Analysis of PWR Station Blackout Sequence Using MELCOR and Generic Severe Accident Management Guidelines within a Human Reliability Model

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Abstract:

Severe Accident Management Guidelines (SAMGs) are an important piece of a utility's strategy for minimizing impacts of an accident. Human Reliability Assessment (HRA) within SAMGs has not been thoroughly investigated. An integrated and history-dependent HRA/SAMG model was developed at The Ohio State University to explore possible human actions that may need to be taken following an accident for their verification though simulation.

The system under consideration is a pressurized water reactor undergoing a short-term station blackout initiating event (IE). The PWR behavior following the IE was simulated using the MELCOR code. The simulations showed that severe accident space is reached quickly based on an assumption of the residual water storage tank being unavailable. The data from the MELCOR model were input into the HRA model every ten minutes after the onset of fuel damage in MELCOR as input for operator actions.

Results were obtained using ten sequences sampled from MELCOR data for two HRA models: the Basic model which considers limited operator actions (five sequences) and the Extended model which considers additional operator actions (additional five sequences).

Keywords: PRA, HRA, Dynamic PRA, MELCOR, SAMG

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1. INTRODUCTION

The development and exploration of Severe Accident Management Guidelines (SAMGs) have been the work of various industries and researchers since the Three Mile Island accident and, more recently, the Fukushima accident [1, 2, 3]. When exploring these guidelines through simulation, it is typically assumed that operators are able to simultaneously receive all the data available in the control room and successfully perform the actions prescribed by the SAMGs [4]. The impact of human behavior is often considered only when focusing on the outcome of a particular scenario [4].

A drawback of using traditional static Probabilistic Risk Assessment (PRA) to investigate the effectiveness of SAMGs revolves around the restriction on the modeling of accident evolution. Static PRA specifies possible sequencing of events (scenarios) based on a limited number if simulations, often relying on expert judgement. This approach makes accounting for the interaction of hardware/process/software/human intervention difficult as the accident evolves. In Dynamic Probabilistic Risk Assessment (DPRA) [5], the constraint of pre-specified event order is removed through interaction with the computer code modeling the reactor behavior (simulator) as the accident evolves which allows modeling such interactions. This feature of DPRA leads to a more comprehensive view of scenario evolution that can produce end states that may not appear in a static PRA [6].

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PRA has three levels of classification [7]:

- Level 1: normal operation until fuel damage is reached;
- Level 2: onset of fuel damage until containment is breached; and
- Level 3: following containment breach (release to environment).

An often used DPRA methodology for all Level 1-2-3 PRA is the Dynamic Event Tree (DET) approach [5] due to its similarity to the conventional event tree approach. DET generation requires a driver and a simulator. The simulator models system evolution in time under a given set of conditions. The driver samples the uncertainty space within the possible sets of conditions specified by the user to explore various ways the system can evolve through time. Previous work (e.g., [8 - 11]) has linked drivers, such as ADAPT [12] or RAVEN [13], with different simulators. The reader is referred to [14] for a more comprehensive review of state-of-the-art work in DPRA.

Validation of SAMGs experimentally is not feasible due to the impact of their possible adverse consequences on the environment. The objective of this paper is to provide an approach for their verification using DETs with a history-dependent Human Reliability Analysis (HRA) model [15] and a nuclear power plant (NPP) severe accident analysis code as the simulator, MELCOR [16]. The overall approach proceeds in the following fashion: an initiating event (IE) occurs and MELCOR runs until a user-defined stopping time is reached. At that time, user-defined data are exported from MELCOR, categorized based on the needs of the HRA model and translated into a binary input file required by the HRA model. The HRA model is then executed and run to completion. A binary string is output and the HRA model state is saved. The binary string contains the HRA model actions requested by the operator. Those actions are translated to a MELCOR-accepted input determined by the user-setup, and MELCOR resumes the scenario execution using the HRA's output actions until another user-defined stopping time is reached. When the HRA model is executed again, the saved model state is used, as the HRA model is history-dependent.

The paper concentrates on the functioning of the HRA model with input from an example MELCOR-simulated power plant simulation following the initiating event (IE). Section 2 presents the methodology behind this work. Results are presented and discussed in Section 3. Conclusions and future work are provided in Section 4.

2. METHODOLOGY

The following subsections describe: (i) the HRA model (Section 2.1), SAMGs (Section 2.1.1), and input/output format (Section 2.1.2); (ii) the example plant and the IE (Section 2.2); and, (iii) the linking process of the two codes (Section 2.3), and the thresholds used (Section 2.3.1).

2.1. HRA Model

The HRA model contains generic SAMGs and is used to explore the outcomes of a Pressurized Water Reactor (PWR) undergoing a Station BlackOut (SBO) event. The HRA model [15, 17] was developed at The Ohio State University and is a history-dependent model. Two HRA models are used in this research. The first model is a Basic model, where limited operator actions are derived from literature related to severe accidents. The second model is the Extended model, where additional operator actions are considered based on a review of the SAMG-D toolkit from the International Atomic Energy Agency [18].

The HRA model is a dynamic, mechanistic model with three components:

1. information perception,

2. knowledge base, and
3. human reasoning and decision making.

Information perception refers to the operator's ability to intake the signals available to him/her in the control room. For example, the control room displays data for the core and status of the steam generator, among a variety of other data for the plant. The operator may be mentally focused on the dynamics in the core. In this case, it is likely that the operator neglects the information about the steam generator. This behavior is modeled by the activation level of each available signal in the operator's mental state. Information perception is an important step in modeling the fact that the operator may fail to recognize signals that are important for diagnosing the state of the plant.

The knowledge base component is represented by a semantic model composed of the operator's knowledge of the system and its interactions. The system knowledge is derived from the operator's understanding of how a particular component works, such as a valve. *What are the characteristics of the valve?* Status and reliability. *What are the states of the characteristics?* Status: open and closed, reliability: low and high. These states are connected via logic gates and lead to the information perceived by the operator.

The final component, human reasoning and decision making, is the footpath from the interpretation of information available to the operator to the operator's decision to take (or not take) an action within nodes of the semantic network. The reasoning and decision-making process can be seen as an information retrieval process. The operator tries to retrieve the information in the knowledge base that can explain the observed signals. There are uncertainties in the reasoning and decision-making process and whether the operator can retrieve the useful information successfully. The uncertainties are represented in the HRA model by sampling based on the activation level of each node in the knowledge base following Anderson's spreading activation theory [19, 20] shown in Eq.(1)

$$A_i = B_i + \sum_j W_j \, S_{ji} \qquad (1)$$

where A_i is the current activation level of node i, B_i is most recent activation level of node i, W_j is the activation level of node j, and S_{ji} is the association strength from node j to i. Using a fixed MELCOR output, five samples each were taken for the Basic and Extended models to represent uncertainties, leading to a total of ten HRA sequences.

Human reasoning and decision-making is influenced by factors internal to the operator: stress and fatigue. The stress level of an operator is a function of the activation level of certain nodes (e.g., activation of a node such as core melt) as a result of the operator's experience in the adverse conditions of the unfolding accident. The fatigue level is expressed as a function of the time duration of the operator's exposure to the event and the activation levels of the nodes in the operator's knowledge base.

2.1.1. SAMGs

Integrated into the HRA model are SAMGs (see Section 3 for examples). These SAMGs were developed for a generic PWR, and thus, are not specific to the PWR model used in this study. Since the SAMGs are integrated into the HRA model and the HRA model (when fully enabled in the MELCOR model) controls the actions taken, SAMGs do not need to be explicitly included in the MELCOR input files for functionality.

As stated above, the two HRA models have SAMGs with different bases. The Basic model's SAMGs are based on a literature review and have fewer possible operator actions. The Extended model has more operator actions and its SAMGs are based on the SAMG-D Toolkit [18]. Table 1 lists the actions each HRA model can request.

Table 1: HRA models' possible operator actions.

Operator action	**Available in Basic model?**	**Available in Extended Model?**
Activate low pressure core cooling system	Yes	Yes
Depressurize the Reactor Pressure Vessel (RPV)	Yes	Yes
Initiate steam generator injection system(s)	No	Yes
Activate high pressure injection pump	Yes	Yes
Depressurize steam generators	No	Yes
Initiate internal containment sprays	No	Yes
Initiate external containment sprays	No	Yes
Turn on hydrogen igniter	No	Yes
Initiate RPV external cooling	No	Yes

2.1.2. Input & Output

The HRA model accepts a binary string representative of the state of a variety of signals the operator may see. This input mode dictates that the signals must be binned into classes. Depending on the signal, categorization levels vary. For example, containment pressure can be normal, high, or high high, while water level in the containment can be normal or high. Likewise for valves, classes of open or closed exist. The thresholds and categories for these cutoffs (shown in Section 2.3.1) were chosen by the authors and are not specific to the PWR model used in this scenario. They are merely for the purposes of demonstrating how these codes can work together.

Output from the HRA model is a binary string of nine characters with each character representing the operator's recommended actions to be taken to modify the current states of various systems in the plant.

2.2. Example Plant and Initiating Event

The nuclear power plant modeled is a three loop PWR. The computational modeling was executed by MELCOR [16], a fully-integrated code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission. Operation of the plant is modeled by control functions within MELCOR based on plant procedures, including Emergency Operating Procedures (but not plant SAMGs). The IE under consideration is a short-term SBO — loss of all on- and off-site power with no batteries available. To minimize computation time, the maximum simulation time was chosen as 12 hours after the IE. In order to reach severe accident space within that time frame, the Residual Water Storage Tank (RWST) was disabled, eliminating a large water source for cooling systems.

2.3. Code Linking

Initial linking was to be completed using ADAPT on a Linux computational cluster. Due to compatibility issues among the MELCOR model, HRA executable, and software policies, a one-way linking method was used with a single example MELCOR run. The MELCOR data were generated in the first stage of simulation with an External Data File (EDF) exporting a specific set of data every minute from MELCOR for use in the HRA input in the second stage of simulation. The time of core damage onset was identified based on MELCOR output and control functions and marked the entrance of severe accident space (the period of time in which the HRA model is valid for use). The data contained in EDFs are:

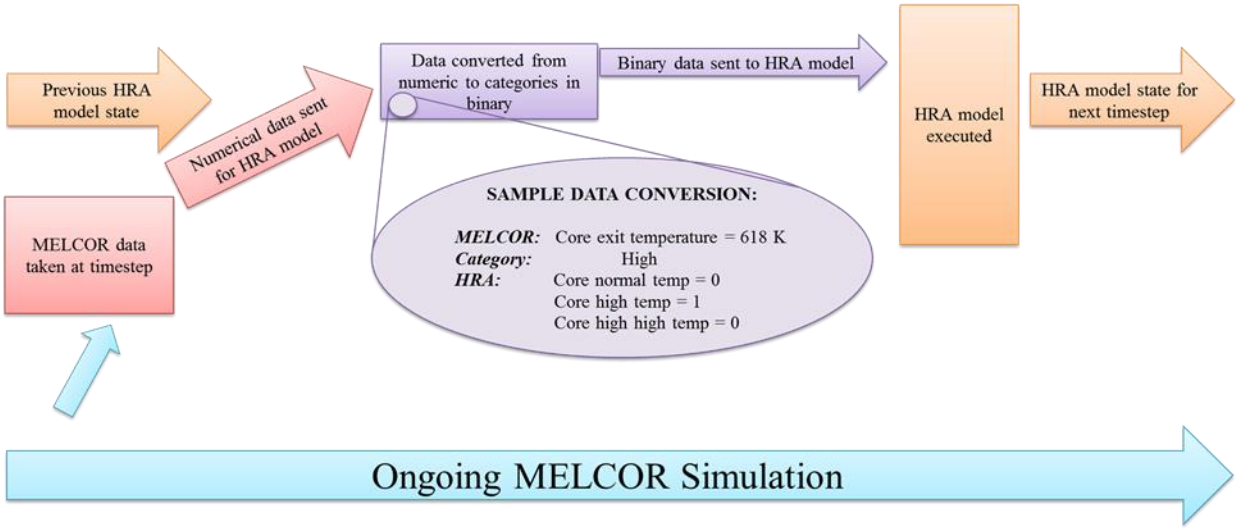
- containment pressure;

- pressurizer pressure;
- core exit temperature;
- containment water level;
- feedwater flowrate;
- hydrogen concentration; and
- steam generator safety valve (SV) open fraction.

At each branching (stopping) point in MELCOR, information is passed from MELCOR to five sequences of each HRA model (see Section 2.1). Each sequence executes the HRA model and produces an output which is stored. This process is depicted in Figure 1.

Data from the MELCOR model were fed into the HRA model every ten minutes, beginning at 5 hours and 30 minutes after the IE (following the entrance into severe accident space). Each resulting HRA sequence was saved individually following execution as the model is history-dependent.

Figure 1: MELCOR & HRA models communication process and sample data conversions.



2.3.1. Thresholds & Categories

As mentioned in Section 2.1, the HRA model does not accept numerical values. Data are binned into categories for the HRA model. The categories vary based on which variable is being input. For example, core exit temperature has categories of normal, high, and high high (see Figure 1 and Table 2), while feedwater flowrate has normal and low categories. The thresholds for these categories were chosen to be realistic but are not plant specific. Table 2 contains the variables transferred from MELCOR to the HRA model along with their thresholds and corresponding categories.

Table 2: Plant data sent to HRA model with thresholds and categories.

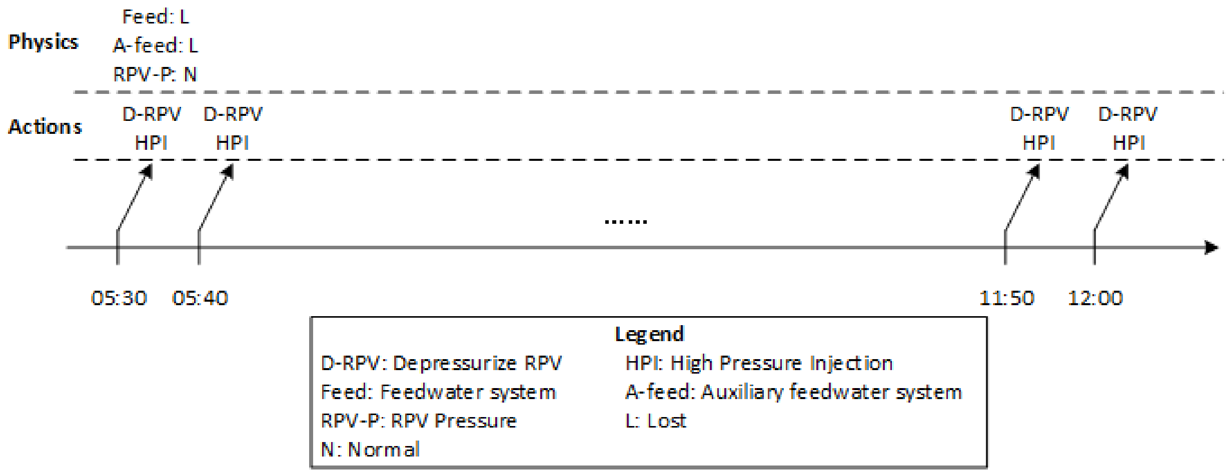
Signal	Low	Normal	High	High high
Change in containment pressure (Pa/hr)	-	otherwise	>5E3	>1.01E5
Containment pressure (Pa)	-	otherwise	>1.3E6	>2.6E6
Pressurizer pressure (Pa)	<1.2E7	otherwise	>1.7E7	-
Core exit temperature (K)	-	otherwise	>616	>650
Containment water level (m)	-	otherwise	>0	-
Feedwater flowrate (m^3/s)	otherwise	>200	-	-
Hydrogen concentration (mole fraction)	<0.01	otherwise	>0.05	-
Steam generator SV open fraction	otherwise (closed)		>0.5 (open)	

3. RESULTS

As indicated in Section 2.3, all ten of the HRA sequences were based on identical plant states. Analysis showed identical outputs for the Basic HRA model and distinct outputs for the Extended HRA model.

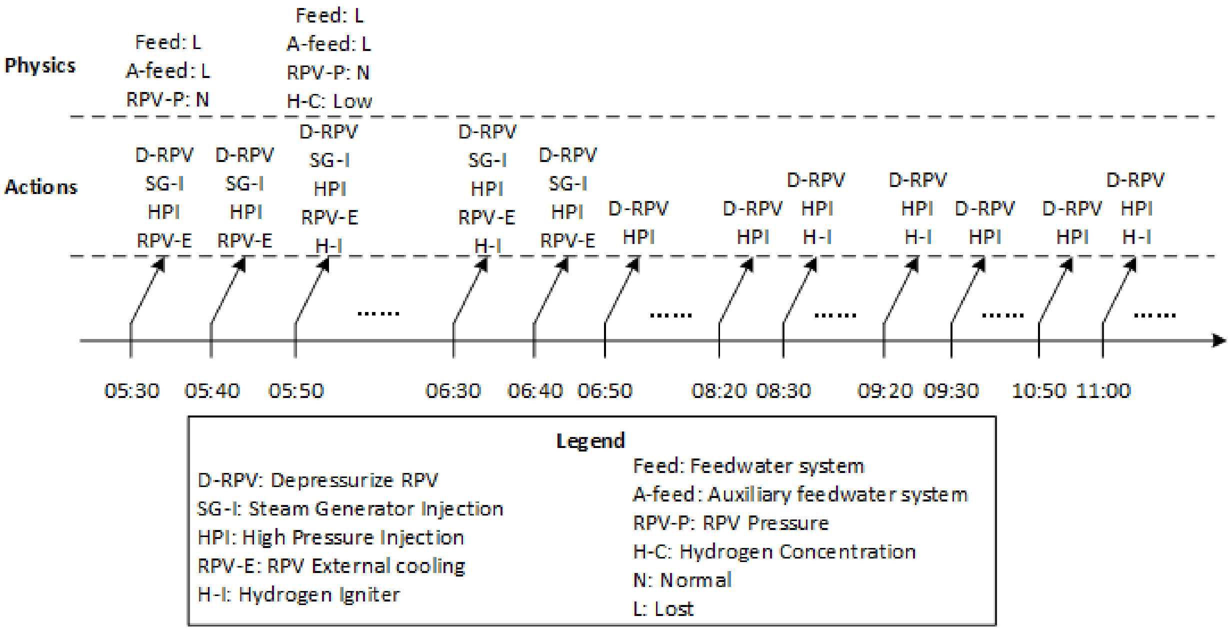
Operator actions (see Section 2.1.1 for full list) in the sequence for the Basic HRA model are shown in Figure 2. In this sequence, two actions are called upon at 5.5 hours after the IE: *depressurize the RPV* (Action 1) and *activate high pressure injection pump* (Action 2). In the HRA model, the logic for activating Action 1 is *loss of feedwater* and *loss of auxiliary feedwater*. These systems are assumed to be non-functioning in the MELCOR simulation so the conditions necessary for the activation of the first action are met. To activate Action 2, three conditions need to be satisfied: *loss of feedwater, loss of auxiliary feedwater*, and *RPV pressure high or normal*. The result of the MELCOR simulation shows that the RPV pressure is at the normal level as during full power operation, which leads to the activation of Action 2. However, these actions are not actually implemented in this study since the emphasis is on the HRA model as indicated in Section 1.

Figure 2. Operator actions for the Basic HRA model



The Extended HRA model did not have identical results. A sample sequence from the five trials is shown in Figure 3 and discussed here. In this sequence, more operator actions are called upon than in the Basic trials. Five and half hours after the IE, four actions are activated: 1) *depressurize the RPV* (Action 1), 2) *initiate steam generator injection system* (Action 2), 3) *activate high pressure injection pump* (Action 3), and 4) *initiate RPV external cooling* (Action 4). The requirements for activating Actions 1 and 3 have been described above. The requirement for activating Action 2 is *RPV pressure high or normal*, which is satisfied. The requirement for activating Action 4 is *RPV pressure high or normal* or *reactor temperature high-high*, which is also satisfied. Following these four actions, another action is called upon at 5:50 hours following the IE: *turn on hydrogen igniter*. This action is requested because the hydrogen concentration level changes categories from *negligible* to *low* based on the MELCOR simulation's data. When the hydrogen concentration level increases from *low* to *high* at 6:40 hours after the IE, the action *turn on hydrogen igniter* is not called upon any longer, consistent with the logic for activating this action in the HRA model (which is to avoid hydrogen explosion in containment). When the hydrogen concentration returns to the level of *low* at 8:30 hours after the IE, this action is called upon again.

Figure 3. Operator actions for the Extended HRA model



Comparing among the five sequences for the Extended HRA model, we also find that their results differ from each other in terms of operator actions requested. For example, in another of the five trials (not shown here), the action *initiate internal containment sprays* is called upon at a very late time in the accident progression (11 hours following the IE), in addition to the five actions activated in this sequence. The times at which the actions are called upon are also different in different sequences. For instance, the action *initiate internal containment sprays* is called upon early (7:10 hours following the IE) in another of the five sequences (also not shown here). These differences in operator response to an accident may lead to different consequences for the accident, which merits further investigation in future research.

4. CONCLUSION

This study illustrates how history-dependent SAMGs and HRA models can be implemented using a DET approach. While the plant results were not differentiable among the end states due to the use of a single MELCOR run, the variance in the HRA model suggests that with increased HRA sampling and adding reciprocal data transfer between the HRA model and MELCOR, different end states are likely to occur.

With the feasibility of the history-dependent SAMGs and HRA modeling coupled with MELCOR, increased HRA sampling and additional branching based on actions taken can be achieved. An additional opportunity lies in allowing the time between sampling (branching) to vary based on the rate of change in the data and is left as an option for future work.

Although operator actions have been augmented in the extended HRA model, there are still a limited number of actions possible. More operator actions can to be added to the HRA model to be considered in future research. These actions can be determined by referring to SAMG documents and consulting experts on severe accident management. Future work is planned to couple the HRA model with the MELCOR plant model and run a larger DET on a computational cluster.

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References

[1] US Nuclear Regulatory Commission, “Prioritization of recommended actions to be taken in response to Fukushima lessons learned,” *Commission Paper SECY-11-0137, October*, vol. 3, 2011.
[2] “Diverse and flexible coping strategies (FLEX) implementation guide,” NEI 12-06 Rev 1, ML15244B006, United States Nuclear Regulatory Commission, October 2015.
[3] “Emergency response procedures and guidelines for extreme events and severe accidents,” NEI 14-01 (Draft), Nuclear Energy Institute, Washington, DC, March 2014.
[4] K. Zhang, X.-W. Cao, and J.-Y. Che, “Study on severe accident mitigation measures for the development of PWR SAMG,” *Nuclear Science and Techniques/Hewuli*, vol. 14, no. 4, pp. 245–251, 2006.
[5] T. Aldemir, “A survey of dynamic methodologies for probabilistic safety assessment of nuclear power plants,” *Annals of Nuclear Energy*, vol. 52, pp. 113–124, Feb 2013.
[6] P. Cacciabue, A. Amendola, and G. Cojazzi, “Dynamic logical analytical methodology versus fault tree: the case study of the auxiliary feedwater system of a nuclear power plant,” *Nuclear Technology*, vol. 74, no. 2, pp. 195–208, 1986.
[7] “Severe accident risks: An assessment for five U.S. nuclear power plants,” NUREG-1150, United States Nuclear Regulatory Commission, Washington, DC, 1990.
[8] Z. Jankovsky, M. Denman, and T. Aldemir, “Extension of the ADAPT framework for multiple simulators,” in *Transactions of the American Nuclear Society*, vol. 115, (Las Vegas, NV), pp. 557– 560, American Nuclear Society, November 2016.
[9] V. Rychkov and K. Kawahara, “ADAPT-MAAP4 coupling for a dynamic event tree study,” in *ANS PSA 2015 International Topical Meeting on Probabilistic Safety Assessment and Analysis*, (Sun Valley, ID), April 2015.
[10] C. Rabiti, A. Alfonsi, D. Mandelli, J. Cogliati, R. Martineau, and C. Smith, “Deployment and overview of RAVEN capabilities for a probabilistic risk assessment demo for a PWR station blackout,” Tech. Rep. INL/EXT-13-29510, Idaho National Laboratory (INL), 2013.
[11] C. Picoco, T. Aldemir, V. Rychkov, A. Alfonsi, D. Mandelli, and C. Rabiti, “Coupling of RAVEN and MAAP5 for the dynamic event tree analysis of nuclear power plants,” in *Conference: The second international conference on engineering sciences and technologies*, 2017.
[12] A. Hakobyan, Severe Accident Analysis using Dynamic Accident Progression Event Trees. PhD thesis, The Ohio State University, 2006.
[13] C. Rabiti, A. Alfonsi, J. Cogliati, D. Mandelli, and R. Kinoshita, “Reactor analysis and virtual control environment (RAVEN) FY12 report,” INL/EXT-12-27351, Idaho National Laboratory, Idaho Falls, ID, September 2012.
[14] T. Aldemir (Ed.), *Advanced Concepts in Nuclear Energy Risk Assessment and Management,* World Scientific Publishing Co., Hackensack, N.J. (2018)
[15] Y. Zhao and C. Smidts, “A dynamic mechanistic model of human response proposed for human reliability analysis,” in *Advances in Human Error, Reliability, Resilience, and Performance* (R.L. Boring, ed.), (Cham), pp. 261–270, Springer International Publishing, 2018.

[16] L. Humphries, R. Cole, D. Louie, V. Figueroa, and M. Young, “MELCOR computer code manuals - Vol. 1: Primer and user’s guide - Version 2.1.6840 2015,” SAND2015-6691R, Sandia National Laboratories, Albuquerque, NM, August 2015.

[17] Y. Zhao, K. Altman, K. Chaudhury, M. Anandika, and C. Smidts, “A systematic method to build a knowledge base to be used in a human reliability analysis model,” in *International Conference on Applied Human Factors and Ergonomics*, pp. 50–60, Springer, 2017.

[18] “Severe accident management programmes for nuclear power plants,” Safety Guide NS-G-2.15, International Atomic Energy Agency, Vienna, Austria, July 2009.

[19] J. R. Anderson, “A spreading activation theory of memory,” *Journal of Verbal Learning and Verbal Behavior*, vol. 22, no. 3, pp. 261 – 295, 1983.

[20] J. R. Anderson, D. Bothell, M. D. Byrne, S. Douglass, C. Lebiere, and Y. Qin, “An integrated theory of the mind,” *Psychological review*, vol. 111, no. 4, p. 1036, 2004.