

Decision Support Tool for Severe Accident Management

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ABSTRACT

The project NARSIS – New Approach to Reactor Safety ImprovementS – has made scientific steps towards addressing the update of some elements required for the safety assessment of nuclear power plants. These improvements mainly concern:

- Natural hazards characterization, in particular by considering co-current external events, either simultaneous-yet-independent hazards or cascading events, and the correlation in intra-event intensity parameters.
- Vulnerability of the elements to complex aggressions, with the integration of new approaches such as vector-based fragility surfaces and reduced models.
- Better treatment of uncertainties through adoption of probabilistic framework for vulnerability curves and non-probabilistic approach to constraining the “expert judgments”.
- Development of decision support tool for severe accident management.

The decision support tool for severe accidents – called Severa – has been developed in this project. It is a prototype demonstration-level decision support system aimed at supporting the technical support center (TSC) while managing a severe accident. Severa represents, stores and monitors selected physical measurements of the NPP. It assesses the current state of barriers: core, reactor coolant system, reactor pressure vessel and containment. The prediction of future accident progression, if no action is undertaken, is one of basic functions. The support tool provides a list of possible management recovery strategies and courses of action. The applicability and feasibility of possible actions in the given situation is identified. For each action, Severa assesses possible consequences in terms of probability of the last barrier (containment) failure and estimated time

window for failure. At the end, Severa evaluates and ranks the feasible actions, providing recommendations for the TSC. The verification and validation of Severa has been performed in the project and is described in this paper.

Keywords: *Severe Accident Management, Decision Support System, Decision Model*

1 INTRODUCTION

Eighteen academic, research and industrial European institutions from Slovenia (GEN, JSI), Croatia (APOSS), Italy (ENEA, UNIPI), France (CEA, BRGM, IRSN, EDF, Framatome – ex Areva NP), Austria (NUCCON), Poland (NCBJ, WUT), Germany (KIT, Framatome - ex. Areva), Finland (VTT), The Netherlands (TU Delft, NRG) and United Kingdom (EDF Energy) were working on the project NARSIS – New Approach to Reactor Safety ImprovementS [1]. The project was funded by the European Commission and has started in autumn 2017, with the duration of 4,5 years. The main vision of the consortium was to fill some gaps identified in existing external events probabilistic safety analyses (PSA) and to improve parts of existing methodologies by three points:

(1) to adapt most up to date frameworks and methodologies already existing or under development outside of nuclear community;

(2) to use knowledge and experience on recent national and international projects;

(3) to develop demonstration cases at the real NPP scale.

The main results are the development of an integrated risk framework for safety analyses and the development of a decision-support tool for demonstration of nuclear facility management. The integrated risk framework consists of:

- Scenarios comprising single or multiple external hazards. Hazards are combined or cascading and include earthquake, flooding (height of external flooding due to increased river flow or precipitation), extreme weather and others.
- The physical and functional fragilities and interdependencies between systems/equipment are considered with aging effects.
- Human factors are considered and may play important role during severe accidents.
- A decision-support tool is developed to demonstrate nuclear facility management during severe accidents due to external natural events.

The project was structured into seven work packages (WP):

- WP1: External hazards characterisation,
- WP2: Fragility assessment of main NPPs critical elements,
- WP3: Integration and safety analysis,
- WP4: Applying & comparing various safety assessment approaches on a virtual reactor,
- WP5: Supporting tool for severe accident management,
- WP6: Dissemination, recommendation, and training,
- WP7: Project management and coordination.

The work package (WP5) dealt with the development of decision support tool for severe accident management and its demonstration. First, the referential nuclear power plant (NPP) was established [1]. The referential NPP was based on operating power plants in European Union. The safety systems, structures and components (SSC) of referential nuclear power plant include design basis safety SSC, safety SSC to mitigate severe accident and mobile SSC (FLEX equipment). The design basis SSC was installed during the construction of NPP and includes high pressure injection, borated water accumulators, and low-pressure safety injections, to supply cooling water and mitigate loss of coolant accident. Emergency diesel generators and batteries are intended to supply energy for operation of pumps, valves and instrumentation and control. Emergency feed water pumps are intended for reactor core cooling. The safety valves are installed at reactor coolant system to decrease pressure below design value. The containment is a building around the reactor to

confine the radioactive material and prevent releases to the environment and radioactive doses to the public. Alternative energy sources in terms of diesel generators and batteries are included in referential NPP. The passive containment venting is included in referential NPP to decrease the pressure in containment in case of low probable high-pressure scenario. The referential NPP includes passive autocatalytic recombiners to reduce the possibility of hydrogen explosions and alternative depressurization system to have high confidence for depressurization of reactor coolant system.

The severe accident management guidelines (SAMG) applicable to referential NPP were described [2]. Similar guidelines are developed by EPRI, with some differences [3]. In case of deviation of important NPP measurements, alarms go off in the control room and the operators use alarm respond procedures to respond to alarms. If they are not able to successfully correct the situation, they use abnormal operating procedures is envisaged. If the problems still persist and reactor trip is activated, it means that design basis accident is occurring and emergency operating procedures are used to activate safety SSC. If such action is not successful, the core starts to heat up due to decay heat and severe accident with core degradation or melting can occur. The management of NPP is transferred from operators in control room to the technical support center. In order to manage severe accidents, the SAMGs are used by managers in the technical support center. The SAMG includes operations such as:

- Injection to steam generator, to remove decay heat from reactor coolant system (HLA1).
- Depressurization of reactor coolant system, to prevent high pressure melted corium ejection, which can damage containment and causes quick rise of containment pressure and hydrogen generation (HLA2).
- Injection to reactor coolant system, which assures coolant water to reactor core to remove decay heat (HLA3).
- Injection of water into containment, to reduce containment pressure and possible radioactive releases.

In contrast to previously used procedures, where operators followed the procedures line by line and no decisions were needed, in SAMG the technical support center needs to take decisions. There can be large amount of information, some of them available only partially, or with high uncertainty. The technical support center managers are under stress due to a large damage in the NPP, potential releases of radioactivity and time pressure. The decision support tool Severa, developed in the project and described below, targets this accident management stage and aims at supporting the managers to make appropriate decisions with prioritization of actions in a well-justified and timely manner.

2 INPUT DATA

The hazard-induced damage states and specific accident progression event tree for demonstration purposes were developed [4]. This includes developing accident progression logic structure for postulated hazard damage states, where damaged SSC are identified.

For this purpose, two major severe accident sequences were evaluated: high pressure and low pressure sequence. The high pressure sequence starts with an initiating event like station black out (total loss of internal and external electricity power), or loss of ultimate heat sink, where decay heat removal is absent and the depressurization of reactor coolant system fails. The core temperature starts to rise and hydrogen production starts in contact of hot steam and cladding. The core starts to melt and can be ejected, if hot leg creep failure did not occur, to containment with reactor vessel failure at high pressure (High Pressure Melt Ejection (HPME)). The fast transfer of corium heat in containment (Direct Containment Heating (DCH)) threatens containment integrity.

The low pressure sequence starts with an initiating event like loss of coolant accident, where the water in reactor coolant system is lost, and there is no medium to remove decay heat. The containment pressure starts to increase with loss of coolant accident, which can threaten the

integrity of containment. The core temperature starts to rise. The core starts to melt and reactor vessel fails at the bottom. The reactor cavity below the reactor pressure vessel can be flooded with water. Hot corium in contact with water can initiate steam explosions, which can threaten containment integrity. The molten corium interaction with concrete and water starts to produce hydrogen and carbon monoxide (CO), which both can form explosive mixture. Potential hydrogen and CO burn or explosion can threaten the integrity of containment.

Severe accident simulations were performed for each sequence, with different safety features available and different time of activation of safety features [5].

The assessment of potential decisions needed to be taken by technical support center, is provided. The set of attributes against which all decisions are evaluated in decision support process is identified. This includes the status of main barriers, fuel cladding, reactor coolant boundary and containment. Since the status of boundaries (e.g. fuel temperatures) is not measured directly some indirect parameters are used.

Figure 1 presents a simplified severe accident progression, important phenomenology effects for both scenarios (LP and HP) including the comparison of expected (predicted by severe accident simulations with MELCOR code) time windows of each accident phase [5].

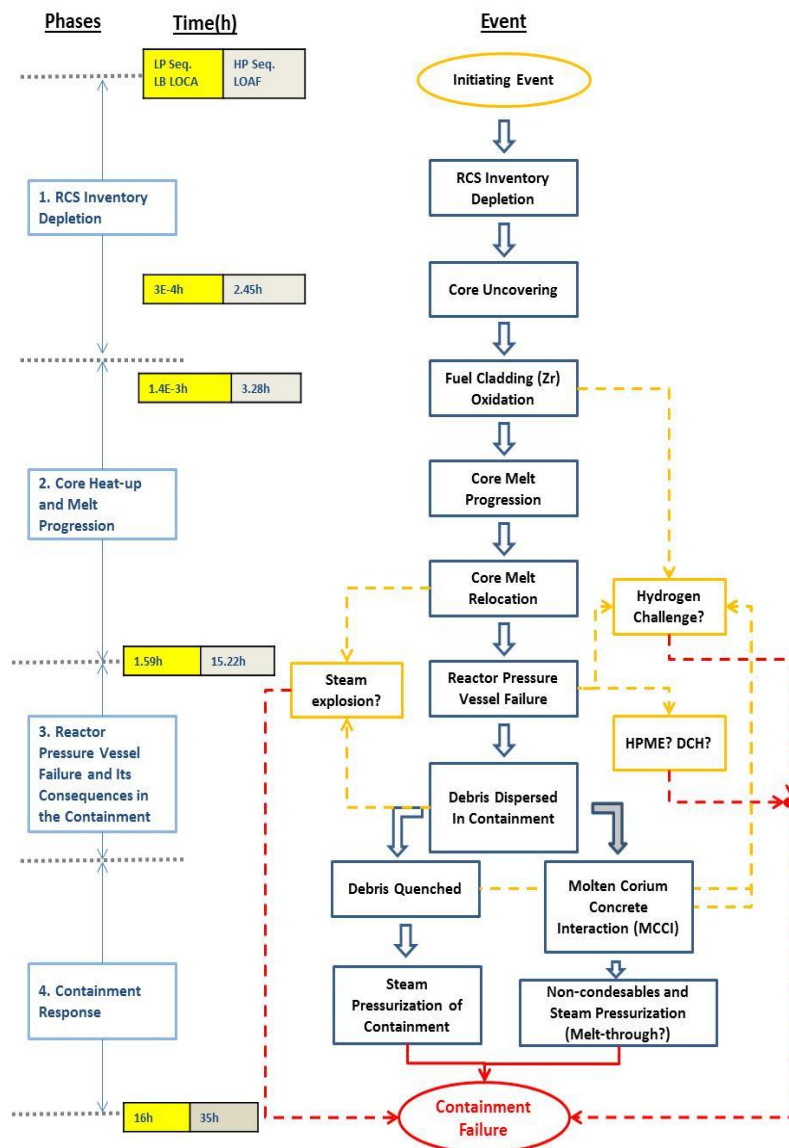


Figure 1: Severe Accident Progression and Phenomenology

3 DECISION SUPPORT TOOL

The decision-support tool called Severa was developed [6]. Severa is a demonstration-level Windows application, aimed at supporting the TSC team while managing a severe NPP accident.

The operation of Severa is based on a time series of eight critical parameters that are periodically measured in the NPP [8]:

- CET: Core Exit Thermocouples [°C]
- SGL: Steam Generator Level [m]
- RPVL: Reactor Pressure Vessel Level [%]
- Prcs: Reactor Coolant System Pressure [MPa]
- Pcont: Containment Pressure [MPa]
- Tcont: Containment Temperature [°C]
- Lcont: Containment Water Level [m]
- H2: Hydrogen concentration [%]

On this basis, Severa makes a first major decision-support contribution by providing the following information to the TSC:

- Whether or not – and when – the conditions in the NPP require the activation of SAMGs?
- Which SAGs (Severe Accident Guidelines) are relevant for the situation? Currently, Severa is restricted to three SAGs: SAG-1 (Inject into SG), SAG-2 (Depressurisation of RCS) and/or SAG-3 (Inject into RCS).
- Given the measurements, what are the expected states of the three barriers: Core, RCS, and Containment?
- What are the expected progressions of the event if no actions are undertaken by the TSC?

The main steps of using Severa are:

1. Monitor operating parameters
2. Access the state of barriers
3. Predict accident progressions
4. Identify recovery actions
5. Assess the feasibility of actions
6. Predict the consequences of actions
7. Assess actions and suggest the best

The final step, selection and implementation of management actions, is on behalf of the technical support center, who are also responsible for repeating the steps until the accident has been resolved.

Figure 2 shows an example of a Severa screenshot that displays the first 200 minutes of a Station Blackout event (simulated with MELCOR) and Severa interpretation of time series in terms of:

- Columns CET to Lcont: Color-coded interpretation of individual measurements, where white, yellow, orange and red colors indicate the states of increasing severity, and magenta indicates an out-of-range or erroneous measurements (the real measurements in NPP have specified range, however the MELCOR simulation results are not limited by measurement range).
- Column SAG: Shows SAGs relevant for the situation (multiple SAGs are possible).
- Column Seq Type: Sequence type, either low-pressure (LP) or high-pressure (HP).
- Columns Core State – Cont State: Assessed current state of the three barriers.

- Column Possible Progressions: Prediction of possible events if no actions are undertaken.

Time [min]	CET [C]	SGL [%]	RPVL [%]	Prcs [MPa]	Pcont [MPa]	TCont [C]	Loort [m]	H2 [%]	SAGs	Seq Type	Core State	RCS State	Cont State	Possible Progressions
0	330	87.7	100.0	15.51	0.101	22	0.0	0.00			OK	OK	OK	
10	309	59.3	100.0	13.29	0.104	26	0.0	0.00			OK	OK	OK	
20	307	42.3	100.0	12.75	0.107	31	0.1	0.00			OK	OK	OK	
30	307	27.9	100.0	12.23	0.109	33	0.1	0.00			OK	OK	OK	
40	307	16.4	100.0	11.18	0.109	34	0.1	0.00			OK	OK	OK	
50	307	4.9	99.5	9.53	0.110	35	0.2	0.00			OK	OK	OK	
60	318	0.0	99.7	11.23	0.110	36	0.2	0.00			OK	OK	OK	
70	335	0.0	100.0	14.81	0.111	37	0.2	0.00			OK	OK	OK	
80	349	0.0	100.0	17.22	0.111	37	0.3	0.00			OK	OK	OK	
90	354	0.0	67.2	17.03	0.153	76	1.1	0.00			OK	OK	OK	
100	354	0.0	56.5	17.11	0.176	84	1.1	0.00			OK	OK	OK	
110	423	0.0	37.1	17.09	0.178	85	1.2	0.00			OK	OK	OK	
120	677	0.0	27.5	17.08	0.173	82	1.2	0.00	1, 2, 3	High	OK	OK	OK	
130	1,074	0.0	23.8	17.08	0.168	80	1.6	0.00	1, 2, 3	High	OK	OK	OK	
140	1,786	0.0	20.3	17.07	0.183	86	1.6	0.01	1, 2, 3	High	OX	IP	OK	CD, RCSdepr, CH, DCH, Bypass
150	1,525	0.0	13.1	17.15	0.189	87	1.6	0.03	1, 2, 3	High	OX	IP	OK	CD, RCSdepr, CH, DCH, Bypass
160	1,410	0.0	13.1	17.23	0.196	89	1.6	0.03	1, 2, 3	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
170	1,531	0.0	12.5	17.20	0.195	89	1.6	0.03	1, 2, 3	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
180	1,612	0.0	9.0	17.09	0.194	89	1.6	0.03	1, 2, 3	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
190	607	0.0	6.6	16.44	0.189	87	1.6	0.03	1, 2, 3	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
200	179	0.0	33.0	0.30	0.294	113	1.6	0.03	1	Low	CD & OX	IFD	OK	RPVmelt, CH, MCCI

Figure 2: A Severa screenshot showing the first 200 minutes of a Station Blackout event

This information is generated by Severa partly by using decision rules encoded in the software and partly by a qualitative rule-based multi-criteria model [8] developed according to the method DEX[9].

The second major decision-support contribution of Severa is related to possible management actions and their expected consequences. In each situation, multiple actions may be available, but their choice and potential success depend of a variety of factors: preconditions for carrying out an action, current and future availability of equipment, available time window, action adequacy, etc. Actions may be mutually exclusive and the success of some action may depend of the success of another one. The possible actions with availability of equipment are used to define alternatives (Figure 3).

Alternative D						Alternative F					
Use Design-Based Equipment Later						Use Flexible Equipment Sooner					
System	Status	Available at [min]	Path	Elements	Time Window(s)	System	Status	Available at [min]	Path	Elements	Time Window(s)
AFW	Unavailable		SP11	AFW	[36,48]	AFW	Unavailable		SP11	AFW	[36,48]
DECCPRPORV	Unavailable		SP12	SGPORV, DECSG	[36,48]	DECCPRPORV	In 10 min	130	SP12	SGPORV, DECSG	[36,48]
DECSG	In 30 min	150	SP13	DECSG, DECSG	[36,48]	DECSG	Unavailable		SP13	DECSG, DECSG	[36,48]
DECSGPORV	Unavailable		SP14	SGPORV, FLEXSG	[36,48]	DECSGPORV	Unavailable		SP14	SGPORV, FLEXSG	[36,48]
DECSI	Unavailable		SP15	DECSGPORV, FLEXSG	[36,48]	DECSI	Unavailable		SP15	DECSGPORV, FLEXSG	[36,48]
FLEXSG	Unavailable		SP21	SGPORV	[36,193]	FLEXSG	In 10 min	130	SP21	SGPORV	[36,193]
FLEXRCS	Unavailable		SP22	DECSGPORV	[36,193]	FLEXRCS	Unavailable		SP22	DECSGPORV	[36,193]
HPSI	Unavailable		SP23	PRPORV	[36,193]	HPSI	Unavailable		SP23	PRPORV	[36,193]
LPSI	Available	0	SP24	DECCPRPORV	[36,193]	LPSI	Available	0	SP24	DECCPRPORV	[36,193]
PRPORV	Unavailable		SP31	LPSI	[36,193]	PRPORV	Unavailable		SP31	LPSI	[36,193]
RCFC	Unavailable		SP32	DECSI	[36,193]	RCFC	Unavailable		SP32	DECSI	[36,193]
SGPORV	In 30 min	150	SP33	FLEXRCS	[36,193]	SGPORV	Unavailable		SP33	FLEXRCS	[36,193]
SMPDECSI	Unavailable		SP34	HPSI	[36,193]	SMPDECSI	Unavailable		SP34	HPSI	[36,193]
SMPFLEX	Unavailable		SP3R1	LPSI, SMPLPSI	[36,193] [280,900]	SMPFLEX	Unavailable		SP3R1	LPSI, SMPLPSI	[36,193] [280,900]
SMPPLPSI	Available	0	SP3R2	DECSI, SMPDECSI	[36,193] [280,900]	SMPPLPSI	Available	0	SP3R2	DECSI, SMPDECSI	[36,193] [280,900]
SP	Unavailable		SP3R3	FLEXRCS, SMPFLEX	[36,193] [280,900]	SP	Unavailable		SP3R3	FLEXRCS, SMPFLEX	[36,193] [280,900]
			SPC1	SP	[280,900]				SPC1	SP	[280,900]
			SPC2	RCFC	[280,900]				SPC2	RCFC	[280,900]

Figure 3: Definition of alternatives in Severa

Each action has a success window, defined based on the 95th and 5th percentile of success times. The expected action success is estimated by cumulative lognormal distribution depending on T_{05} and T_{95} (Figure 4).

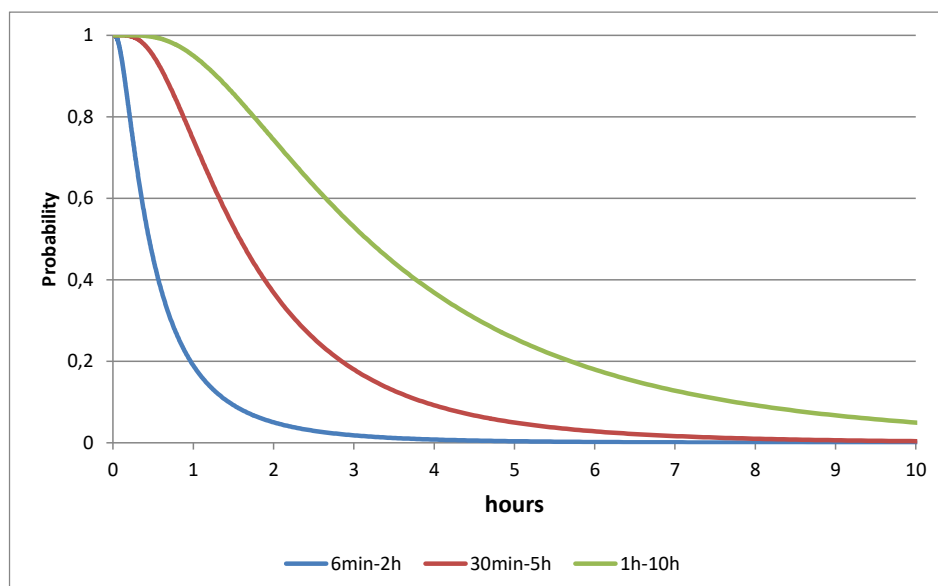


Figure 4: Management actions' success windows in Severa

In Severa, the expected outcome of actions is assessed using a probability distribution of expected radioactive releases with respect to four categories of radioactivity release [8]:

- RC-E: Containment failure with a significant release of radioactivity is expected within several hours.
- RC-I: Containment failure with a significant release of radioactivity within several days.
- RC-L: No significant release of radioactivity is expected within several days.
- RC-N: Long-term concern (in-vessel recovery and/or intact containment).

Figure 5 shows an example of such a probability distribution, assessed by Severa for the situation at the 180's minute of the time series from Figure 2, assuming that all equipment is available and all mitigation actions can be started immediately. Notice that at that time, the station blackout event has been already in progress for about one hour.

Generally, when choosing between alternative actions, the action whose probabilities are the highest around RC-N and the lowest around RC-E is recommended for implementation.

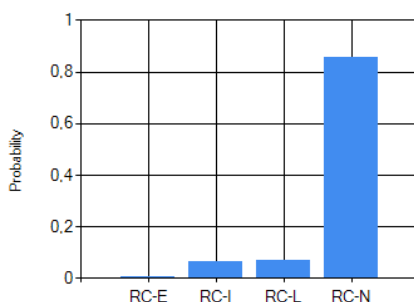


Figure 5: An example of radioactive-release probability distribution

The main model for producing such assessments is based on an accident progression event tree (APET [4], [5]). In Severa, the APET is implemented in terms of an equivalent probabilistic DEX model [8].

It needs to be pointed out that Severa is a proof-of-concept tool which was developed in order to investigate the possibilities of this kind of decision support in severe accident management,

primarily for the training purposes. As any such tool, it has its limitations. Among the most important is a treatment of time dependency of the probabilistic parameters incorporated in its prognostic logic. A number of phenomenological probabilities are presented by values which apply at an early phase of the accident and, therefore, its accurate performance is limited to this time window. Due to the complexity of the process, Severa relies on a simplified representation of logic models for “success paths” and system functions, as well as a simplified consideration of adequacy of equipment included in the model and feedback from the implemented actions. However, even under the limitations, the development of Severa was very challenging and took quite a considerable programming effort. Verification and validation exercises showed that it can provide reasonable predictions of probability profiles of major release categories for the scenarios considered.

4 VERIFICATION & VALIDATION OF SEVERA

4.1 Basic definitions

There are two terms which have very specific meaning in the context of Severa [10] and are, therefore, essential for its verification and validation (V&V) process. They are “Time Delay” and “Alternative”.

“Time Delay” is a user-provided input concerning critical systems / equipment availability. Availability of particular systems and their combinations is defined in terms of a “time delay” (TD), i.e., the time (measuring from the time-point at which the assessment by Severa is being made) at which the respective item is expected to become available. Following are some important points, with regard to V&V:

- There are 20 TD entries (single cases + combinations of systems);
- The designator TD_x represents the time (starting from now (“now” meaning the time-point at which Severa is used)) at which system “x” would become available;
- It is noted that: $0 < TD_x < \infty$:
 - The value “0” means that system (item “x”) is available or is already operating, e.g. as a part of a high level action (HLA) which is under implementation;
 - The upper bound “∞” (or any large value representing the infinity) means that the system is known to be failed beyond repair.

“Alternative” in the Severa terminology represents one specific set (or a “vector”) of 20 values of “time delay” terms discussed above.

For V&V purposes, it is useful to have in mind the format in which the results of the prognostic part of Severa are provided: for each considered “alternative”, Severa provides the conditional probabilities of four mutually exclusive categories of an outcome: RC-E, RC-I, RC-L, RC-N. As the categories are considered mutually exclusive, the four conditional probabilities sum to 1.0.

The results are presented both numerically and graphically. However, just to mention it, there is an issue which makes graphical presentation difficult: Quantitative results (probabilities) appear in the range of 3 or even 4 orders of magnitude, as illustrated by Figure 6. There is a possibility to use logarithmic scale for presentation of the results. However, this can be confusing in a stressful situation and not very suitable for intuitive interpretation of the results.

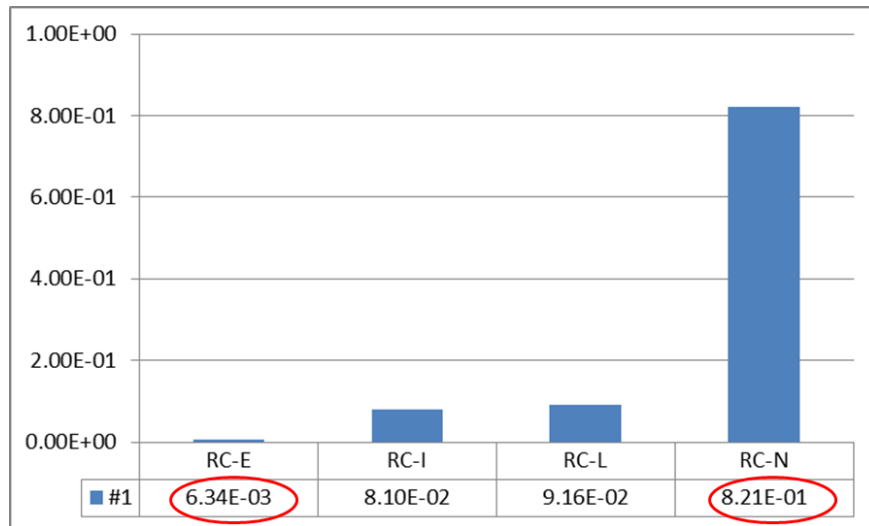


Figure 6: Presentation of results – The range of probabilities of radioactive release categories

4.2 Approach to V&V

Two general aspects of any V&V can be described in a simplified way as:

- 1) **Verification:** check whether the product is in accordance with predefined specifications (“see whether you really got what you wanted”);
- 2) **Validation:** check whether the product is suitable for the intended purpose / application (“see whether you really wanted what you got”).

It is usually the second aspect which is considerably more challenging than the first.

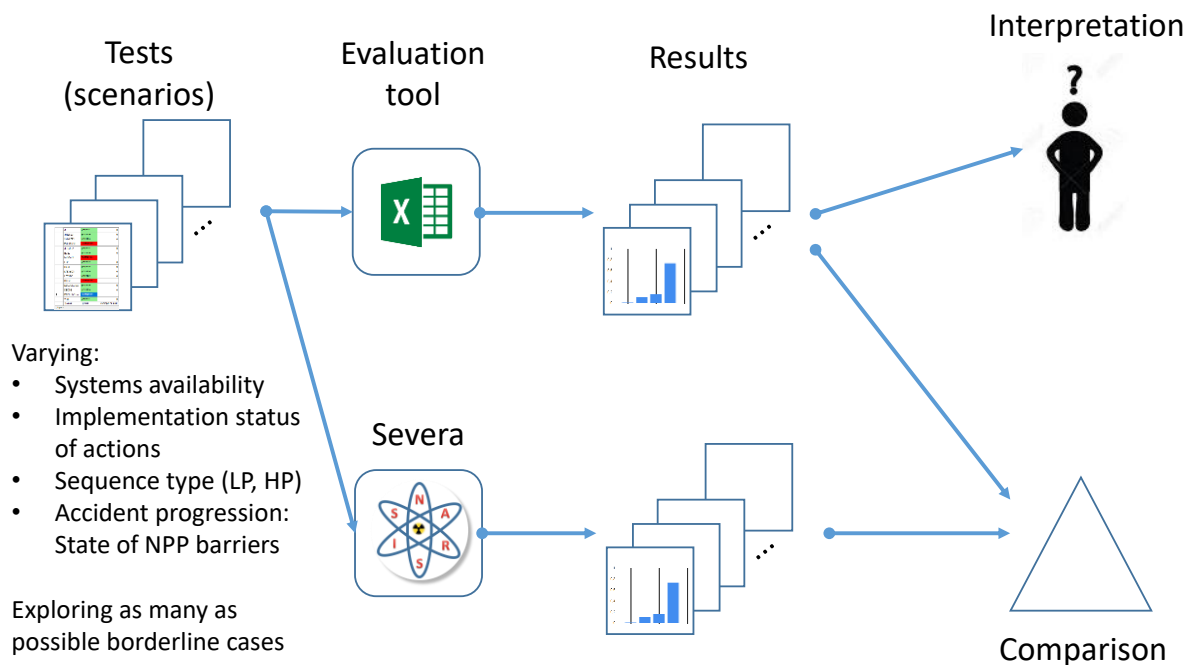


Figure 7: Approach to verification and validation of Severa

The first step of the overall V&V of Severa was the deterministic verification and validation of possible recovery actions. It should be noted that a number of accident sequences were studied by MELCOR deterministic analyses ([5]) to evaluate phenomenological aspects of severe accidents, timing of important consequences without any recovery (operator) actions and success of performed recovery actions. According to [4] and [10], each SAG (SAG-1, SAG-2, SAG-3) is associated with

a corresponding High-Level Action (HLA): HLA1, HLA2, HLA3. Each HLA contains several Success Paths (SPs), i.e., alternative and mutually exclusive ways of responding to the accident. Each SP uses one or more systems from the inventory of plan systems, such as pumps and power generators, which must be available and in a working condition in order to pursue the SP.

Two general types of severe accidents scenarios studied in[5]: high pressure scenario and low pressure scenario. These two types generally differ with regard to the pressure behaviour in the Reactor Coolant System (RCS) following the assumed initiating event (IE). For each of them, a number of deterministic analyses by MELCOR were performed.

The second part of the overall V&V process for Severa was established along the following lines (Figure 7):

- Verification part for Severa tool. It was rather straightforward and consisted of the following activities:
 - Define test cases involving different formulas embedded in the tool;
 - Pre-calculate results independently (externally to Severa), e.g., by a spreadsheet;
 - Perform runs by Severa and compare;
- Validation part for Severa tool:
 - Define test cases for different conditions predictable by supporting analyses or knowledge / experience;
 - Define the expectations with regard to results. Those were related to likelihood profile of containment failure / release categories;
 - Calculate the results and interpret / evaluate them against the expectations;
 - Do also sanity checks against other test results;

The procedure which was followed for a particular test can be summarized with the six steps:

1. Define the test case;
2. Describe the expectations concerning the results;
3. Pre-calculate results independently;
4. Evaluate results against expectations and against other relevant tests under V&V;
5. Obtain corresponding results by Severa and compare against step 3;
6. Do any adjustments or corrections, if needed.

It should be noted that both verification part as well as validation part have resulted with certain (mostly although not necessarily minor) corrections and adjustments of tests and Severa itself. It also should be noted that a considerable number (275) of test cases were done and passed successfully.

To illustrate the process, we present an example involving a group of rather simple test cases / subcases.

4.3 Example of a V&V case

Checks with Different Alternatives ($TD_x > 0$). No Action under Implementation. Time Point: Entering SAMGs.

Time point is set shortly after reaching $CET = 650^\circ C$ (time point = 126 min with the Station Blackout time series). Tested is a set of alternatives with different TD_x terms for specified functions. In all cases:

- TD_x large (TD goes to infinity) is simulated with $TD_x = 60000$ min;
- $TD_{xy} = TD_x + TD_y$; (TD terms for combinations);
- »Under Implementation« = 0 for each HLA / Function. No function is under implementation.

Case 1.0. Zero Alternative, A0: All TD_x large

- **Zero alternative:** no function available and no actions will be taken (no recovery).
Therefore: all TD_x large

- **Expectation:** Release: RC-E if SG creep rupture, or RC-I if no SG creep rupture. If no SG creep rupture, Containment is expected to fail in intermediate time window due to mass and energy release (MER) challenge.
- **Results:** considered OK. Reproduced by Severa OK.

RC-E	RC-I	RC-L	RC-N
1.06E-02	9.89E-01	0.00E+00	0.00E+00

Case 1.1. Comparing Different Subcases with Availability (at TD = 0) of HLA1, HLA2 and / or HLA3. (No Containment Heat Removal)

Subcase 1.1.1: HLA1 (Inject to SGs)

- **Expectation:** This HLA can address SG creep rupture and reduce the likelihood of early release (RC-E). However, it cannot address containment challenge in later time frames. Primary inventory will be lost through RCP seals and PORVs and VF and Containment challenge cannot be avoided and is expected at intermediate time frame. Therefore: RC-E decreases on account of RC-I. RC-L = RC-N = 0.
 - The best option is AFW (smallest RC-E). For other options RC-E increases.
- **Results:** OK (Table 1). Reproduced by Severa: OK. It is noted that graphical presentation is not very useful for comparing cases like these (Figure 8), because the results may cover the range of several orders of magnitude.

Table 1: V&V example (Subcase 1.1.1)

#	TDx = 0	RC-E	RC-I	RC-L	RC-N
0	Zero (All TD>>)	1.06E-02	9.89E-01	0.00E+00	0.00E+00
A	(AFW)	1.50E-03	9.99E-01	0.00E+00	0.00E+00
B	(SGPORV) (DECSG)	1.90E-03	9.98E-01	0.00E+00	0.00E+00
C	(DECSGPORV) (DECSG)	2.40E-03	9.98E-01	0.00E+00	0.00E+00
D	(SGPORV) (FLEXSG)	2.70E-03	9.97E-01	0.00E+00	0.00E+00
E	(DECSGPORV) (FLEXSG)	3.30E-03	9.97E-01	0.00E+00	0.00E+00

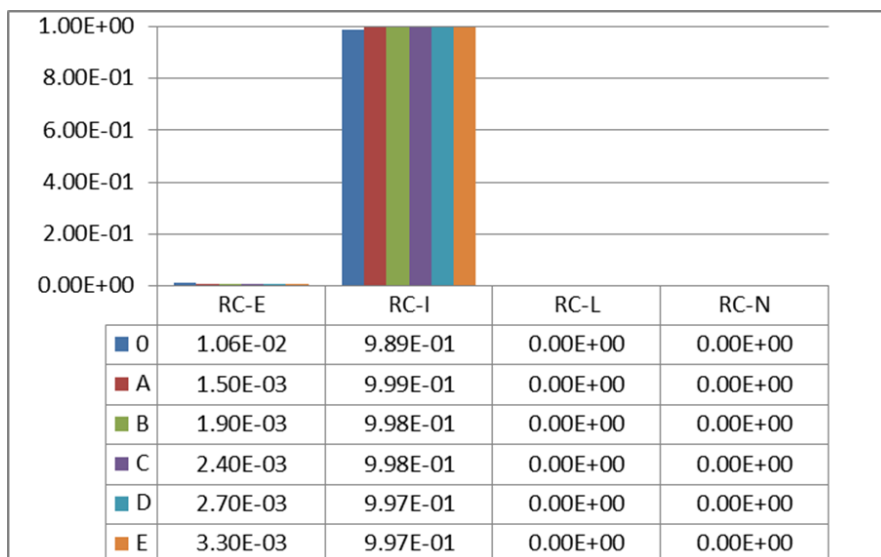


Figure 8: V&V example (Subcase 1.1.1) - Graphical presentation

Subcase 1.1.7: Combined HLA1 /HLA2 with HLA3I / HLA3R (SG Flooded / RCS Depressurization and RCS Injection / Recirculation)

- **Expectation:** With RCS depressurized and injection / recirculation available, there is a possibility to achieve in-vessel recovery (IVR). However, there is no containment heat removal. Containment will fail under RC-I only if IVR is unsuccessful and challenge to containment develops. Otherwise: long term concern.
 - Therefore, expectation for all subcases is: RCE-E low (SG flooded). Containment failure at RC-N (most likely) or RC-I.
 - Note: RC-L is not expected: if IVR fails then RC-I expected. If IVR successful then long term concern applies.
- **Results:** Considered OK (Table 2) In accordance with expectations.
 - Note: Split between RC-I and RC-N is in accordance with adequacy of available HLA3 function: RC-I(LPSI) < RC-I(DEC) < RC-I(FLEX) (Figure 9).

Table 2: V&V example (Subcase 1.1.7)

#	TDx = 0	RC-E	RC-I	RC-L	RC-N
0	Zero (All TD>>)	1.06E-02	9.89E-01	0.00E+00	0.00E+00
HLA1 and HLA3I / HLA3R					
A	(AFW) (LPS) (SMPLPSI)	1.27E-03	3.00E-02	0.00E+00	9.69E-01
B	(SGPORV) (DECSG) (LPSI) (SMPLPSI)	1.67E-03	3.00E-02	0.00E+00	9.68E-01
C	(DECSGPORV) (FLEXSG) (LPSI) (SMPLPSI)	3.08E-03	6.83E-02	0.00E+00	9.29E-01
D	(AFW) (DECSI) (SMPDECSI)	1.28E-03	1.27E-01	0.00E+00	8.72E-01
E	(DECSGPORV) (FLEXSG) (FLEXRCS) (SMPFLEX)	3.11E-03	3.47E-01	0.00E+00	6.50E-01

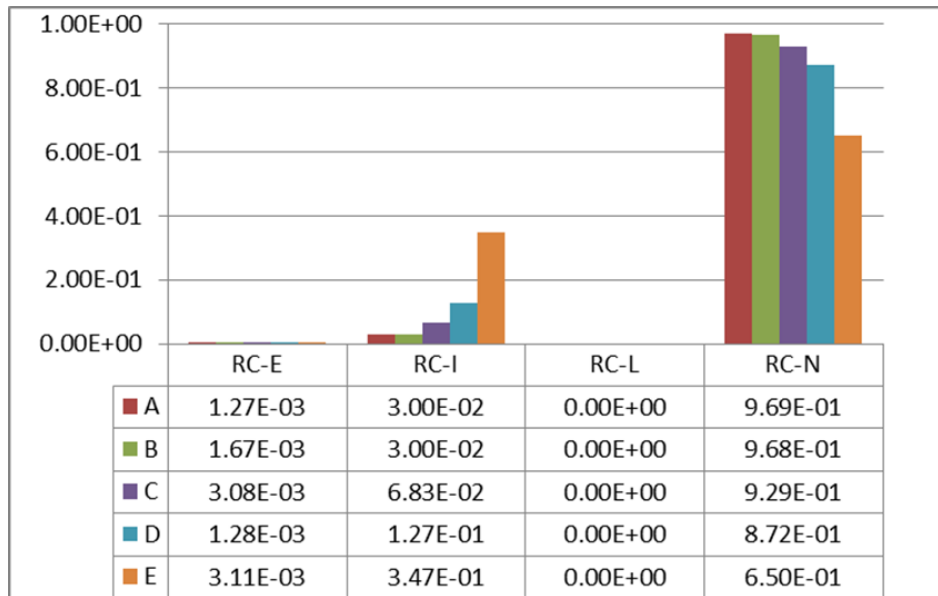


Figure 9: V&V example (Subcase 1.1.7) - Graphical presentation

4.4 Implementation of V&V

In order to run V&V tests, a special software module was added to Severa. After loading some time series (such as Station Blackout), the user can iteratively load test scripts, which are run by Severa, comparing the achieved radioactive release results with the ones obtained by the alternative evaluation tool and prescribed in scripts.

A test script is a JSON data file that contains a description of multiple hypothetical alternatives together with their expected radioactive releases. Each test/alternative is described by a number of data items that set up the hypothetical environment (time series data, the current time point) and the states of plant systems (in terms of TD and completed actions).

5 CONCLUSION

Severa is a proof-of-concept tool which was developed with an idea to investigate the possibilities of using a computer decision-support tool in severe accident management, primarily for the training of NPPs Technical Support Center (TSC) staff. The demonstration version of Severa is capable of evaluating potential successes of available severe accident management actions (SAGs), based on the assumed time windows for successful recovery actions and predetermined probability profiles of expected major radioactive release categories for different plant status/configurations. The appropriate timely executed operator actions should reduce the early containment failure or/and minimize other types of radiological releases. The TSC staff decisions based on additional information and training with Severa tool can lead to better understanding and management of severe accidents in nuclear power plants. Although the prototype version is somewhat simplified with regard to the real accident situations, the verification and validation exercises showed that it can provide reasonable predictions of probability profiles of major release categories for the scenarios considered.

With regard to the limitations in probabilistic risk quantifications, it is important to recognize that the objective of the tool itself is not to calculate the “realistic” or best estimate probabilities of releases associated with particular alternative being evaluated. Rather, the objective is to be able to learn which alternatives are relatively better than the others. In any case, this definitely represents an opportunity for future improvements, particularly the time dependency of the release category

probability matrix, which would enable using the tool also in the later phases of accident management.

ACKNOWLEDGMENTS

The work in the project NARSIS is supported by European Commission, Directorate general research & innovation, by grant agreement 755439. Marko Bohanec also acknowledges the financial support from the Slovenian Research Agency, research core funding No. P2-0103.



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