of off-site power – LOOP (this event is identical to the corresponding event analyzed in the reactor PSA, i.e. loss of power supply to all the three 6 kV safety buses from the grid).

Concerning internal hazards, detailed probabilistic modeling was performed for fire and flooding events, while other internal hazards had previously been screened out from detailed analysis.

IV.C. Update of Input Reliability Data

The results of a recent comprehensive update of component reliability data in the reactor PSA were incorporated into the PSA for the SFP too. On one hand updated component reliability data from the reactor PSA were directly applied to the SFP PSA for those systems that are part of the event logic model for both the reactor as well as the SFP. On the other hand the reliability data for those safety systems that are only considered in the SFP PSA were reviewed identically to the data update performed in the reactor PSA. The loss of SFP heat removal system is induced by the loss of the operating as well as the standby heat removal train simultaneously. Since the failure of any train can be induced by several component failures, the loss of SFP heat removal system initiating event was modelled by a fault tree as opposed to a single basic event in the PSA. The data update has yielded considerably increased failure rates for the SFP heat removal pumps including both the start-up failure rate and the failure rate during operation. As a result, the frequency of the loss of SFP heat removal system has increased to ~6E-02/a from the earlier value of ~4E-03/a.

Credit is given in the SFP PSA to recover the loss of SFP heat removal system as well as a pipe rupture in the heat removal system. The estimation applied to time to recovery was re-visited and re-considered, and 24 hours was found a more credible assessment of mean time to recovery of pipe ruptures and recovery from a fire-induced loss of the SFP heat removal system than 12 hours used previously in PSA. The review left the mean time to recovery from loss of the SFP system due to internal failures unchanged.

V. FINDINGS

PSA model development and re-quantification to show the effects of EOP improvement were completed by the end of 2015. Hereby we summarize the quantified fuel damage risk for the relevant POSs using the upgraded PSA model for the SFP of unit 1.

Based on the results of risk quantification using the upgraded PSA model, the point estimate of the cumulative annual FDP for all the POSs of 1 to 6 is $4.36 \cdot 10^{-6}$. This figure includes the contribution of all internal events as well as internal fires and internal flooding, but it excludes external hazards. The results reveal, that the FDP for the SFP is favorably moderate, the difference among the units is insignificant. The overall FDP after the PSA upgrade is quite comparable to the figure obtained prior to the reassessment, however, the contribution of the different initiating events has changed considerably. As mentioned earlier, not only the implementation of the new EOPs causes the difference in the values of FDP and the risk contribution of each initiating events, but many other refinements made during the assessment. Since the model has been changed substantially, the effects induced purely by the implementation of SFP-SOEOP cannot be quantified separately.

The risk contribution of the different initiating events to the overall FDP is shown in Figure 2, while the FDP distribution among the POSs is presented in Figure 3. According to the results, loss of the heat removal system is the dominant initiating event with a contribution of 82,99%. The contribution of fire events to the overall risk is also significant (13,81%). Besides these two risk significant contributors, all the other transients (LOOP, isolable and non-isolable loss of coolant accidents, internal flooding) show a risk contribution lower than 5%.

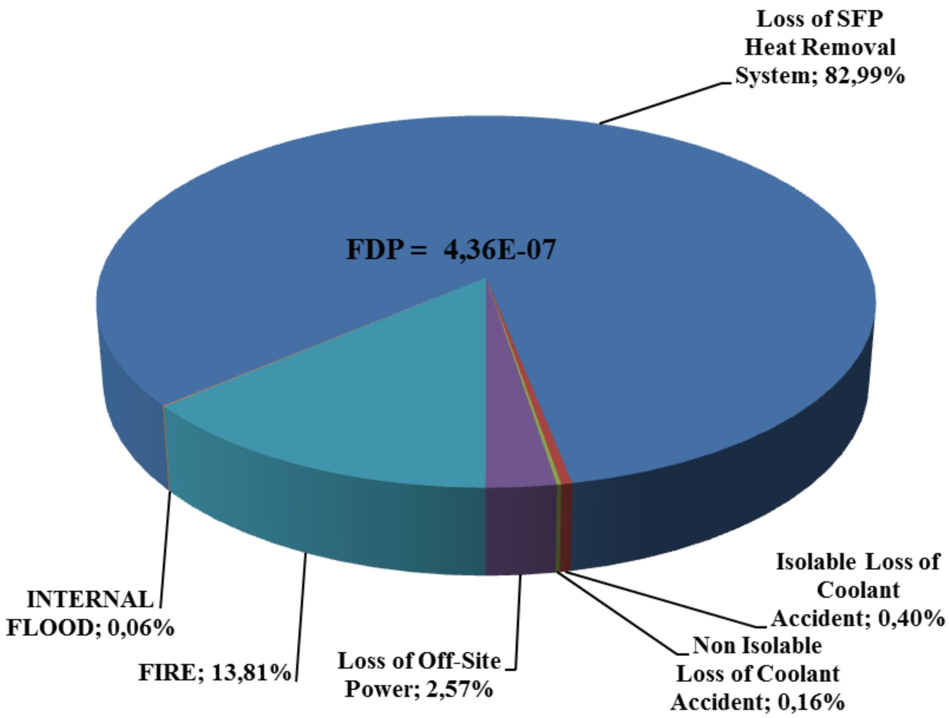

Fig. 2. Contribution of Initiating Events to FDP, Paks NPP Unit 1

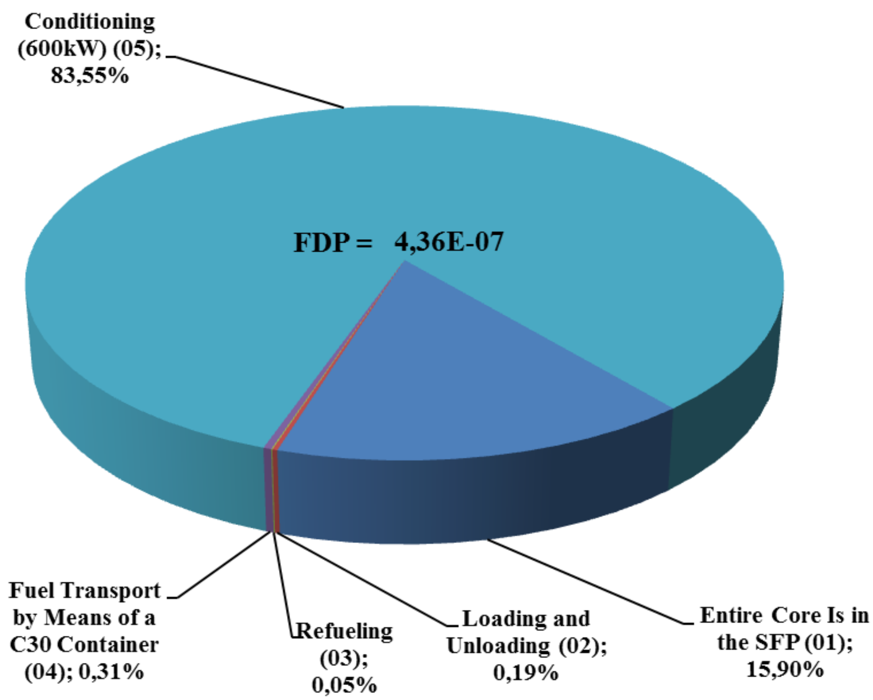

Fig. 3. Distribution of FDP over POSs, Paks NPP Unit 1

Some distinguishing features of the new HRA model has to be highlighted first in the interpretation of results obtained from the upgraded SFP PSA. Prior to the implementation of the SFP-SOEOP, post-initiator operator actions were generally modelled as independent events in the SFP PSA. In the model upgrade, dependence among the different, seemingly independent operator actions was also taken into consideration to eliminate some unjustifiably optimistic assumptions applied in the former assessment. A rigorous modeling of the EOP use and the steps to be followed resulted in the identification of certain dependencies among different operator interventions.

A modified assumption was introduced in the upgraded model to assess the failure probability of local human actions, i.e. opening motor operated valves. Prior to the upgrade a failure probability of 0.1 was used as a screening value to all local actions considered in PSA. The upgraded model assumes that SFP transients induced by fire events require local actions in any case. The failure of these actions was taken into account in the execution error. In other words, local interventions, regardless of the fire impact, also include corrective actions on valves mis-positioned due to fire-induced spurious actuations, as well as local operation of valves that cannot be operated from the control room due to fire.

Lastly, we updated the component reliability data that resulted in a significant change in frequency of the loss of heat removal system initiating event. This has greatly increased the contribution of this event to the FDP. Overall, the loss of heat removal system initiating event is the dominant event in the upgraded PSA, whilst the fire events appeared significant prior to the model upgrade.

Uncertainty assessment was made separately for internal events and for internal hazards. Some results of the uncertainty analysis performed for internal events by using the upgraded model are indicated in Table III. The simulation values are based on the RiskSpectrum PSA Professional runs, while the regression values were determined as a lognormal approximation of the simulation results. The error factor of the lognormal distribution is considered moderate: 4.55.

TABLE III. Uncertainties in Annual FDP Estimates Obtained from the Upgraded PSA Model

	Annual FDP	
	Simulation	Regression
Median	$2.19 \cdot 10^{-7}$	$2.26 \cdot 10^{-7}$
5%	$5.80 \cdot 10^{-8}$	$4.96 \cdot 10^{-8}$
95%	$1.01 \cdot 10^{-6}$	$1.03 \cdot 10^{-6}$

VI. CONCLUSIONS

The accident sequence models in the level 1 PSA for the SFP of the Paks NPP have been modified to reflect changes due to the transition from event based to symptom oriented EOPs for the SFP. The whole area of HRA has been reviewed and updated too. Use was made of the methodology elaborated previously in the full power PSA for the reactor of the Paks plant in support of PSA model development and re-quantification to reflect the effects of EOP improvement for the SFP. Data and findings from simulator observations performed earlier (2005) have been integrated into the HRA process for the SFP PSA. Besides modifications induced by EOP changes, the reassessment included several other potential model upgrades, primarily with respect to refinements in the definition of adequate emergency responses to SFP transients. Some additional elements of the SFP PSA model were also updated substantially, e.g. redefinition of POSs for the SFP, update of input reliability data, changes to the SFP initiating events. PSA re-quantification covered SFP PSA for internal events, internal fires and flooding. This paper summarized the change in fuel damage risk estimates for internal events and internal hazards due to the model upgrade.

ACKNOWLEDGMENTS

This work has been supported by the MVM Paks NPP Ltd., as well as by the National Research, Development and Innovation Fund in the frame of VKSZ_14-1-2015-0021 Hungarian project.

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