Dynamic Context Quantification for Design Basis Accidents List Extension and Timely Severe Accident Management

Emil Kostov[a,b] and Gueorgui Petkov[a]
[a] Technical University, Sofia, Bulgaria
[b] WorleyParsons, Sofia, Bulgaria

Abstract: The complexity of nuclear power stations (NPS) makes them more difficult to be controlled and increases the need for their comprehension and automation. Severe accidents (SA), besides the manifestation of deficiencies in design basis, the emergency plan preparedness, accident management, organization and safety culture, showed that a kind of more elusive dependence between the expected and unexpected events existed. Safety investigations of complex installations are unavailing without comprehensive evaluation of human, organization and technology (HOT) context, development and improvement of methods and concepts for finding effective decision-making and keeping the HOT balance. The paper presents the capacities of the Performance Evaluation of Teamwork procedure for HOT context quantification during the accidents for retrospective monitoring and event analysis, which reveal the SA mechanisms and assess their probabilities as well as the perspective for systematic understanding of the effect-cause relationships. The SA context quantification is proposed as an additional criterion for the extension of the detailed list of credible postulated initiating events and for increasing the effectiveness of timely SA management. The context quantification is exemplified by the situations at the Fukushima Daiichi NPS and the Onagawa NPS after the Great East Japanese Earthquake and by two SA simulations on VVER-1000 MELCOR 1.8.5 model.

Keywords: PSA, DSA, Accident context, DBA PIE list extension, Timely SA management.

1. INTRODUCTION

The accident at the Fukushima Daiichi Nuclear Power Station (NPS) focused the attention of the society and researchers on the NPS safety and the interaction between the methods for accident investigation, prevention and management. The method of defining the accidents to be considered in the design was subsequently named “Deterministic Safety Analysis” (DSA), to be distinguished from the “Probabilistic Safety Analysis” (PSA) based on the evaluation of the probability of the various accidental events. Compared to PSA, the DSA analysis does not give any general parameter or firm criterion about accident impact. It just gives a list/set of postulating initiating events (PIE) based on engineering insights, criteria and requirements. As a firm criterion for the process of PIE determination and selection is not available, the process is a combination of iteration between design basis and safety analysis, between technical judgments and previous design and operational experience. The exclusion of certain event sequence should be justified and explained. However, it involves the risk to overlook any initiating event that is crucial for the severe accident (SA), e.g. combination between earthquake, tsunami and blackout was not included in the PIE list as a design basis accident (DBA) of the Fukushima Daiichi NPS. Consequently, some additional criterion or calibration between the probabilistic and deterministic approaches is necessary when imposing the probability and impact as safety criteria for choosing PIE [1].

The risk management process, which describes the actual method of identifying, analyzing, and treating risks, as defined by ISO 31000, is “multi-step and iterative; designed to identify and analyze risks in the organizational context” [2]. Severe nuclear accidents, besides the manifestation of extended DBA, deficiencies in emergency plan preparedness, accident management, organization and safety culture, showed that a kind of more elusive dependence between the expected and unexpected events existed [3]. This dependence is based not only on the continuously growing complexity of technological systems, but also on the fact that a lot of opportunities for safety management are underlying in the context of interaction between human, organisation and technology (HOT).

The severe accidents (SAs) progress in a very uncertain and highly stressful context for the decision-makers. The SA environment requires a combination of contemporary engineering safety features and complex communication model for accident management. Therefore, the communication contexts assessment of different teams is of primary importance. All of the accidents occurred in the past were unexpected and demonstrated a number of erroneous human actions (HA), unforeseen weaknesses of the design and the interface between the organizations responsible for safety. Therefore, a serious commitment and responsibility for operational safety as well as promoting strong safety culture at all organizational levels are required in order to avoid emergency situations [4,5].

The NPS safe operation is ensured by the consistent application of normal operation procedures, emergency operation procedures (EOPs) and severe accident management guidelines (SAMGs). It should be noted that while the normal operation procedures and the EOPs consist of well-defined instructions and actions, the application of SAMGs requires evaluating the current plant status of the available equipment. That is why it is appropriate to evaluate the dynamic symptom-based context as a basis for the decision-making [6].

The aim of this paper is to present the capacities of a Performance Evaluation of Teamwork procedure [6] for HOT context quantification during SA retrospective studies and simulations to analyse the interrelation between the HOT context and the capability for accident management considering the application of SAMGs and the different layers of the emergency response organization. The Performance Evaluation of Teamwork is a human reliability analysis (HRA) method that distinguishes between three basic models determining the reliability of team performance: individual cognition/execution, team communication and leadership. They are based on the quantification of the context probability (CP) and communication context probability (CCP) of the team members by consecutive application of the violation of objective kerbs method in the combinatorial context model. The results of the context quantifications are used for obtaining human error probability (HEP) of individual and team decision-making [7].

A HOT context quantification procedure is applied during the SA retrospective analyses and simulations in order to serve as an additional criterion for the evaluation of the HA feasibility, for the extension or screening of the credible DBA PIE list and for increasing the effectiveness of timely SA management.

The conducted SA retrospective analysis is based on the dynamic context quantification of the situations at the Fukushima Daiichi NPS and the Onagawa NPS after the Great East Japanese Earthquake on March 11th 2011. Also two accident scenarios are simulated – large break loss of cooling accident (LB LOCA) during the station blackout (SBO) and long-term SBO (LT SBO) – using MELCOR 1.8.5 model of the VVER-1000/V-320 type reactor. The HOT context and the HEPs are calculated for each of the scenarios in order to provide an insight for the level of uncertainty of the HAs and the decision-making process. For the lack of space, in the paper only contexts are presented.

2. ADDITIONAL CONTEXT CRITERION FOR ACCIDENT MANAGEMENT

2.1. The HOT Context

The ISO 31000 states that safety management could be used to minimize, monitor and control the probability and/or impact of unfortunate events or to maximize the realization of opportunities. In engineering terms, the safety management is measurement of the probability and impact of accidents that could be arisen out of a PIE. The trivial approach is to calculate the PIE frequency and to prepare an adequate deterministic physical model of the installation processes to be simulated. Adverse scenarios and hazards damaging installation and progressing to SA could be identified based on the PIE list. Appropriate technical and organizational measures have been designed against these challenges. The total sum of the products of quantifiable impact (consequences) and calculated frequencies of consequences for all PIEs, mitigated by adequate safety measures, is the risk. Risk management requires knowledge about scenario details as significance, cause, etc. They are based on profound retrospective analyses and usually describe qualitatively accident context of analysts' understanding. However, successful accident management requires understanding of front-line operators' and managers' "second-by-second" context with their terms, images and organization, i.e. the dynamic HOT context [6,8].

Any component of the HOT contributes to the safety and could not be distinguished. The HOT composition should be measured and controlled by the SA management. However, the impact of each contributor is unavailing without a comprehensive and dynamic evaluation of the HOT context, development and improvement of the methods for determination and evaluation of decision-making HEP and keeping the HOT balance.

The HOT context consists of the ideas, situations, events, parameters, functions and all sorts of information that relates to it and makes possible its full understanding [9]. On the one hand, a context description of a given situation should reflect dynamically all specific information about the mind, technology and environment before and after the PIE [10]. On the other hand, the description of the HOT ensemble and context elements must be sufficiently general for the HOT of specific control area [11]. Consequently, the use of several levels of context elaboration is necessary.

2.2. In-Depth Calibration between Probabilistic and Deterministic Approaches

SAs are studied using a deterministic approach, with less conservative assumptions than DBAs in view of their low probability of occurrence. Probabilistic methods are used for the identification of those accidents which should be considered in a safety analysis. Today, probabilistic techniques are sometimes used to aid decisions concerning the deterministic approach – risk-informed and performance-based. For example, if a new candidate appears (e.g. from research or operating experience) for inclusion in the PIE list of DBAs, the decision about its inclusion can be aided by a probabilistic comparison to other situations, that are already inserted in the list.

There are thousands of event sequences that have to be investigated and analysed by integral code simulations with specific boundary conditions (BCs) and actual availability (configuration) of NPS structures, systems and components (SSC) to understand and predict SA sequences. Certainly, it is possible to group and limit these event sequences to make the task practical. The detailed deterministic analysis is reduced to a number of representative event sequences and an identification of bounding cases that have similar accident progressions is made, but it is quite subjective and questionable. The probabilistic tools for risk monitoring may help to obtain a risk-informed notion of what is planned to happen. The situation is even more complicated, since it is needed to take into account not only installation BCs but the whole HOT accident context. If the HOT context is quantified and the risk is calculated and monitored for the determined configuration dynamically, then some risk-informed criteria for accident management could be proposed and the additional knowledge of the integral code simulations could allow an effective planning of safety measures. Such in-depth calibration between risk monitoring (probabilistic approach), integral code simulations (deterministic approach) and context quantification (DSA & PSA approach) would enable the restriction of the detailed analysis to a limited number of representative event sequences.

2.3. Uncertain Criteria for the PIE Selection

"A PIE is defined as an event identified in design as leading to anticipated operational occurrences or accident conditions" [12], i.e. the PIEs are unintended events that directly or indirectly challenge the critical safety functions (SF). However, if the events are intended, unfortunate or unexpected with very specific context, timing and dynamics of SF challenging, then this variability of important accident events could be omitted from the PIEs list.

"Some PIEs may be specified deterministically, on the basis of variety of factors such as experience of previous plants, particular requirements of national licensing bodies or perhaps the magnitude of potential consequences. Other PIEs may be specified by means of systematic methods such as a probabilistic analysis … [12].” Any PIE necessitates SA management decision-making and protective actions to prevent or mitigate undesired consequences. A PIE may be of a type that has minor consequences, if the plant is successfully protected from the postulated accidents. However, it may have serious consequences, such as the Fukushima Daiichi accident, where extended SBO after an earthquake, tsunami and floods had not been included as a DBA [1]. There are no firm deterministic and probabilistic criteria for the selection of PIEs. The exclusion or inclusion of a specific event sequence needs to be justified by iteration between the design and analysis, engineering judgement and experience from previous plant design, operation, incidents and accidents. Consequently, context quantification could be used as a systematic probabilistic method for PIEs identification uncertainty reduction that will be exemplified below.

2.4. EOP and SAMG Interaction

The interaction between EOPs and SAMGs also involves shifting of the responsibilities from the main control room (MCR) personnel to the technical support centre (TSC) and the emergency response officer (ERO), who leads the emergency management team. This shifting of the primary responsibility in the SA context, together with the additional obligations for the interfered off-site emergency activities, is also a source of uncertainty. This uncertainty of the SA context (due to an inadequate operation of the control equipment, insufficient information about the accident status and delays caused by the necessity to judge required actions) creates difficulties for decision-makers.

Therefore, SA management requires special advanced design features combined with adequate response and decision-making actions. The post-Fukushima studies, including the European Stress Tests, highlighted the need for SAMG improvement. They should provide a range of comprehensive knowledge, reliable organizational rules and skilled mitigating actions or allow additional evaluation and alternative actions. The separation and interface between SAMGs and EOPs are also critically dependent on the timely SA management, the HOT balance and optimal context because the decision-making process shifts from the MCR to the TSC.

3. SEVERE ACCIDENT CONTEXT CRITERION EXEMPLIFICATION

3.1. Retrospective Analyses of Negative and Positive Scenarios in Real Accidents

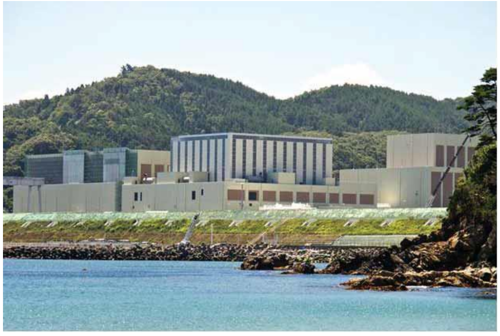
3.1.1. Time Sequences Comparison for the Unit 1 of the Fukushima Daiichi and Onagawa NPSs

The comparison of the time sequences, described in Context Factors and Conditions (CFCs) for Unit 1 of both NPS is shown in Table 1, where CFCs are: E - scenario Event, T - Transient, SF – safety Function, HA – Human Action, UT – Upset Trend, G – Goal and V is Violation (Circumvention).

Table 1: Time Sequences Comparison for the Fukushima Daiichi and Onagawa NPSs

Time	Fukushima Daiichi CFCs [3]	Time	Onagawa CFCs [13]
11.03 14:46	E1: The initiating event was an earthquake with magnitude 8.9 (Richter scale) (0'÷T').	11.03 14:46	E1: The initiating event was an earthquake with magnitude 8.9 (Richter scale) (0'÷T').
	E2: Automatical shutdown (0'÷10').		E2: Automatical shutdown (0'÷10')
	SF1: Reactivity control (0'÷1')		SF1: Reactivity control (0'÷1')
	G1: Cold shutdown (0'÷T').		G1: Cold shutdown (0'÷612').
	SF2: Core Heat Removal (0'÷T').		SF2: Core Heat Removal (0'÷T').
	SF3: Reactor Coolant System Inventory Control (0'÷T').		SF3: Reactor Coolant System Inventory Control (0'÷T').
	V1-T: <u>The circuit breakers and disconnectors in switchyard were damaged</u> (0'÷T') <u>by the earthquake</u> because the Fukushima Daiichi NPS had been designed for magnitude 8.2.		V1-T: There was <u>a failure of 275 kV startup transformer</u> due to a failure in the high voltage circuit breakers caused by the earthquake (9'÷679')
	SF4: Containment Integrity (0'÷T').		SF4: Containment Integrity (0'÷T')

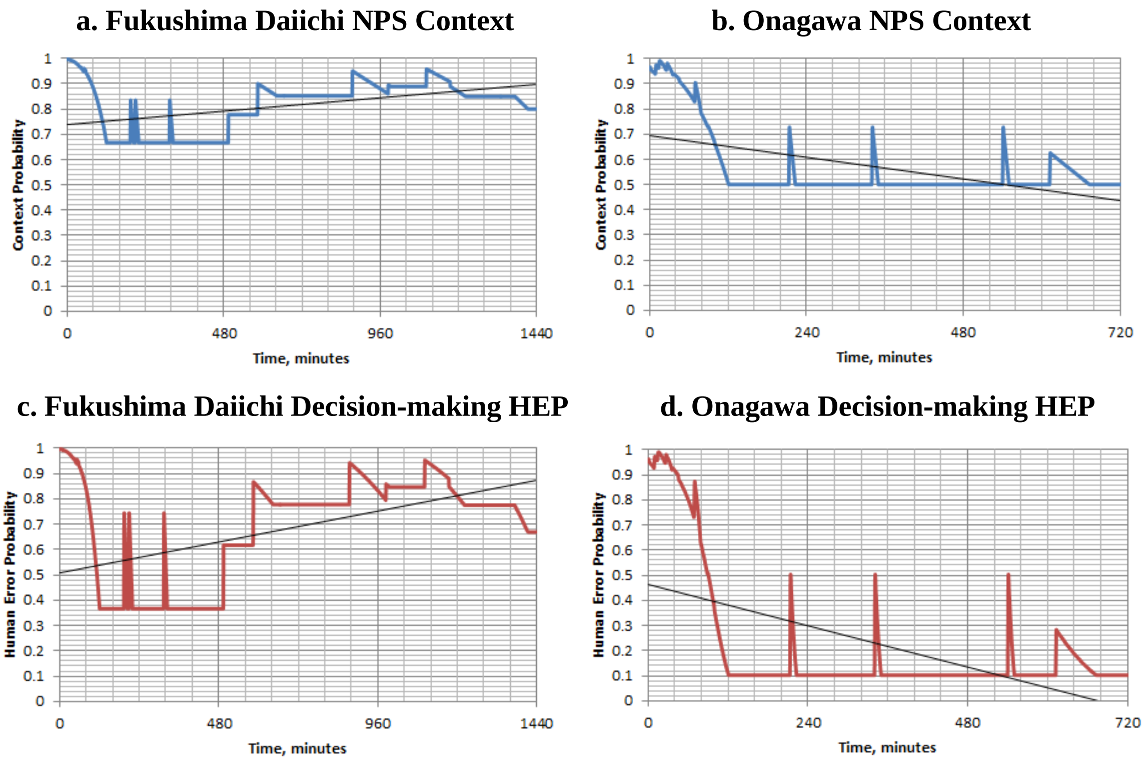
Table 1 (cont.): Time Sequences Comparison for the Fukushima Daiichi & Onagawa NPSs

Time	Fukushima Daiichi CFCs [3]	Time	Onagawa CFCs [13]
11.03 14:46	UT1: Make-up system injected $10m^3/h$ (0'÷T').	11.03 14:55	E3: DGs A & B started and worked properly (9'÷T')
11.03 14.46	UT2: Reactor pressure dropped to 9atm (0'÷719')	11.03 15:00	HA1: The operators started the RCIC system to cool reactor (14'÷24').
	UT3: Containment pressure started to increase (to 9.41atm) (0'÷719')		HA2: They used the safety relief valves to control reactor water level and pressure (14'÷24').
11.03 14:47	E3: 2 Diesel-Generators (DG) started and worked properly (0'÷109')		G2-SF2: RCIC is in operation for Core Heat Removal (14'÷270')
	UT4: High Pressure Coolant Injection injects water from condensate storage tank (0'÷109')	11.03 15:01	HA3: The operators manually depressurized the reactor (15'÷45').
	T1: SVs are opened, RPV is connected with primary containment (0'÷11')		HA4: The RHR system (pump A) was manually started (15'÷25').
11.03 14:52	G2-SF2: Isolated condenser in operation for Core Heat Removal (6'÷514')	11.03 15:12	HA5: The RHR system (pump B) was manually started (26'÷36').
11.03 15:27	E4: Tsunami >10m (analytical value ~13m) (41'÷51')	11.03 15:21	E4: Tsunami <13.8m (analytical value ~13.6m) (35'÷45')
11.03 15:37	E5: 2 DG stopped and failed (51'÷57')	11.03 15:55	HA6: The RHR system (pump D) was manually started (69'÷79').
11.03 15:37	E5: 2 DG stopped and failed (51'÷57')	11.03 16:15	E5: RHR pumps A & B failed. (89'÷99')
11.03 15:42	V2-T: DG on a basement submerged by Tsunami because the Fukushima Daiichi NPS had been designed for 6.51m. The total station black-out was for Units 1-6 up to March 18th (56'÷T')	11.03 18:29	HA7: By Main Steam Relief Valve the reactor pressure was decreased (213'÷223').
		11.03 19:30	E6: The system RCIC stopped automatically (274'÷284').
11.03 16:36	E6: Batteries depleted	11.03 21:56	HA8: Pump A for hydraulic control of reactor water supply system was started by operators (340'÷350')
	UT5: HPIS fails (110'÷120')		
11.03 18:10	HA1: Valves 2A and 3A of IC were opened (194'÷209')	11.03 23:46	HA9: The operators restarted pump A of the RHR system (540'÷550').
11.03 18:25	HA2: Valve 3A of IC were closed manually (209'÷219')	12.03 00:58	E6: Cold shutdown, G1 is fulfilled (612'÷T')
11.03 21:30	HA3: Valve 3A of IC were opened manually (314'÷324')		*Onagawa Nuclear Power Station* 
11.03 23:00	V3-E: Doses increased in turbine building - potential leaks via steam lines (MSIVs, SDS-C, stop valves, etc.) (494'÷T')		
12.03 00:30	V4-SF2: Decay heat is being removed only through isolated condenser. Assumed to be inefficient after 00:30 JTC 12 March due to tanks depletion (584'÷1174')		
	V5-SF4: Possibility of 600kPa in CV dry well (Design basis: 427 kPa) (584'÷2529')		
	G3-SF4: To keep containment integrity (584'÷2529')		

3.1.2. Comparison of the Positive and Negative Experience after the Great East Japan Earthquake

The comparison of the HOT context profiles during the first day after the Great East Japan Earthquake for the Fukushima Daiichi NPS, Unit 1 [3] and the Onagawa NPS, Unit 1 [13] is made. Some data concerning the negative and positive experience of the installations with similar design, processes and risk assessments (assumptive frequency and impact) were obtained. Both NPSs have units with BWR (Boiling Water Reactor) but the Fukushima Daiichi designs are older. The results for the decision-making context probabilities (CPs) and human error probabilities (HEPs) are shown on Figure 1.

Figure 1: CPs and HEPs for the Fukushima Daiichi and Onagawa NPSs during the first 12 hours after the Great East Japanese Earthquake on March 11th 2011.



As seen from the CPs and HEPs calculations, the results for the Fukushima Daiichi case are very high and the tendency for it is to increase. In this situation the context is quite difficult and a successful decision-making is improbable. In the Onagawa NPS case, the CP is not so high and the situation is manageable. The basic reasons for the disadvantages of the Fukushima Daiichi NPS are the violations of the accident context that are underlined in Table 1. They are connected with the following causes:

- Inappropriate design basis (DB): 1) Earthquake with magnitude 8.9 but DB is 8.2; 2) Tsunami wave height is about 13m but DB is 10m; 3) Diesel Generators are submerged because DB is 6.51m; 4) The decay heat removal by the isolated condenser is short-range by DB than necessary in this accident context; 5) The DB pressure for over-pressurizing in CV dry well is less than necessary in this accident context.
- Some smaller design margins, inceptive ageing or fragility of the Fukushima Daiichi equipment could be also noticed, compared to the Onagawa equipment: damaged circuit breakers, disconnectors in switchyard or leaks via steam lines.
- It can be also seen that the Onagawa NPS personnel had taken more timely HAs to mitigate the accident because the context was more favourable and more equipment was available.

3.2. Severe Accident Simulations and Management

The following two scenarios for VVER-1000/V320 were identified for simulation and exemplification of the timely SA management in burdensome decision-making context:

- LB LOCA with simultaneous loss of all alternating current (ac) power supply sources (Station Blackout – SBO) – LB LOCA SBO;
- Long-term (LT) loss of all ac power supply sources postulated in the beginning of the transient simulation – LT SBO.

The accident sequences and the phenomena in the simulated SA scenarios bring serious challenge to the safety barriers and HAs in case of emergency. They provide good highlights to the highly uncertain environment for decision making during the SA progression. Some more details regarding the SA simulation are presented below.

3.2.1. Assumptions, System Configuration, Initial and Boundary Conditions

With regard to the simulation the following assumptions are adopted: 1) End of the 19th fuel cycle; 2) Total loss of ac power supply at time "0.0s"; 3) Main coolant pumps (MCP) operate till the loss of power supply event; 4) Normal operation systems are inoperable; 5) Active safety systems are inoperable; 6) Reactor power is nominal; 7) All of the four hydro-accumulators are available and starts injecting when the respective pressure is achieved; 8) Containment design pressure - 5 bars.

The main parameters that characterize the initial conditions of the reactor installation are for the nominal power. The boundary conditions and system/component configurations (availability) for both scenarios are the same. All active systems powered by ac power supply are failed. Only equipment powered by batteries is available and it can be opened remotely from the MCR: pressurizer safety relief valves (PSRV), emergency off-gas system, steam dump device to atmosphere (SDD-A), safety injection tanks (SIT), steam generator (SG) safety valves (SV), SG isolation valves and turbine generator control and isolation valves. For the LT SBO accident the following HAs are added:

- The PSRV is opened by the operator when the temperature at the exit of the reactor core exceeds 650oC;
- All SG SV are permanently opened by the operator after the PSRV opening.

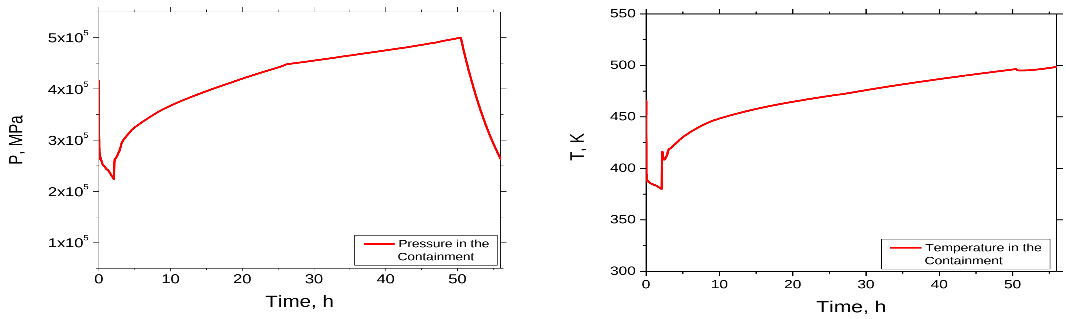
3.2.2. Large Break LOCA and Station Blackout

The main results for LB LOCA with Dn=850 mm with simultaneous SBO are presented below. It is assumed that the break is initiated at the beginning of the transient simulation at the cold leg. Due to the loss of power supply the active systems are inoperable and only passive engineered features can be used for the accident mitigation.

The containment pressure response is presented on Figure 2a. When the break is initiated, a lot of water and steam mixture propagates from the primary circuit to the containment and the pressure increases very rapidly up to 4.1 bars. Later active steam condensation on the containment walls takes place and the pressure decreases up to 2.3 bars till the moment when the reactor vessel (RV) fails.

Figure 2: Containment Pressure and Temperature Response during LB-LOCA SBO

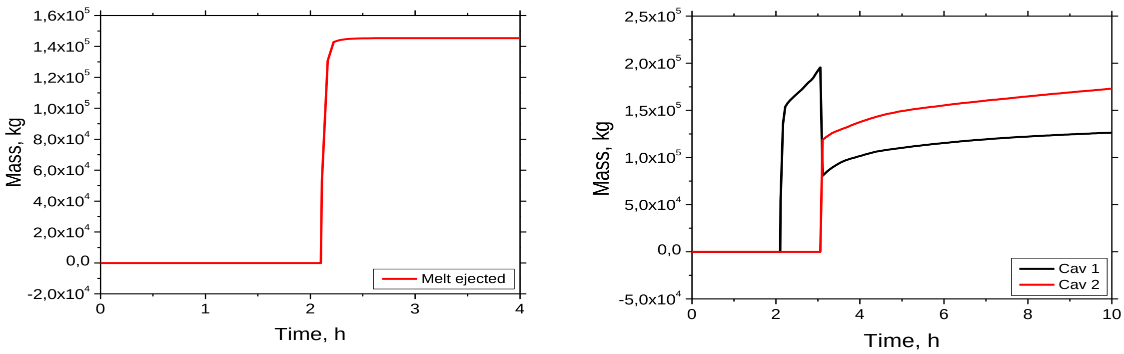
a. Pressure in the Containment **b. Temperature in the Containment**



As a result of the loss of cooling the reactor starts heating up and different processes (ballooning and rupture of the fuel elements, zircaloy oxidation, melting and relocation of structural elements, melting and relocation of the fuel cladding, fuel melt and relocation) occur that lead to core damage and melt formation. The molten material is transferred to the reactor downer part and comes in contact with the RV walls. As a result a pressure and thermal attack is initiated which lead to thinning of the walls and breach formation – the reactor pressure vessel failure occurs at 2.2 h after the initiating event and big amount of molten material (about 150t) is ejected to the cavity – Figure 3a. The pressure in the containment starts increasing again (Figure 2a) due to the steam and the non-condensable gases generated by the molten core concrete interaction (MCCI) between the corium and the concrete of the reactor cavity. The heat radiation from the corium itself and the decay heat of the spread radioactive material are also a serious contributor to the temperature increase. The containment pressurization continues more than 54 hours, until the moment when the containment design pressure (5 bars) is achieved and the containment filtered venting system is expected to actuate. The temperature profile (Figure 2b) follows the pressure in the containment – it is 310°K at the beginning of the accident, increases up to 420°K when the primary inventory is discharged and then starts decreasing (till 370°K) due to the condensation processes. At the end of the simulation (at pressure 5 bars) the temperature in the containment is about 480°K.

Figure 3: Melt Ejection during LB-LOCA SBO from the RV (a) and Transition of Molten Material from the Cavity to the Second Compartment (b).

a. Melt Ejected **b. Transited Molten Material**



For the simulation of the corium behavior in the cavity, it is considered that the molten material consists of stratified layers – heavy oxides at the bottom, metal layer in the middle and light oxides on the top. This configuration provides additional conservatism (in comparison to homogeneous and mechanistic mixtures) to the calculation due to the more intensive heat transfer and the MCCI, respectively. It leads to a more rapid increase of the pressure and temperature in the containment during the ex-vessel phase of the accident. The width of the cavity walls is modeled to be 3.2 m and the thickness of the basemat concrete – 3.6 m. During the course of the simulation the axial and radial concrete ablation remain below the thickness of the walls and there is no penetration through the cavity. However, if the molten material is not cooled and stabilized properly, a basemat melt through may appear (in case the accident progression continues more than few days).

It is possible for a part of the molten material to be transferred to the second adjacent compartment in order to be spread on a larger surface. The cavity and the second compartment are separated by metal wall, which is expected to fail under the high pressure and temperature in the cavity. This case is demonstrated on Figure 3b, where the door fails about 1h after the ejection of the molten material from the reactor to the cavity (approximately 3.2h after the initiating event). The black line shows the mass of the molten material in the cavity, which is about 150t after the vessel failure but due to the MCCI slowly increases. The red line demonstrates the mass of the melt in the second adjacent compartment which is initially zero, but when the door is penetrated, a lot of molten material is transferred. It should be noted that an uncertainty exists regarding the exact behavior of the molten material in the cavity, and especially the exact configuration and the viscosity change as a result of the concrete ablation.

The time sequence during the first 56 hours of the accident progression, together with the relevant CFCs, is presented in Table 3.

3.2.3. Long-Term Total Station Blackout

It is assumed that the loss of all ac power supply sources is initiated at the beginning of the simulation (moment 0.0s). All active systems are inactive, the plant relies on the passive safety systems and only dc power supply is provided by the batteries. An operator action for decreasing the primary circuit pressure through opening the PSRV valve is simulated as a strategy for avoiding reactor pressure vessel failure at high pressure, which potentially may lead to a direct containment heating. It decreases also the risk from steam generator tube rupture later during the accident, when the water level in the secondary sides of the SG drops below the tube bundle. The strategy is in accordance with the SAMGs. The HAs for opening the SG SV of the steam generators are also simulated.

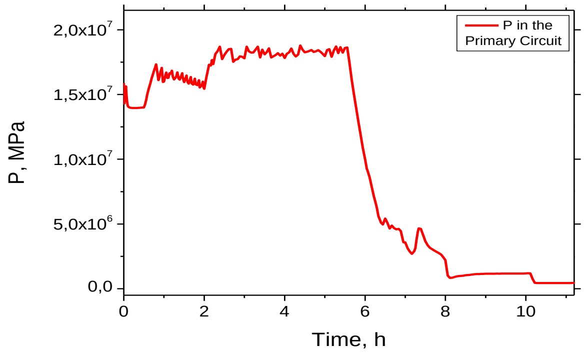
Table 3: Time Sequence in the VVER-1000 LB LOCA SBO (T=3377min)

Duration, min	Context Factors and Conditions
0÷2	E1: External initiating event – an earthquake with magnitude X (Richter scale).
0÷2	T1: Total Station Blackout (SBO)
0÷6	T2: LB LOCA.
0÷T	V1-T1: SBO: The circuit breakers and disconnectors in switchyard were damaged by the earthquake because the NPP is designed for magnitude Y<X
0÷T	V2-T2: LB LOCA: The primary circuit is connected to the containment.
0÷6	E2: Automatical shutdown
0÷(5)	SF1: Reactivity control
0÷T	G1: Cold shutdown.
0÷T	SF2: Core Heat Removal
0÷T	SF3: Containment Integrity
0÷1	E3: The 4 hydro-accumulators inject water into the primary circuit
1÷T	E4: The hydro-accumulators are empty
127÷142	UT1: No water inventory in the RV
127÷T	V3-UT1: RV failure
127÷T	UT2: The containment pressure increases
127÷157	T3: About 140t of molten material is transferred from the RV to the cavity
127÷T	V3-T3: About 140t of the molten core concrete interaction in cavity 1
187÷T	E5: Cavity door failure
187÷T	V4-E5: Start of the molten core concrete interaction in cavity 2
3017÷3377	UT3: Containment temperature at the venting start-up, 223 ºC
3017÷T	UT4: Containment atmosphere concentration at the venting start-up - Steam = 53.3%, H2=13.6%, O2=4.8%. The containment pressure achieved 5 bars.
3017÷3377	E6: Containment venting
3377÷T	E7: End of venting

The pressure in the primary circuit is shown on Figure 4 – after the initiating event the pressure in the primary circuit starts increasing but a natural circulation is established and the decay heat is removed through the secondary side by the operation (opening and closing) of the SDD-A. The SDD-A is cycling during the first 30 minutes, and then it is stuck in closed position because of depleted batteries. As a result, the primary and secondary pressures continue increasing. In about 50 minutes the pressure on the secondary side achieves the set points for opening the safety valves of the steam generators (SG SVs). The SG SVs start cycling and the pressure in the primary circuit is decreased by upward and downward deviations till 16.5 MPa about two hours after the initiating event (Figure 4). However, the active evaporation and the number of steam releases (firstly by SDD-A and later by the SG SV) leads to low water level in the steam generators and hence to worsening of the heat transfer between the primary and the secondary circuit. As a result, a rapid increase of the primary pressure is observed (Figure 4) – the pressurizer safety relief valve (SRV) is actuated and the steam is released into the containment (198 min). The SRV opens when the pressure is 18.3 MPa and closes in 17.4 MPa.

However, 315 minutes after the beginning of the accident the temperature at the exit of the reactor core is measured to be more than 650°C, which is one of the criteria for transition from EOPs to SAMGs. In this moment the operator opens permanently the SRV as a part of the strategy for depressurization of the primary circuit and in order to avoid high pressure melt ejection. A few minutes later the operator opens the safety valves of the steam generators, too. The large amount of released steam leads to a rapid pressure decrease, which allows the hydro-accumulators to start injecting into the primary circuit (pressure below 5.8 MPa). The hydro-accumulators are not emptied at once but for a certain period of time due to pressure fluctuations caused by the evaporation and condensation of the injected boric solution (Figure 4). As a result, the fuel is damaged and a big part of the core is melted and relocated in the downer part of the reactor – in about 10 hours the reactor pressure vessel fails and the molten material is transferred to the cavity, where the MCCI is initiated.

Figure 4: Primary Circuit Pressure during LT SBO

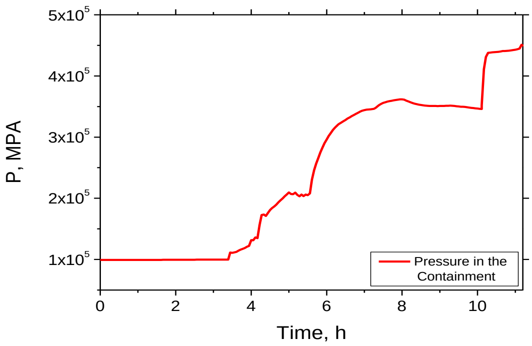


The exact moment of the decision about the depressurization of the reactor coolant system is of high importance in view of the fact that this strategy ensures rapid and efficient reduction of the pressure, on the one hand, but the release of a big amount of coolant inventory may lead to earlier vessel failure, from the other hand. In this simulation a scenario for rapid depressurization was chosen, however, it is also possible that the ERO and the emergency management team have to decide on the type of the depressurization depending on their expectations regarding the actions for recovery of the functions. Moreover, the moment may vary depending on the communications with and within the team.

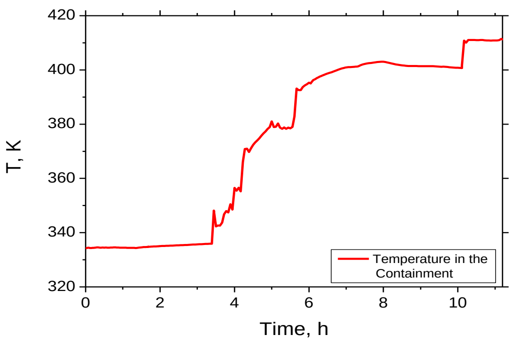
The containment pressure (Figure 5a) is a function of the in-vessel processes and the SA mitigation actions. The pressure starts increasing after the first opening of the PSRV and a second rapid increase is caused by the injected boric solution from the hydro-accumulators. After the melt ejection (approximately at the 10^{th} hour), a third phase of rapid pressurization can be distinguished. The simulation is terminated in 11.2 hours at containment pressure 4.55 and the further behavior will be a competition between the generation of gases (during the MCCI) and steam condensation. The temperature in the containment (Figure 5b) follows a similar profile as the pressure. At the end of the calculation a temperature of 412 K is achieved.

Figure 5: Pressure (a) and Temperature (b) in the Containment during LT SBO

a. Pressure in the Containment



b. Temperature in the Containment



The time sequence during the first 13 hours of the accident progression, together with the relevant CFCs, is presented in Table 4.

3.2.4. Comparative Analysis of the LB LOCA SBO and LT SBO Simulations

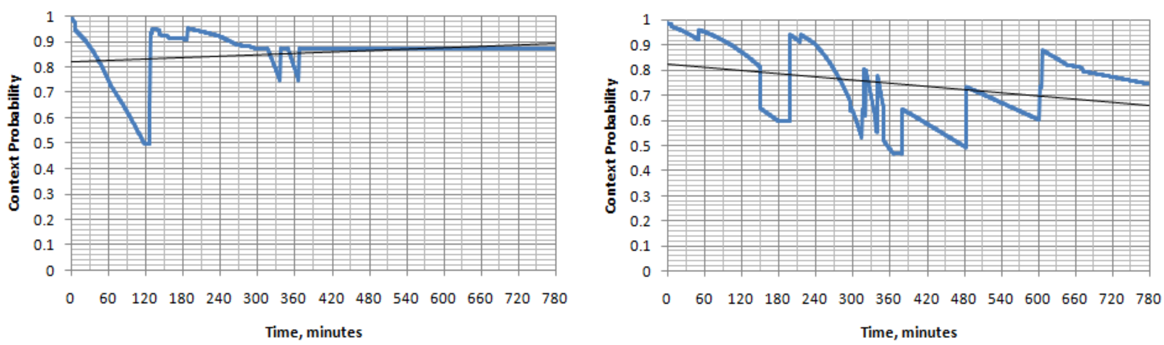
The CP regarding the LB LOCA with SBO and the LT SBO progressions are presented on Figure 6 during the first 13 hours.

Table 4: Time Sequence in the VVER-1000 LT SBO (T=672min)

Duration, min	Context Factors and Conditions
0÷2	E1: The initiating event was an earthquake with magnitude X (Richter scale).
0÷2	T1: Station Blackout (SBO)
0÷T	V1-T1: Loss of all AC power supply sources: The circuit breakers and disconnectors in switchyard were damaged by the earthquake because the NPP is designed for magnitude Y<X
0÷6	E2: Automatical shutdown
0÷T)	SF1: Reactivity control
0÷T	G1: Cold shutdown.
0÷T	SF2: Core Heat Removal
0÷T	SF3: Containment Integrity
2÷30	E3: SDD-A opening (cycling)
2÷327	G2-SF2: To manage the SF2 -Core heat removal
30÷T	E4: SDD-A stuck in a close position
50÷150	E5: The SG SV start cycling
150÷T	V2-SF2: Low level in the SG. The heat removal is not effective through the SGs
198÷327	E6: The PSRV start cycling in order to decrease the primary pressure.
198÷606	T1: The RV is connected with the containment (due to the PSRV opening)
198÷T	UT1: The containment pressure increases
215÷372	UT2: The RV collapsed level drops
315÷T	UT3: The coolant temperature at the reactor core exit is 650°C
318÷320	HA1: Disabling of the interlocks related to the PSRV manual opening
321÷337	HA2: PSRV was opened by the operator
340÷350	HA3: The SG SV is opened by the operator in order to decrease the pressure on the secondary side
372÷380	UT4: The reactor vessel is empty
380÷483	E7: The 4 hydro-accumulators inject water into the primary circuit
483÷T	E8: The hydro-accumulators are empty
483÷T	E9: No water inventory in the RV
602÷T	E10: All batteries depleted
602÷ T	V3-E10: Batteries depleted
606÷T	UT4: No water inventory in the RV
606÷672	V5-UT4: RV failure
606÷672	T2: About 140t molten material is transferred from the RV to the cavity
606÷672	V6-T2: About 140t molten core concrete interaction in cavity 1
672÷T	E11: The containment pressure is 4.65 bar; Hydrogen concentration = 4.1%

Figure 6: Contexts for the LB LOCA SBO and LT SBO Cases

a. LB LOCA SBO Context b. LT SBO Context



As seen from the CP calculations, the results for LB LOCA with SBO case are very high and the tendency for it is to increase. In this situation the context is quite difficult and a successful decision-making is improbable. On the base of previous retrospective accident analyses, if CP>0.6, then no reasonable or adequate HAs could be planned or fulfilled [3].

In the LT SBO case, the CP is not so high at the beginning of SA (10 hours) and the situation could be manageable. The fulfilled HAs are timely, however, if the SBO continues more than 10 hours, the CP also becomes very high and the tendency is going to increase. Consequently, no more reasonable HAs are appropriate, and if any are taken, they may contribute in a negative way.

4. CONCLUSIONS

Two real and two simulated SA were analysed and a methodology for quantification of the HOT context was applied.

The quantified SA CP may serve as an additional criterion for the DBA PIE list extension and screening. They can be used as a base for timely implementation of HAs for mitigation of the SA consequences and for verification of SAMGs.

The overall application of the engineered safety features and the management strategies should be considered in terms of the HOT context (violations reduction) in order to choose the most effective and correct approach.

The dynamic context quantification (together with DSA and PSA models) enables us to find, interpret and assess the characteristics of the decision-making process so that we could improve accident procedures, invite new safety measures and open more opportunities to manage accident situations.

References

[1] J. Chao and D. Johnson. "*A fundamental cause for the Fukushima accident and a proposed effective remedy*", Proceedings of PSAM Topical Conference in Tokyo, PSAM2013-1097, (2013).
[2] ISO 31000:2009. "*Risk Management – Principles and Guidelines*".
[3] G. Petkov and E. Kostov. "*Risk resonance context in nuclear facilities*", Proceedings of PSAM Topical Conference in Tokyo, PSAM2013-1029, (2013).
[4] G. Petkov and E. Kostov. "*Fukushima Daiichi Nuclear Accident Context Research-In-Depth,"* Proceedings of IAEA-CN-212, Vienna, (2013).
[5] G. Petkov. "*How to Elucidate and Judge Shadow of Nuclear Safety Culture,"* ANS Transactions, Washington, (2013).
[6] G. Petkov. "*Identification and Fitting of Human Cognition Model*," Proceedings of the ASME-PVP Conference, Vol. 400, pp. 35-48 (2000).
[7] G. Petkov, V. Todorov, T. Takov, V. Petrov, K. Stoychev, V. Vladimirov and I. Chukov. *"Safety Investigation of Team Performance in Accidents,"* Journal of Hazardous Materials, Vol. 111, pp. 97-104, (2004).
[8] E. Hollnagel. "*Human Reliability Analysis: Context and Control,*" Academic press, 1993, London.
[9] A. Spurgin and G. Petkov. "Advanced Simulator Data Mining for Operators' Performance Assessment," *"Intelligent Data Mining: Techniques and Applications"*, Physica Verlag - Springer, pp.543-572, 2005, Heidelberg.
[10] G. Petkov. *"Dealing with Dynamic Aspects of Operators' Performance,"* Proceedings of the FLINS'2004 Conference *"Applied Computational Intelligence,"* World Scientific, ISBN 981-238-873-7, pp. 677-683, (2004).
[11] G. Petkov, P. Antão and C. Guedes Soares. "*Context Quantification of Individual Performance in Accidents,"* Proceedings of ESREL'2001, vol. 3, pp. 1851-1858, (2001).
[12] IAEA NS-R-1. "*Safety of nuclear power plants: design: safety requirements*." - International Atomic Energy Agency, 2000, Vienna.
[13] IAEA Mission Report. "*IAEA mission to Onagawa nuclear power station to examine the performance of systems, structures and components following the Great East Japanese Earthquake and Tsunami*." - International Atomic Energy Agency, 2012, Vienna.