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Use of Supporting Software Tool for Decision-Making During Low-Probability Severe Accident Management at Nuclear Power Plants

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***Summary* — In the project NARSIS – New Approach to Reactor Safety ImprovementS – possible advances in safety assessment of nuclear power plants (NPPs) were considered, which also included possible improvements in the field of management of low probability accident scenarios. As a part of it, a supporting software tool for making decisions under severe accident management was developed. The mentioned tool, named Severa, is a prototype demonstration-level decision supporting system, aimed for the use by the technical support center (TSC) while managing a severe accident, or for the training purposes. Severa interprets, stores and monitors key physical measurements during accident sequence progression. It assesses the current state of physical barriers: core, reactor coolant system, reactor pressure vessel and containment. The tool gives predictions regarding accident progression in the case that no action is taken by the TSC. It provides a list of possible recovery strategies and courses of action. The applicability and feasibility of possible action courses in the given situation are addressed. For each action course, Severa assesses consequences in terms of probability of the 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 also described in this paper. Although largely simplified in its current state, Severa successfully demonstrated its potential for supporting accident management and pointed toward the next steps needed with regard to further advancements in this field.**

***Keywords* — Severe accident management at NPP, decision supporting tool, decision model, technical support center**

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I. Introduction

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) collaborated on the project NARSIS – New Approach to Reactor Safety ImprovementS [1]. The project was funded by the European Commission for the period of 4,5 years.

Based on recent theoretical progresses, the NARSIS project aimed at making significant scientific step forward towards addressing the update of some elements required for the safety assessment of NPPs. These improvements mainly concerned:

- Natural hazards characterization, in particular by considering concomitant 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".

The effectiveness of these improvements were tested and validated in the frame of the project through a set of laboratory experimentations and numerical simulations using generic nuclear power plant and real case applications.

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

- WP1: External hazards characterization,
- WP2: Fragility assessment of main NPPs critical elements,
- WP3: Integration and safety analysis,
- WP4: Applying and 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 main goal of work package (WP5) was the development of decision support tool for severe accident management and its demonstration., The referential nuclear power plant (NPP) was established [1]. The referential NPP was based on operating fleet in the European Union. The safety systems, structures and components (SSC) of referential nuclear power plant include design basis safety SSC, safety SSC to mitigate severe accidents and mobile SSC ("flexible" or FLEX equipment). The design basis SSC 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 prevents releases to the environment and radioactive doses to the public. Alternative energy sources in terms of diesel generators and batteries are included in the referential NPP.

Severe accident management guidelines (SAMGs) applicable to referential NPP were described [2]. In the 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. In the case of unsuccessful correction of the situation, they use abnormal operating procedures. 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 (TSC). In order to manage severe accidents, the SAMGs are used by managers in TSC. The SAMGs include operations such as:

a) Injection to steam generator, to remove decay heat from reactor coolant system (the so-called high-level action HLA1).

b) 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).

c) Injection to reactor coolant system, which assures coolant water to reactor core to remove decay heat (HLA3).

d) Injection of water into containment, to reduce containment pressure and possible radioactive releases.

The SAMGs imply that the TSC needs to take decisions. There could be large amount of information, available only partially, or with high uncertainty. The TSC managers are under stress due to an extensive damage in the NPP, potential releases of radionuclides and time pressure. The decision support tool Severa, developed in the project and described in this paper, targets accident management stage and aims at supporting the managers to make appropriate decisions with prioritization of actions in a well-justified and timely manner.

II. Input Data

The hazard-induced damage states and specific accident progression event tree for demonstration purposes were developed [3]. 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 (HP) and low pressure (LP) sequence. Figure 1 provides an indication of the time scale for the major phenomena. 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 [4].

The assessment of decisions needed to be taken by technical support center was carried out and the main decisions were identified and characterized. Then, the attributes against which all decisions are evaluated in decision support process were considered. Those included the status of main barriers, fuel cladding, reactor coolant boundary and containment. Since the status of boundaries (e.g. fuel temperatures) cannot be measured or observed directly, the related measurable parameters need to be used for diagnosis. Those are discussed in the next section.

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 [4].

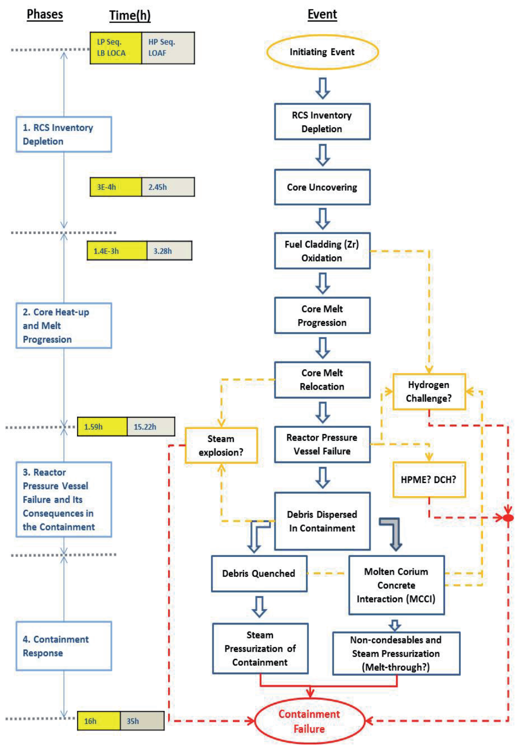


Fig. 1. Severe Accident Progression and Phenomenology

III. Decision Support Tool

The decision-support tool called Severa [5] is a demonstration-level Windows application, aimed at supporting the TSC team while managing a severe NPP accident.

Severa operates in 10–20-minute decision-support cycles that consist of the following steps:

1. Monitor the key NPP operating parameters and the availability and performance or plant systems.
2. Assess the damage state of the barriers. Identify barriers that are already challenged on may be challenged soon.
3. Predict possible future accident progressions and possible consequences in the case that no management actions are taken.
4. Identify possible alternatives (action courses); identified action courses include the actions which are required by the SAMGs (including the priorities given by the SAMGs) and consider the availability of plant systems/functions and time windows required for the implementation of each action.
5. For each identified alternative, assess its feasibility in the given situation.
6. Predict the possible consequences associated with each action course in terms of expected radioactive releases in the environment.
7. Compare the alternatives based on the expected releases and recommend the alternative to proceed with.
8. Implement the selected actions and observe plant's response.

Among these, the steps 1–7 are supported by Severa, mostly by carrying out the necessary simulations and calculations, and presenting the results in terms of (editable) tables, reports and charts to the users. Based on this information, the final step 8 is on behalf of the TSC team, who are also responsible for repeating the steps until the accident has been resolved.

Conceptually, the seven supported steps belong to two functional categories:

1. *Monitoring*: Observing and assessing the situation "as-is", without any human intervention. This category includes the steps 1–3.
2. *Management*: Supporting the decision-analysis and decision-support activities of the TSC, according to the steps 4–7. This encompasses the identification of possible management actions in a given situation and assessment of the possible consequences, including expected radioactive releases.

Another partial categorization of Severa's functionality can be made to:

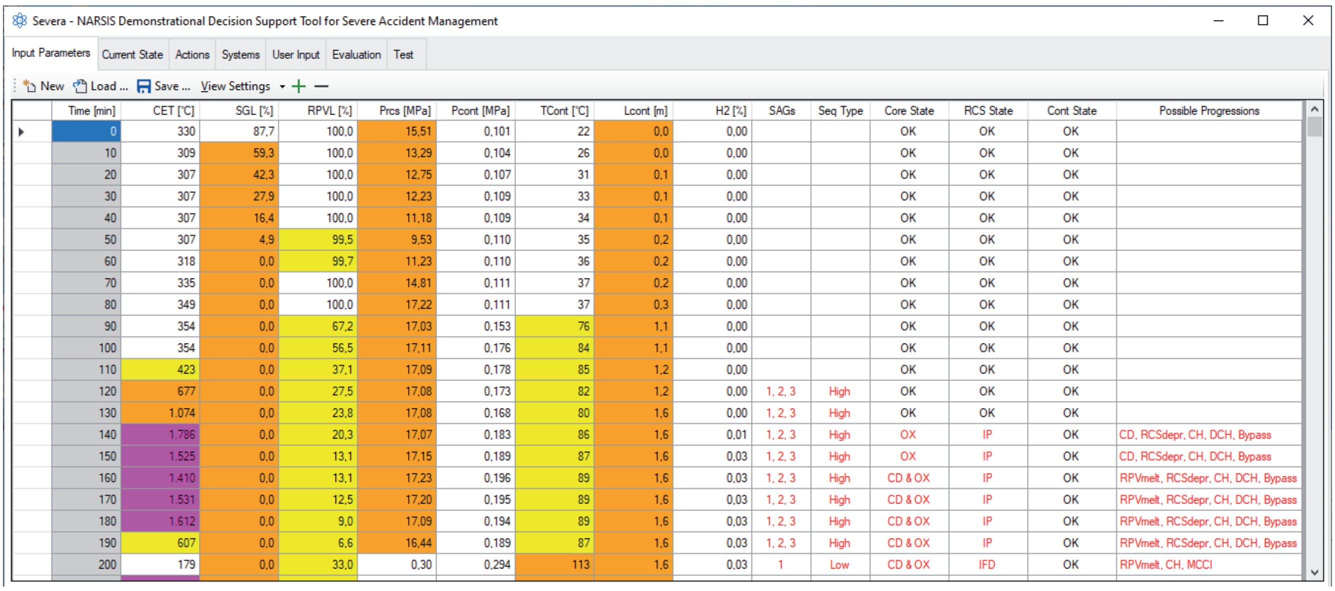
1. *Diagnostics*: Assessing the current state of the NPP and its barriers (steps 1 and 2).
2. *Prognostics*: Predicting future events: accident progression (step 3), feasibility of actions (5) and their consequences (6).

The operation of Severa is based on a time series of eight critical parameters that are periodically measured in the NPP [7]:

- "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 supports the monitoring steps 1–3 and 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 (Depressurization of Reactor Coolant System (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?


Severa - NARSIS Demonstrational Decision Support Tool for Severe Accident Management

Input Parameters | Current State | Actions | Systems | User Input | Evaluation | Test

New | Load ... | Save ... | View Settings

Time [min]	CET [°C]	SGL [%]	RPVL [%]	Prcs [MPa]	Pcont [MPa]	TCont [°C]	Lcont [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

Fig. 2. A Severa screenshot showing and interpreting the first 200 minutes of a Station Blackout scenario

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

- Columns "CET" to "Lcont": Color-coded interpretation of individual measurements. White, yellow, orange and red colors indicate the states of increasing severity, and magenta indicates an out-of-range or erroneous measurements.
- Column "SAG": Shows SAGs relevant for the situation (multiple SAGs are possible).
- Column "Seq Type": Sequence type, either low-pressure or high-pressure.
- Columns "Core State" – "Cont State": Assessed current state of the three barriers. The acronyms represent cladding oxidation (OX), core damage (CD), intact pressurized (IP) and intact/failed depressurized (IFD).
- Column Possible Progressions: Prediction of possible events if no actions are undertaken.

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 [9] developed according to the method DEX [8].

An important consequence of this approach is that each assessment, put forward by Severa, can be justified and explained in more detail when requested by the TSC. For example, Figure 3 shows a detailed description of the situation in the 120th minute of the Station Blackout scenario from Figure 2. The left-hand side shows the summary input parameter values at that time point, the set of entered SAGs, sequence type and a summary of barrier states. The right-hand side of Figure 3 displays detailed results of the evaluation carried out by the Barriers Progression DEX model. The report shows the hierarchical structure of attributes; for each individual attribute, it displays the qualitative value assigned to that attribute, which was determined from input sequence values and decision rules formulated in the model.

The second major decision-support contribution of Severa is related to possible management actions and their expected consequences (steps 4–7). In general decision-analysis terms, alternatives (or "decision alternatives") consist of multiple alternative courses of actions that may be undertaken in order to satisfy the decision makers' objectives. While managing a severe accident, the main objective is to mitigate the accident with minimum damage to the NPP and environment. 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 expected 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 ones. In Severa, the possible actions with the availability of equipment are used to define alternatives.

Each action has a success window, defined using the 95th and 5th percentile of success times. The expected action success probability is estimated by cumulative lognormal distribution depending on T05 and T95 (Figure 4). Here, the lognormal distribution was selected because of its convenience and because it is often used for phenomenological probability quantifications in the Level 2 PSAs. Any other probability distribution or probability model may be used in future versions, based on the human reliability or human factor analyses.

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 [7]:

- RC-E: Containment failure with a significant release of radioactivity is expected within several hours.

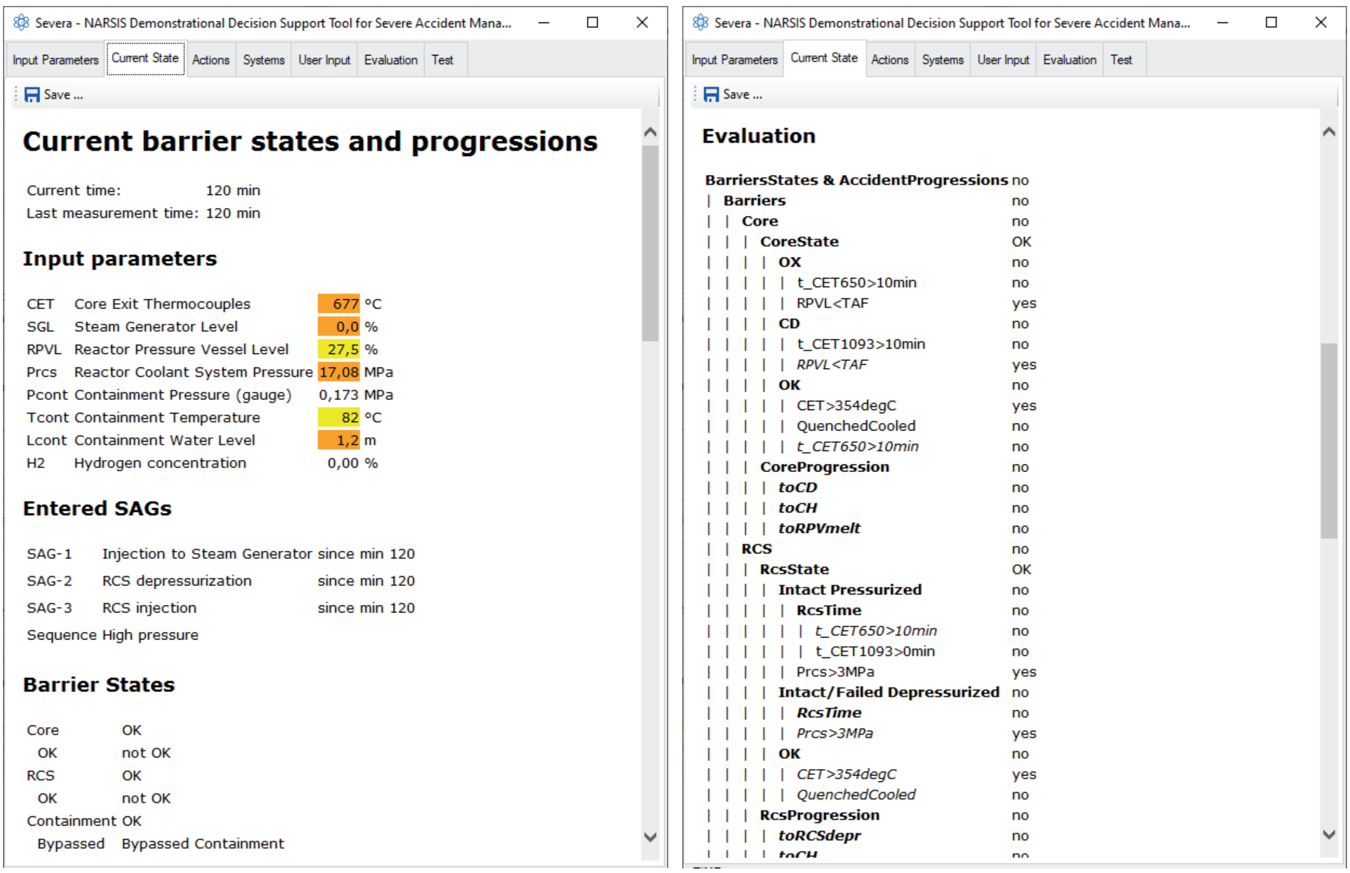


Fig. 3: The Current State report for minute 120 of Station Blackout

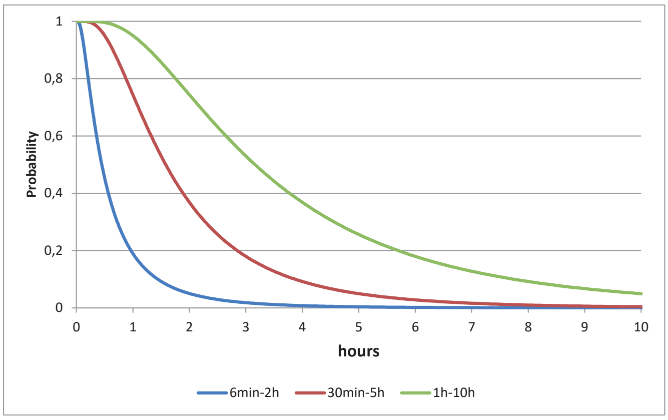


Fig. 4. Management actions' success windows in Severa

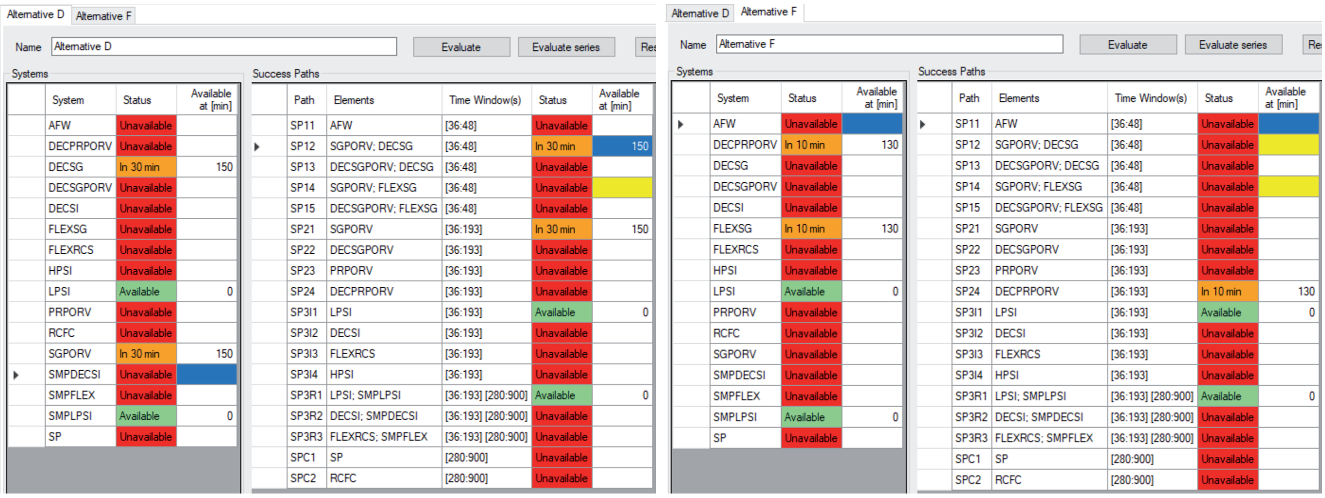
- RC-I: Containment failure with a significant release of radioactivity is expected within several days.
- RC-L: No significant release of radioactivity is expected within several days.
- RC-N: Long-term concern (in-vessel recovery and/or intact containment).

The main model for producing such assessments is based on an accident progression event tree (APET) [3], [4]. In Severa, the APET is implemented in terms of an equivalent probabilistic DEX model [7].

Let us illustrate the above concepts on an example of two hypothetical alternatives available to the TSC team in the 120th minute of station blackout (Figure 5). Acronyms in the figure denote plant systems that can generally be used to mitigate the situation. For instance, AFW denotes Auxiliary Feedwater Pump, SGPORV SG Power-Operated Relief Valve, etc. The abbreviation "DEC" refers to "Design Extension Condition". In most cases it is here used with a reference to the systems or equipment provided to cope with DEC conditions. Colors in Figure 5 denote the availability of those systems. The prevailing color is red, indicating that the corresponding systems are damaged beyond repair or otherwise unavailable. Only a few systems are available (green) or expected to be available (activated or repaired) in the future (orange).

Fig. 5. Definition of two alternatives in Severa

The two alternatives, available to the TSC team in minute 120, are:



Alternative D | Alternative F

Name: Alternative D — Evaluate | Evaluate series | Res

Systems

	System	Status	Available at [min]
	AFW	Unavailable	
	DECPRPORV	Unavailable	
	DECSG	In 30 min	150
	DECSGPORV	Unavailable	
	DECSI	Unavailable	
	FLEXSG	Unavailable	
	FLEXRCS	Unavailable	
	HPSI	Unavailable	
	LPSI	Available	0
	PRPORV	Unavailable	
	RCFC	Unavailable	
	SGPORV	In 30 min	150
▸	SMPDECSI	Unavailable	
	SMPFLEX	Unavailable	
	SMPLPSI	Available	0
	SP	Unavailable	

Success Paths

	Path	Elements	Time Window(s)	Status	Available at [min]
	SP11	AFW	[36:48]	Unavailable	
▸	SP12	SGPORV; DECSG	[36:48]	In 30 min	150
	SP13	DECSGPORV; DECSG	[36:48]	Unavailable	
	SP14	SGPORV; FLEXSG	[36:48]	Unavailable	
	SP15	DECSGPORV; FLEXSG	[36:48]	Unavailable	
	SP21	SGPORV	[36:193]	In 30 min	150
	SP22	DECSGPORV	[36:193]	Unavailable	
	SP23	PRPORV	[36:193]	Unavailable	
	SP24	DECPRPORV	[36:193]	Unavailable	
	SP3I1	LPSI	[36:193]	Available	0
	SP3I2	DECSI	[36:193]	Unavailable	
	SP3I3	FLEXRCS	[36:193]	Unavailable	
	SP3I4	HPSI	[36:193]	Unavailable	
	SP3R1	LPSI; SMPLPSI	[36:193] [280:900]	Available	0
	SP3R2	DECSI; SMPDECSI	[36:193] [280:900]	Unavailable	
	SP3R3	FLEXRCS; SMPFLEX	[36:193] [280:900]	Unavailable	
	SPC1	SP	[280:900]	Unavailable	
	SPC2	RCFC	[280:900]	Unavailable	

Alternative D | Alternative F

Name: Alternative F — Evaluate | Evaluate series | Re

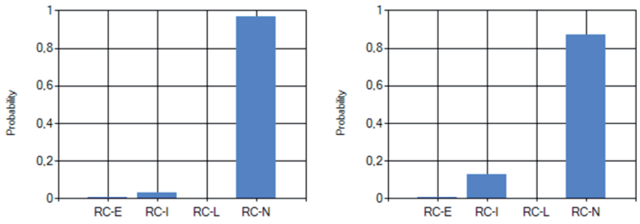
Systems

	System	Status	Available at [min]
▸	AFW	Unavailable	
	DECPRPORV	In 10 min	130
	DECSG	Unavailable	
	DECSGPORV	Unavailable	
	DECSI	Unavailable	
	FLEXSG	In 10 min	130
	FLEXRCS	Unavailable	
	HPSI	Unavailable	
	LPSI	Available	0
	PRPORV	Unavailable	
	RCFC	Unavailable	
	SGPORV	Unavailable	
	SMPDECSI	Unavailable	
	SMPFLEX	Unavailable	
	SMPLPSI	Available	0
	SP	Unavailable	

Success Paths

	Path	Elements	Time Window(s)	Status	Available at [min]
▸	SP11	AFW	[36:48]	Unavailable	
	SP12	SGPORV; DECSG	[36:48]	Unavailable	
	SP13	DECSGPORV; DECSG	[36:48]	Unavailable	
	SP14	SGPORV; FLEXSG	[36:48]	Unavailable	
	SP15	DECSGPORV; FLEXSG	[36:48]	Unavailable	
	SP21	SGPORV	[36:193]	Unavailable	
	SP22	DECSGPORV	[36:193]	Unavailable	
	SP23	PRPORV	[36:193]	Unavailable	
	SP24	DECPRPORV	[36:193]	In 10 min	130
	SP3I1	LPSI	[36:193]	Available	0
	SP3I2	DECSI	[36:193]	Unavailable	
	SP3I3	FLEXRCS	[36:193]	Unavailable	
	SP3I4	HPSI	[36:193]	Unavailable	
	SP3R1	LPSI; SMPLPSI	[36:193] [280:900]	Available	0
	SP3R2	DECSI; SMPDECSI	[36:193] [280:900]	Unavailable	
	SP3R3	FLEXRCS; SMPFLEX	[36:193] [280:900]	Unavailable	
	SPC1	SP	[280:900]	Unavailable	
	SPC2	RCFC	[280:900]	Unavailable	

- *Alternative D* (for "Design-based", Figure 4, left): The use of adequate design-based equipment (SGPORV and DECSG) will be possible only after extensive reparation work that will take about 30 minutes.
- *Alternative F:* (for "Flexible", Figure 4, right): Using less adequate flexible equipment (DECSGPORV, FLEXSG) that can be set up in 10 minutes.



	Alternative	RC-E	RC-I	RC-L	RC-N
▸	Alternative D	0,0017	0,0321	0	0,9662
	Alternative F	0,0058	0,1258	0	0,8684

Fig. 6. Probability distributions of radioactive releases for Alternatives D (left) and F (right)

Figure 6 shows the Severa's assessment of these alternatives in terms of probability distributions of RC-E, RC-I, RC-L and RC-N. 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. In this respect, Alternative D appears better than Alternative F, as its RC-N probability is considerably higher (0.9662 vs. 0.8684), while RC-E and RC-I are lower (0.0017 vs. 0,0058 and 0,0321 vs. 0.1258, respectively). Consequently, the TSC would be expected to choose Alternative D, initiate appropriate actions, and continue managing the accident carrying out next decision-making cycles.

It needs to be pointed out that Severa is a proof-of-concept tool which was developed in order to investigate the feasibility 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 its logic models, as well as a simplified consideration of adequacy of equipment included in the model and feedback from the implemented actions. At this point, Severa reflects three SAGs: SAG-1 (Inject into SG), SAG-2 (Depressurization of RCS) and/

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https://doi.org/10.37798/2022714423

or SAG-3 (Inject into RCS). However, even under the limitations, the development of Severa was very challenging and took quite a considerable decision-analysis, decision-modelling and programming efforts. Verification and validation exercises showed that it can provide reasonable predictions of probability profiles of major release categories for the scenarios considered.

IV. Verification & Validation Of Severa

A. Basic definitions

Two terms are essential for Severa verification and validation (V&V) process. They are "Time Delay" and "Alternative". These two terms have a specific meaning in the context of Severa [9].

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

- There are a certain number of TD terms for which values need to be entered by a user. Some of the terms relate to particular single systems, the others to combinations of systems. Those systems or their combinations comprise different possible "success paths" via which considered critical safety functions may be established / recovered.
- Generally, the designator "TDx" 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: $0 < TDx < \infty$:
    - 0 →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;
    - 0 →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 values of TD terms. For illustration, Figure 5 compares the TD terms for systems (which are then translated to the TD terms for the success paths) for two different alternatives.

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.

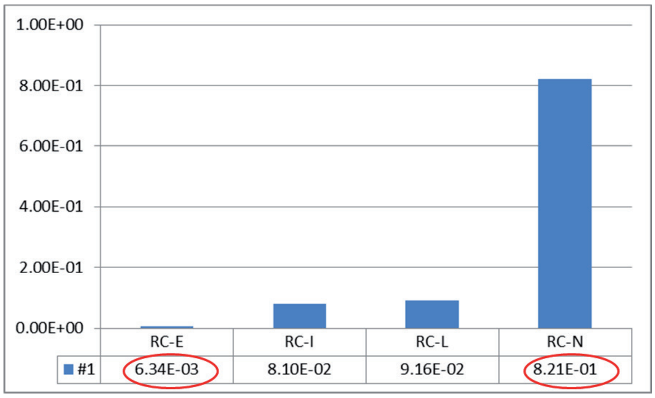

Fig. 6. Presentation of results – The range of probabilities of radioactive release categories

B. 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").

Usually, the second aspect is considerably more challenging than the first.

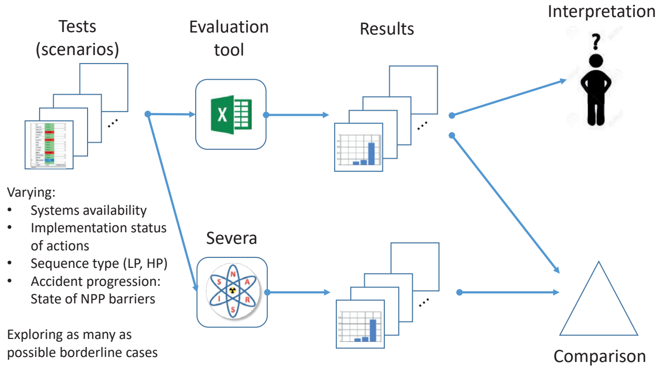

Fig. 7. Approach to verification and validation of Severa

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 [4] to evaluate phenomenological aspects of severe accidents, timing of important consequences without any recovery (operator) actions and success of performed recovery actions. According to [3] and [9], 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.

As already mentioned, two general types of severe accidents scenarios were studied in [4]: high pressure scenario and low pressure scenario. These two types generally differ with regard to the pressure behavior 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.

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

C. Example of a V&V

All cases presented below for illustration purposes represent checks involving a comparison of different alternatives. The considered situation is as follows. The time point at which tests are made is the time point at which the SAMGs are entered, i.e., the time point is set shortly after reaching CET = 650°C. Specifically, this occurs at time point = 126 min at the Station Blackout time series. No management action is under implementation. Tested is a set of alternatives with different TDx terms for specified functions. In all cases the following applies:

- Large TDx (TD goes to infinity) is simulated with TDx = 60000 min;
- TDxy = TDx + TDy when restoring combinations of systems;
- For each HLA / Function: No function is under implementation.

Case 1.0. Zero Alternative, A0: All TDx Large

- Zero alternative is defined as: no function available and no actions will be taken (no recovery). Therefore: all TDx terms are 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: As below. 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)

For this case, the initial / underlying conditions and assumptions are the same as under the Case 1.0 above. Various subcases which were then quantified reflect the assumption that particular function / combination of functions became available with TD = 0 (i.e., became available "now", at the time a decision is to be made). For example, in the Subcase 1.1.1 below it is assumed that the function "inject to SG" (HLA1) becomes available, while all other conditions are as under the Case 1.0 above. Presented below, for illustration, are two subcases: the mentioned Subcase 1.1.1, and the Subcase 1.1.7 under which it was assumed that a combination of critical functions becomes available with TD = 0. The purpose of all subcases under this Case 1.1 was to see whether the quantified results fulfill the expectation when compared against the initial results from the Case 1.0.

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 the Reactor Coolant Pump (RCP) seals and Pressurizer PORVs. Thus, reactor vessel failure (VF) and containment challenge cannot be avoided and they are expected at intermediate time frame. Therefore: RC-E probability decreases on account of RC-I. RC-L = RC-N = 0.

  o→The best option, according to the assumptions, is AFW. (This is because it is a design-basis safety system, with most strict design, installation and maintenance requirements.) Thus, this option is expected to give the smallest RC-E probability. For other options RC-E probability increases.
- Results: As shown in Table I. Considered OK. (Note that the row #0 shows the results from the Case 1.0 above, for comparison.) 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.

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Table I.

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

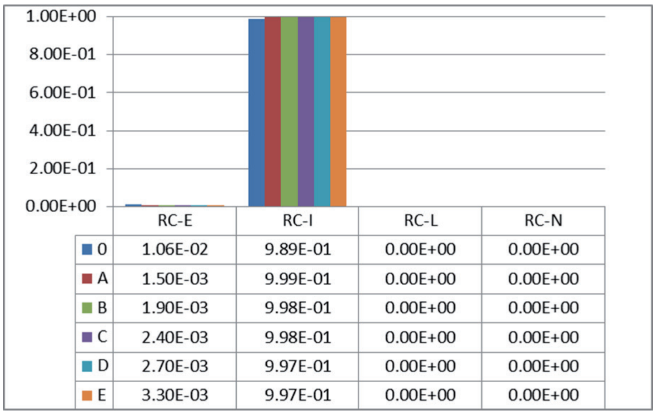

Fig. 8. V&V example (Subcase 1.1.1) - Graphical presentation

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.
  - o→Therefore, expectation for all options 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: As shown in Table II. Considered OK. In accordance with expectations.
  - o→Note, also: Split between RC-I and RC-N is in accordance with adequacy of available HLA3 function: Probability of RC-I(LPSI) is smaller than probability of RC-I(DEC), which in turn is smaller than probability of RC-I(FLEX) (Figure 9).

D. 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 (JavaScript Object Notation) 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).

Table II.

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

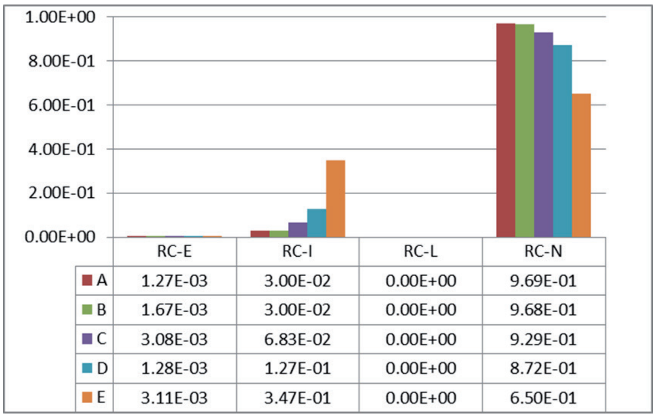

Fig. 9. V&V example (Subcase 1.1.7) - Graphical presentation

V. Conclusion

Severa is a proof-of-concept tool which was developed with an idea to investigate the feasibility 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 guideline (SAG) action courses, 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 largely simplified with regard to the real situations, the extensive 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.

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References

[1] L. Štrubelj, E. Foerster, G. Rasitello, J. Daniell, B. Bazargan-Sabet, P. Gehl, P. J. Vardon, V. K. Duvvuru Mohan, "The Goal of the New Approach to Reactor Safety Improvements (NARSIS) Project", Proceedings of the 12th International Conference of the Croatian Nuclear Society, Zadar, Croatia, 3-6 June 2018

[2] K. Debelak, L. Štrubelj, I. Bašić, "Report on characterized EOP/EDMG/SAMG – Deliverable 5.2", NARSIS project, 2018

[3] I. Vrbanić, I. Bašić, L. Štrubelj, K. Debelak, M. Bohanec, "Definition of Hazard-Induced Damage States and Development of State-Specific APETs for Demonstration Purposes – Deliverable 5.3", NARSIS project, 2019

[4] P. Darnowski, P. Mazgaj, I. Bašić, I. Vrbanić, M. Skrzypek, J. Malesa, A. Silde, J. H., L. Štrubelj, Severe Accident Simulations Dedicated to the SAMG Decision-Making Tool Demonstration, NENE-2020, 2020

[5] M. Bohanec, A. Kikaj, I. Vrbanić, I. Bašić, K. Debelak, L. Štrubelj, "Supporting Severe Accident Management in Nuclear Power Plants", ICDSST 2020, Zaragoza, Spain, 27.5-29.5.2020

[6] M. Bohanec, I. Vrbanić, I. Bašić, K. Debelak, L. Štrubelj : A decision-support approach to severe accident management in nuclear power plants. Journal of Decision Systems, 13 p., 2020, doi: 10.1080/12460125.2020.1854426.

[7] N. Trdin, M. Bohanec: Extending the multi-criteria decision making method DEX with numeric attributes, value distributions and relational models. Central European Journal of Operations Research, 26(1), 1–41, 2018. doi:10.1007/s10100-017-0468-9

[8] Bohanec, M.: DEX (Decision EXpert): A qualitative hierarchical multi-criteria method. Multiple Criteria Decision Making (ed. Kulkarni, A.J.), Studies in Systems, Decision and Control 407, Singapore: Springer, doi: 10.1007/978-981-16-7414-3_3, 39-78, 2022

[9] M. Bohanec, I. Vrbanić, I. Bašić, D5.4a - Supporting SAMG DM tool for demonstration purposes, NARSIS Deliverable, 2021