ADVANCED THERMAL HYDRAULIC SIMULATIONS FOR HUMAN RELIABILITY ANALYSIS

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This paper outlines a new approach to coupling traditional Human Reliability Assessment (HRA) methods with Thermal Hydraulic (TH) simulations. This was done in order to improve the calculation of failure probabilities calculated from HRA methodology that used conservative estimates. We utilized Electricité de France's (EDF) HRA method, MERMOS, TH software, MAAP4, and uncertainty software, DAKOTA. We identified TH inputs that are subjected to uncertainty and ran a Monte Carlo analysis with the reactor in operation as well as after a planned scram event. We tested a case in which a small break loss of coolant accident (SBLOCA) occurs in a generic Westinghouse 4-Loop Pressurized Water Reactor (PWR) and safety injection (SI) fails to automatically actuate. The operator mission is to manual begin safety injection. For each of the operating modes, we were able to demonstrate a reduction in scenario failure probability of over 80%.

I. INTRODUCTION

Most of the running fleet of nuclear power plants were designed and built using deterministic safety approach (DSA). Manual operator actions are rarely analyzed in DSA. With time, Probabilistic Safety Analysis (PSA) became an obligatory complement to DSA. PSA considers a wider range of initiating events, safety system configuration and operator actions. An event tree is used to represent the scenario progression after an initiating event; fault trees are used to evaluate the reliability of protection systems. Necessary operator actions can be explicitly modeled in an event tree if the action is transient specific, or it can be a part of the safety system fault tree.

There exist many methods used to evaluate the reliability of operator actions in nuclear units, known as Human Reliability Analysis (HRA), which implement different approaches to evaluate the reliability.[1,2,3,4] Most use transient (thermal hydraulic) data to evaluate the reliability of a needed operator action. The quality of the transient data is crucial to perform a realistic HRA.

After an initiating event, a variety of uncertainties in the thermal hydraulic response of a nuclear unit can significantly impact the performance of operators and thus the reliability of the needed action. Simulation based HRA methods and dynamic event trees are probably the most appropriate tools to handle this problem but their industrial application is yet to be seen.[6] In this work we demonstrate how uncertainties in thermal hydraulic response of a nuclear unit can be merged with the traditional HRA methods.

II. BACKGROUND AND MOTIVATION

There is a need for a method that will account for the thermal hydraulic (TH) uncertainties of a plant response to a transient in an HRA (Ref. 5). In particular, we seek to improve the calculations of risk through utilizing TH simulations rather than conservative estimates. Perhaps the best methodology for this relationship would be through the use of simulation-based HRA methods or dynamic event trees.[6] However, the goal of this research was to produce a methodology for accounting for the TH uncertainties of a plant response to a transient that is applicable in industry today. This eliminated the use of either simulation-based HRA methods or dynamic event trees, as they are not yet seen in industrial applications.[7] Instead we use existing HRA methodology, MERMOS (Ref. 4), coupled with TH simulation software, MAAP4 (Ref. 8) and uncertainty software, DAKOTA (Ref. 9). MERMOS was utilized because the research is supported by EDF and MERMOS is an EDF HRA methodology. MAAP4 is representative of advanced TH simulation software accessible in industry.

MERMOS is an approach to HRA used at EDF that analyses the failure scenarios comprehensively. Each different path leading to failure of the identified operator mission is labeled as a "failure scenario." The goal of MERMOS is to identify and examine as many of these independent scenarios leading to the mission failure. Since each failure scenario is assumed independent, the total mission failure can be calculated using Equation 1.

$$P_{mission\_failure} = \sum P_{failure\_scenario} + P_{residual} \qquad (1)$$

To calculate the failure probability of the failure scenarios, each scenario has three properties: situation characteristics (PS), context (CICA), and non-reconfiguration (NR). Utilizing the three properties, one can find the probability of a failure scenario occurring by using Equation 2.

$$\sum P_{failure\_scenario} = \prod P_{PS} \prod P_{CICA|PS} \prod P_{NR|CICA} \tag{2}$$

Since 2011, EDF has been working on an upgrade to MERMOS, named the "MERMOS catalogue" which is based on a generic analysis.[10] This generic analysis incorporates all existing analyses performed for similar cases. It integrates all variations considered by the analysts without further analysis (for example further TH analysis). We examine the probability of non-reconfiguration as a function of TH conditions in a plant for two case studies. These two case studies were determined to be dominant failure scenarios in the MERMOS Catalogue for a small break loss of coolant accident (SBLOCA) with failure of automatic safety injection.[10] Case 1 is modeled in three operational modes of the nuclear unit. Mode A is the "on-power" mode where the unit is operating normally. Mode B spans the first 6.5 hours after a scram for a planned outage. Mode C spans 6.5 hours to 13.5 hours after a scram for a planned outage. Case 2 is only modeled with Mode C because this is the operating mode reported in the MERMOS Catalogue. We seek to improve the estimate of the failure probability by improving the estimate of the probability of non-reconfiguration. A more detailed view on these cases from the HRA point of view can be found in Ref. 11.

III. METHODOLOGY

We analyze an SBLOCA (break size below 0.0508m diameter) initiating event for a 4 loop Westinghouse type PWR plant for on-power (Mode A) and shutdown states (Modes B and C). Mode C is further split into 4 sub-modes: C1, C2, C3, and C4. These are defined based on the pressure and temperature of the primary system. Safety injection (SI) is necessary to mitigate the core damage. On power, a manual action is needed in case of the failure of an automatic SI signal. In the shutdown states, automatic SI signal based on the low primary pressure threshold is off and core damage can be prevented only by an operator action. The reliability of the operator action, "manual safety injection," was reported in the MERMOS Catalogue. This indicated that there are two dominant failure scenarios for this action[10] as outlined in Case Study 1 and Case Study 2, respectively.

There are two variables measured: Saturation Time, Δt_{st}, and Mission Time, Δt_{mt}. Saturation Time is the time from the scram until the steam-liquid mixture in the primary system is saturated. If this time is large and the operator tests the system before the system is saturated, then there is a possibility misdiagnosing the transient. In our paper, we define Mission Time as the time from the first indication of a loss of coolant accident (steam-liquid saturation in primary system) until permanent damage is incurred (core uncovery).

III.A. Case 1

The vapor from the break can trigger a nearby spurious fire alarm. Operators give the priority to the fire procedures and do not start SI before the core uncovers. Fire procedures consist of 1) localizing the fire, 2) identifying the functions that are likely to be affected by the fire, and 3) conducting electric cut-offs.[10] The probability of non-reconfiguration is a function of Mission Time as shown in Equation 3.

$$P_{NR|CICA} = \begin{cases} 1, & \Delta t_{mt} \le 25\,\text{min.} \\ 0.9, & 25\,\text{min.} < \Delta t_{mt} \le 35\,\text{min.} \\ 0.3, & 35\,\text{min.} < \Delta t_{mt} \le 45\,\text{min.} \\ 0.1, & 45\,\text{min.} < \Delta t_{mt} \le 90\,\text{min.} \\ 0.01, & 90\,\text{min.} < \Delta t_{mt} \end{cases} \tag{3}$$

III.B. Case 2

Emergency Operator Procedures (EOPs) demand to confirm/manually start safety injection when steam-liquid mixture in the primary system is saturated. Usually the system saturation is tested at the beginning of the EOPs.[10] Because the first time that the operators test saturation of primary circuit can be before it occurs, it may be too late to start safety injection by the time they test for saturation the second time. The probability of this occurring is a function of Saturation Time. We assume the situation will occur (probability of 1) if the Saturation Time is greater than 3 minutes. The probability of failure to recognize the mistake is a function of Mission Time as seen in Equation 4.

$$P_{NR|CICA} = \begin{cases} 1, & \Delta t_{mt} \le 25\,\text{min.} \\ 0.3, & 25\,\text{min.} < \Delta t_{mt} \le 35\,\text{min.} \\ 0.1, & 35\,\text{min.} < \Delta t_{mt} \le 50\,\text{min.} \\ 0.01, & 50\,\text{min.} < \Delta t_{mt} \end{cases} \tag{4}$$

III.C. Uncertainty Parameters

The distributions of the inputted parameters are summarized in Table I for Modes A, B, and C. The selection of the parameters and their distributions are described below:

- **PPSL (Low Pressurizer Pressure Trip Point)**: This represents the uncertainty in the pressure measurements taken within the pressurizer. The lognormal distribution was selected with the mean, 12.5 MPa, because this is the default signal set point in MAAP4 (Ref. 12). The standard deviation was selected as 0.1 MPa to give an appropriate measurement error.
- **FCDBRK (Break Discharge Coefficient)**: Since the cause of the break is unknown, the shape of the break will likely be unknown. The uncertainty here captures this. The MAAP4 Application Guidance recommends testing sensitivity of this parameter.[8] The distribution was chosen with the mode at the default MAAP4 value of 0.75 (Ref. 12) and a reasonable range of 0.4.
- **VFSEP (Void Fraction Threshold)**: This is the void fraction threshold above which the primary system is no longer a homogeneous two-phase mixture. It was shown to have an impact on the TH progression during a transient. The distribution range was selected to be the same as the MAAP4 Applications Guidance.[8]
- **QCR0 (Initial Core Thermal Power)**: This is chosen as an uncertainty based on the load following of French reactors. At approximately 30% of the time, the reactor will run at 70% full power. At other times, it is assumed the reactor runs at 100% power (approximately 3236 MW). The thermal powers were selected as the default in MAAP4 (Ref. 12).
- **ABB (Break Size Diameter)**: This is chosen as an important uncertain parameter because the break size could be anything below a 0.0508m diameter to be considered a small break. We utilized the analysis of Ref. 13 to determine the histogram distribution. We used a parametric approach for the simulations where the break sizes 0.00635m, 0.0127m, 0.0191m, 0.0254m, 0.0318m, 0.0381m, 0.0445m, and 0.0508m were modeled uniformly. This distribution was only used in finding the scenario failure probability.
- **TWPS0 (Initial Primary Side Temperature)**: This is only analyzed in Modes B and C as the reactor is going offline. While the operator follows a shut down path, there is room for error. We developed a function for the mean temperature based on the shutdown profile from Ref. 14.
- **PPS0 (Initial Primary Side Pressure)**: This is only analyzed in Modes B and C as the reactor is going offline. While the operator follows a shut down path, there is room for error. We developed a function for the mean pressure based on the shutdown profile from Ref 14.
- **Time From Scram (t)**: This is used in determining the mean initial pressure and temperature of the primary side. The uniform distribution was selected as we assumed a break at any time is equally likely as others.

TABLE I. Uncertainty Parameter Distributions

MAAP4 Variable	Distribution	Characteristics	Operating Mode(s)
PPSL	Lognormal	Mean = 12.5 MPa Std. Dev. = 0.1 MPa	A

FCDBRK	Triangular	Mode = 0.75 Maximum = 1.0 Minimum = 0.6	A,B,C
VFSEP	Triangular	Mode = 0.5 Maximum = 0.8 Minimum = 0.2	A
QCR0	Discrete Binary	p(3236 MW) = 0.70 p(2265 MW) = 0.30	A,B,C
ABB	Histogram	p(0.00635m≤D≤0.0127m) = 0.57 p(0.00127m<D≤0.0381m) = 0.25 p(0.00381m<D≤0.0508m) = 0.18	A,B,C
TWPS0	Normal	Mean = f(t) Std. Dev = 5°C	B,C3/4
TWPS0	Normal	Mean = f(t) Std. Dev = 2°C	C1/2
PPS0	Normal	Mean = f(t) Std. Dev = 0.25 MPa	B
PPS0	Normal	Mean = f(t) Std. Dev = 0.1 MPa	C
Time from Scram (t)	Uniform	Maximum = 6.5 hr Minimum = 0 hr	B
Time From Scram (t)	Uniform	Maximum = 13.5 hr Minimum = 6.5 hr	C

III.D. Monte Carlo Analysis

We used system analysis code, MAAP4, coupled to uncertainty propagation software, DAKOTA, to study the variability of the parameters used in HRA MERMOS method as a function of TH uncertainties described previously. We used a Latin hypercube sampling method.[9] We ran 44,592 runs for Mode A, 38,097 runs for Mode B, and 12,547 runs for Mode C.

IV. RESULTS

We found that we can significantly reduce the conservatism of initial HRA analysis, demonstrating that the conservative failure option is not always supported by the thermal hydraulic state of the unit. For each operating mode, the probability of non-reconfiguration (PNR) was calculated in three ways:

- With Uncertainties (Integrated): The PNR was calculated using Equation 3 for each simulation run and then averaged.
- With Uncertainties (Averaged): The Mission Times from all of the simulation runs were averaged and then the PNR was calculated using Equation 3 from this average.
- Without Uncertainties: The default values (mean/mode of the distributions) were used in the simulation to get a Mission Time. The PNR was calculated from this Mission Time using Equation 3.

IV.A. Case 1: Mode A

Mode A is normal operation of the nuclear power plant. Figure 1 depicts the Mission Time histogram with a total count of 44,592 runs. Any Core Uncovery times past one day are excluded. As can be seen, the Mission Time is heavily dependent upon the uncertainty inputs. There are many modes in the histogram, corresponding to the different break sizes. Figure 2 shows the Mission Time as a function of the break size. The solid line represents the mean Mission Time and the shaded region represents 90% of the Mission Times.

We calculated the probability of non-reconfiguration for Case A. This is shown as a function of break size below in Figure 3. For smaller break size diameters (below 0.0318m), the probability of non-reconfiguration is at the lowest, 0.01, because there are no Mission Times below 90 minutes. As the break gets larger, the Mission Times become smaller and so the probability that the operator will not recognize the mistake increases. The maximum probability of non-reconfiguration occurs at the largest break size, 0.0508m diameter, and is 0.102. We also see a decrease in the range of Mission Times as break size decreases.

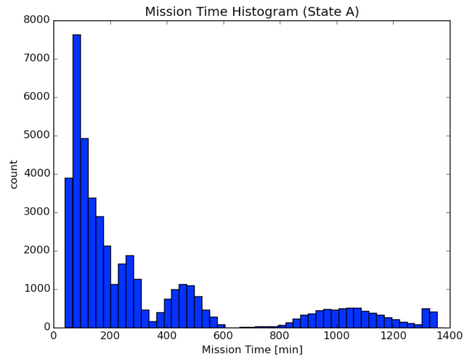

Fig. 1. Mission Time Histogram for Mode A, Case 1

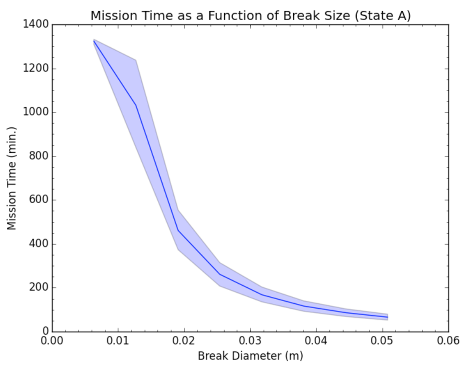

Fig. 2. Mission Time as a Function of Break Size Diameter for Mode A, Case 1

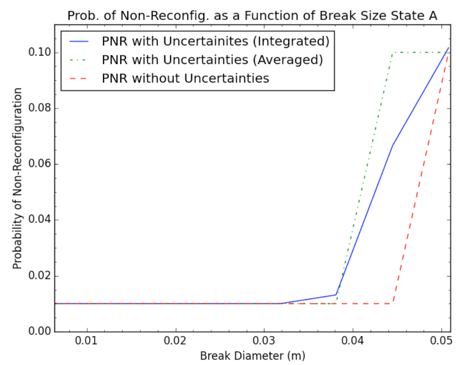

Fig. 3. Probability of Non-Reconfiguration as a function of Break Size Diameter for Mode A, Case 1

IV.B. Case 1: Mode B

Mode B is the first 6.5 hours of the nuclear reactor being taken offline. Figure 4 depicts the Mission Time histogram with a total count of 38,097 runs. Any Core Uncovery Times past one day are excluded. As can be seen, like Mode A, the Mission Time is heavily dependent upon the uncertainty inputs. It is most affected by the break size and so Figure 5 shows the Mission Time as a function of the break size. The solid line represents the mean Mission Time and the shaded region represents 90% of the Mission Times.

We again calculated the probability on non-reconfiguration for Case B. This is shown as a function of break size below in Figure 6. For smaller break size diameters (below 0.0381m), the probability of non-reconfiguration is at the lowest, 0.01, because there are no Mission Times below 90 minutes. As the break gets larger, the Mission Times become smaller and so the probability that the operator will not recognize the mistake increases. The maximum probability of non-reconfiguration occurs at the largest break size, 0.0508m diameter, and is 0.0246. The probability of non-reconfiguration is lower for Mode B than for Mode A, indicating that an operator will be less likely to fail this failure mode in Mode B than in Mode A.

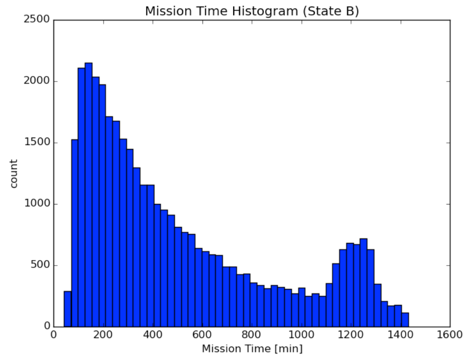

Fig. 4. Mission Time Histogram for Mode B, Case 1

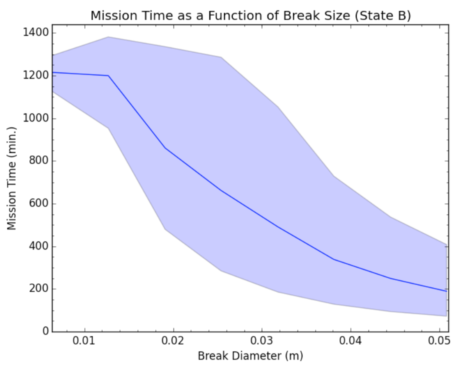

Fig. 5. Mission Time as a Function of Break Size Diameter for Mode B, Case 1

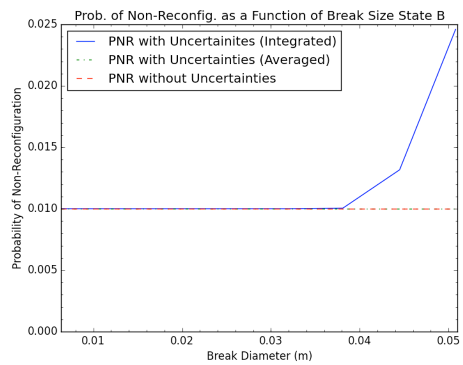

Fig. 6. Probability of Non-Reconfiguration as a function of Break Size Diameter for Mode B, Case 1

IV.C. Case 1: Mode C

Mode C is the time from 6.5 hours to 13.5 hours of the nuclear reactor being taken offline. Figure 7 depicts the Mission Time histogram with a total count of 12,547 runs. Any Core Uncovery Times past one day are excluded. As can be seen, like Mode A and B, the Mission Time is heavily dependent upon the uncertainty inputs. In addition, there are no Mission Times below 90 minutes. Figure 8 shows the Mission Time as a function of the break size. The solid line represents the mean Mission Time and the shaded region represents 90% of the Mission Times.

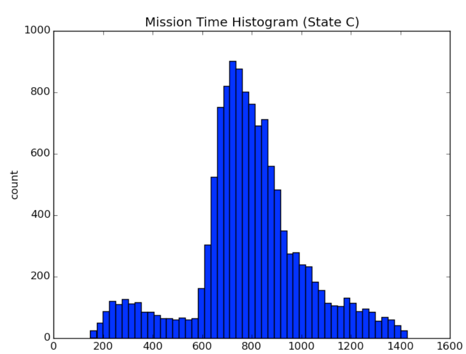

Fig. 7. Mission Time Histogram for Mode C, Case 1

Figure 9 depicts the probability of non-reconfiguration as a function of break size. Since there are no Mission Times below 90 minutes, the probability of non-reconfiguration is uniform across the break sizes at 0.01. Of the modes studied, in Mode C the operator is most likely to realize the mistake of entering fire procedures,

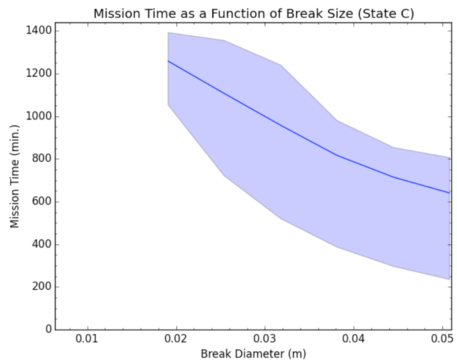

Fig. 8. Mission Time as a Function of Break Size Diameter for Mode C, Case 1

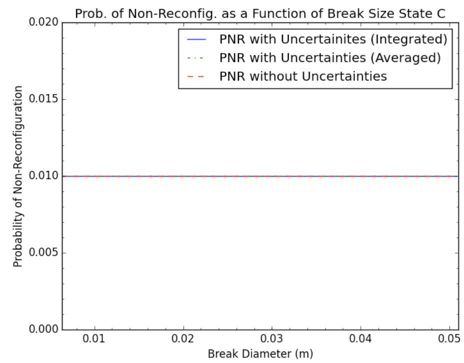

Fig. 9. Probability of Non-Reconfiguration as a function of Break Size Diameter for Mode C, Case 1

IV.D Case 2: Mode C

In Case 2, the operator tests for the steam-liquid saturation of the primary system before it occurs. We assume this will happen if the system reaches saturation after 3 minutes. For each simulation run where this occurs, we calculated the Mission Time the operator will have to retest for saturation. Saturation was reached after the 3 minutes threshold in 11,326 simulation runs out of the 12,547 simulation runs. A histogram of the Mission Times for the 11,326 simulation runs were this occurs is below in Figure 10.

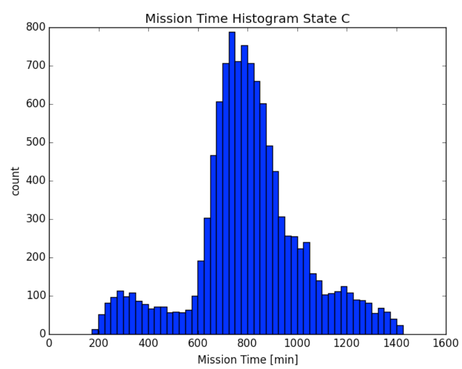

Fig. 10. Mission Time Histogram for Mode C, Case 2

There are no Mission Times below 50 minutes. Therefore, for each simulation run in Case 2 the probability that the operator will not retest for saturation (probability of non-reconfiguration) is 0.01 as per Equation 4. This is the same probability reported in the MERMOS Catalogue.

IV. CONCLUSIONS

We found that we can significantly reduce the conservatism of initial HRA analysis in Case 1. We accomplished this through demonstrating that the conservative failure option is not always supported by the TH analysis of a transient. Specifically, for a SBLOCA, we showed that the catalog analysis of Mission Time is consistently smaller than the TH analysis Mission Time. This allows the scenario failure probability to be reduced. Table II shows this for Case 1, where the operator goes into fire procedures and must realize that it is in fact a LOCA occurring not a fire.

TABLE II. Summary of Failure Probabilities for Case 1 for Each Reactor Operating Mode

	Mode A	**Mode B**	**Mode C**
MERMOS Catalogue Scenario Failure	0.22	0.22	0.012
Scenario Failure from Analysis with TH Uncertainty (Integrated)	0.0052	0.0025	0.0022
Percent Reduction	97.6%	98.9%	81.7%

In addition, we found that the failure probability of Case 2 for Mode C agrees with the analysis in the MERMOS Catalogue. The catalog stated the failure probability of this to be 0.00045. For Case 2, Modes A and B were not analyzed in the MERMOS Catalogue as it was assumed that the steam-liquid saturation would occur before the three minute threshold. This should be analyzed using this new approach to see if the TH simulation results agree with the assumption.

This study supports further research into the effect of coupling of advanced TH simulations with HRA due to the proven benefit the former can have on the latter. In addition, the method we utilized with coupling uncertainty software with a TH systems code is applicable in today's industry as there is greater access to computing power.

ACKNOWLEDGMENTS

The authors would like to thank the Massachusetts Institute of Technology and Electricité de France for supporting this work.

REFERENCES

1. L. ROOK, “A method for calculating the human error contribution to system degradation,” *Human error quantification, SCR-610* (1964).
2. A.D. SWAIN, “Method for performing a human-factors reliability analysis,” *No. SCR-685*, *Sandia National Laboratories* (1963).
3. S.E. COOPER, et al. "Knowledge-base for the new human reliability analysis method: a technique for human error analysis (ATHEANA)," *International topical meeting on probabilistic safety assessment moving toward risk based regulation*, Park City, UT (US), October 1, (1996).
4. C. BIEDER et al., "MERMOS: EDF’s new advanced HRA method," *Probabilistic Safety Assessment and Management (PSAM 4)*, September 13 (1998).
5 V. RYCHKOV, " Risk informed safety margins characterization via failure domain quantification. Loss of main feed water example," *Proc. ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis* (2013).
6. A. MOSLEH and Y. CHANG, “Model-based human reliability analysis: Prospects and requirements,” *Reliability Engineering and System Safety*, **83** (2014).
7. R.L. BORING et al., "Human performance modeling for dynamic human reliability analysis," *International Conference on Digital Human Modeling and Applications in Health, Safety, Ergonomics and Risk Management*. Springer International Publishing (2015).
8. F. RAHN, “MAAP4 applications guidance: Desktop reference for using MAAP4 software,” *Tech. Rep. 1020236*, *EPRI* (2010).
9. B. ADAMS et al., “DAKOTA, a multilevel parallel object-oriented framework for design optimization, parameter estimation, uncertainty quantification, and sensitivity analysis: Version 6.1 user’s manual,” *Tech. Rep. SAND2014-4633 Sandia National Laboratories* (2014).
10. P. LE BOT and H. PESME, “MERMOS Catalogue: Use of generic analysis to improve the HRA method," *Proc. PSAM11 ESREL* (2012).
11. V. RYCHKOV et al., "Advanced thermal hydraulic simulations for human reliability analysis," *Proc. ESREL* (2016).
12. EPRI, “MAAP: Modular accident analysis program for LWR power plants: User manual,” *EPRI* (1994).
13. K. FLEMING and B. LYDELL, " LOCA frequencies for GSI-191 applications," *Proc. ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis* (2013).
14 P. COPPOLANI et al., *La Cahudiere des reacteurs a eau sous pression*, EDP Sciences (2004).