NARSIS

New Approach to Reactor Safety ImprovementS

Newsletter # 6

Edito

Evelyne Foerster NARSIS project coordinator

Dear Reader,

As the NARSIS project has finally come to an end, I am pleased to present this last issue, which concludes our adventure started in 2017 and which aimed to improve the safety and reliability of generation II and III reactors.

NARSIS was above all a network of trust, mutual support and knowledge transfer between the major players of the community. Together, seventeen partners encompassing leading universities, research institutes, technical support organizations (TSO), nuclear power producers and suppliers, reactor designers and operators from ten countries have collaborated in the frame of this H2020 EURATOM project. This European cooperation effort for a safer use of the nuclear energy has demonstrated it can lead to a stronger and better preparation for the future. During this project, the consortium gave its best to develop and update important topics related to the Probabilistic Safety Assessment (PSA), focusing on external natural events.

I am most proud of our numerous accomplishments over all these years. Among others, we developed and implemented the NARSIS Multi-Hazard Explorer (MHE), an open-source open-access tool to be used as part of the steps related to Initiating Events and Screening (deterministic or probabilistic) analyses in extended PSA. The MHE is suitable not only for multi-hazards scenarios but also for independent single hazards assessment. We also developed some refined fragility derivation methods, in order to increase the estimation accuracy of the Systems, Structures and Components (SSC) failure rates.

Various numerical models and approaches were investigated in order to integrate cumulative effects such as ageing mechanisms, successive loadings (e.g. thermal fatigue combined to earthquake events), or soil-structure interactions. We integrated the human factors in the reliability analysis, as a potential source of epistemic uncertainty. A novel Bayesian Networks (BN)-based methodology for human error probability was developed and connected to technical BNs used for risk integration. A new approach to the analysis of Common Cause Failures (CCFs) was developed improving existing methods in both impact calculation and visualisation. The applicability of two strategies to reduce the computational costs for plant safety assessment, was investigated, namely the metamodelling techniques and a novel solving strategy, combining the LArge Time INcrement and Proper Generalized Decomposition methods for model-order reduction. Finally, the demonstrative decision supporting tool SEVERA was developed and implemented for severe accident management, encompassing both diagnostic and prognostic parts, useful for the technical support center staffs, so to speed up the decision process and make it more informed. We prepared and organized lectures for students and young professionals, in order to disseminate our scientific findings widely. In As you will discover in the following pages, we presented the most significant results of the NARSIS project during our final workshop in February 2022, fostering dialogue between scientific communities conducting similar research in other fields.

While NARSIS comes to its end, we are of course well aware that much remains to be accomplished. We can now rely on a strong multidisciplinary network to ensure the sustainability of our initiative. To conclude, I would like to thank all the NARSIS partners who contributed to this success story. I look forward to having new exciting collaboration opportunities with you.

I wish you good inspiration!

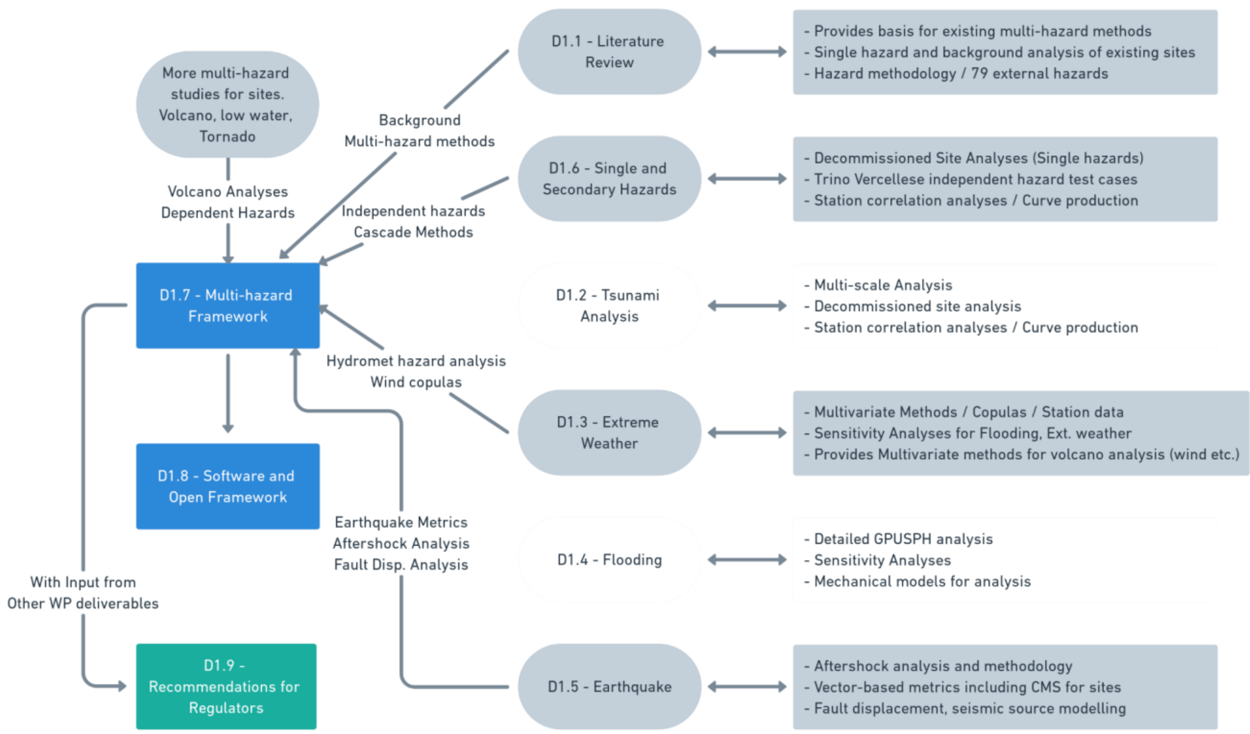
FINAL WORKSHOP

The NARSIS final workshop, held on February 2022, aimed at presenting the main results and achievements from the research activities carried out within the NARSIS project. It was also an opportunity to bring together different safety-related and risk management communities. Due to the Covid-19 pandemic, the workshop was held online and offered virtual presentations from the NARSIS consortium as well as from different invited speakers. Hereafter is a summary of the main topics presented. All the presentations can be found at: [NARSIS Final Workshop (narsis.brgm.fr)](narsis.brgm.fr).

Topic1: The Multi-Hazard (MH) framework

Main objectives: Propose new approaches for characterization of potential physical threats a nuclear installation can be exposed to, due to different external natural hazards and scenarios. Develop an integrated multi-hazard framework for nuclear safety assessment, accounting for single and combination events at different time scales, including the potential impact on supply and infrastructure in which the NPP is embedded and on which its functionality depends on.

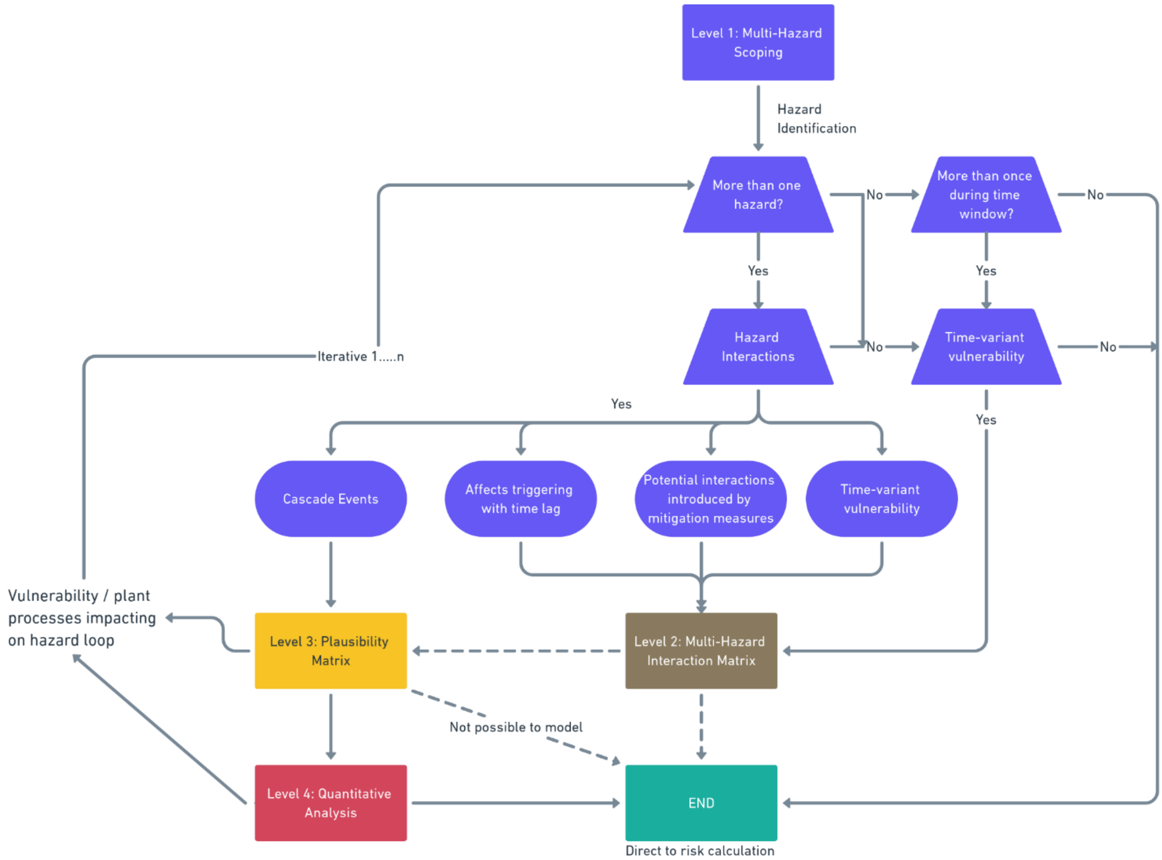
The Multi-Hazard (MH) framework assesses and quantifies primary and secondary hazards including cascading effects as well as uncertainty, in order to allow studying the consequences of combinations of potential well-characterised physical threats due to different external hazards and scenarios. These new approaches focus on earthquakes, flooding, tsunamis and extreme weather, as they have been identified as priorities by the PSA End-Users community in the ASAMPSA-E project.



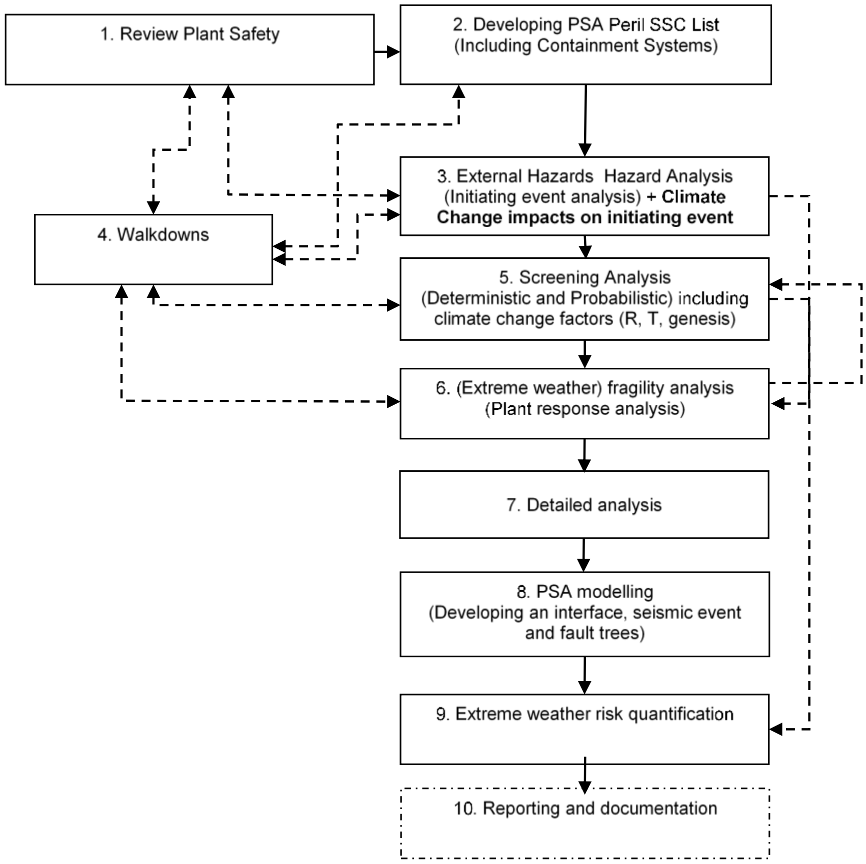
The final methodology proposed in NARSIS is based on the 3-level framework proposed by the FP7 MATRIX project, with complements and adaptations for the NPP specific nature. Hence, the framework includes five successive levels:

- Level 0 : Single hazard assessment through standard practice or improved methods
- Level 1: Multi-hazard assessment scoping through potential site specific hazards
- Level 2: Multi-hazard interaction matrix and scoring
- Level 3: Modellability matrix
- Level 4: Quantitative analysis of multiple hazard probabilities

Various pathways for analysis of multi-hazard scenarios can be followed as shown below.



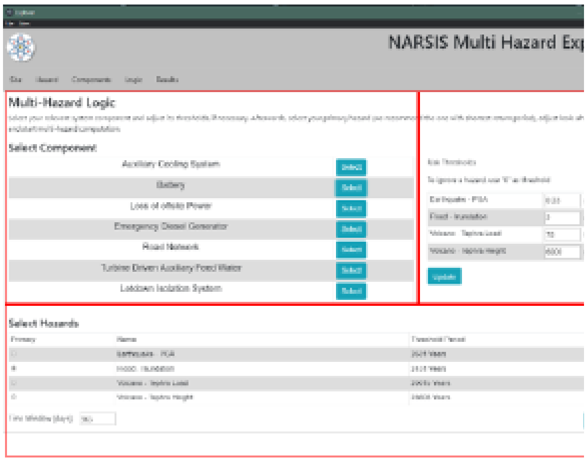
The flow chart for extended PSA with the proposed location of the multi-hazard framework component is presented hereafter.



A collection of hazards datasets was examined and analysed for earthquake (and secondary effects), tsunami, flood, hail, lightning, tornado, rainfall, temperature, volcano and wind; including screening.

The integration of hazard analyses and sites examined, has been performed in an open-source and open-access software framework, namely the NARSIS Multi-Hazard Explorer (MHE), so to be able to model and simulate single as well as multiple hazards (e.g. high winds or earthquake & high precipitations leading to structural damage and equipment flooding; or earthquake with fire-following or flooding-following due to damaged spent fuel pool or pipes, etc.).

The NARSIS MHE is an entry-level tool to quickly review and assess multi-hazard scenarios. It allows to view and manipulate hazard curves of which various samples are included to assess potential variations on the given sample data. Hazard curves can be combined for multi-hazard assessment with consideration of secondary effects (e.g. landslides or liquefaction). Multi-hazard is hereby defined as the linear combination of 2 independent hazards. Combining more than 2 hazards is not part of the software but can be integrated by rerunning analyses. The tool uses given hazard return period curves of independent hazards and computes occurrence probabilities for both hazards happening in the same time window. Similarly, dependent hazards can also be applied and brought in as stochastic event set probabilities on one component of the linear combination. The generic nature of the software allows to be used outside the nuclear field and also provides a standalone which can be adapted by plant operators or modellers only for internal use on a specific site. The NARSIS MHE tool is available at: https://github.com/a-schaefer/NARSIS-MHE



The NARSIS partners also presented some innovative solutions developed as part of the multi-hazard framework, and dedicated to the improvement of existing Probabilistic Hazard Assessment (PHA) methodologies, respectively for:

- ***Tsunami***
  - Fast high-resolution Seismic- Probabilistic Tsunami Hazard Assessment approach to estimate hazard at a coastal level, through accurate numerical tsunami propagation and inundation modelling on high resolution bathymetry/topography grids along the coastlines;
  - Stochastic modelling of the fault slip spatial heterogeneity for near-field tsunami simulations, in order to assess the impact on the near-shore wave propagation and flooding;
  - Probabilistic tsunami modeling for decommissioned sites.
- ***Extreme weather and flooding***
  - Uncertainty Quantification and Global Sensitivity Analysis for dependent model inputs and levee breaches consideration, in order to better understand the numerous uncertainties in hydraulic models; application to the 2D hydraulic model of the Loire River in France.
  - Multivariate analyses for extreme weather analysis (heatwaves, extreme rainfalls), with application to decommissioned plant sites.
- ***Extreme earthquake***
  - Use of a vector-based methodology to select the appropriate hazard parameters for a given site, and to assess multiple metrics to describe a same hazard event;
  - Aftershocks – Ground Motion (GM) severity and probability assessment, given a mainshock GM at a site.
  - Modelling of earthquake – flood interaction, considering independent and dependent (e.g., levee break due to the earthquake) events.

Topic 2: Fragility assessment

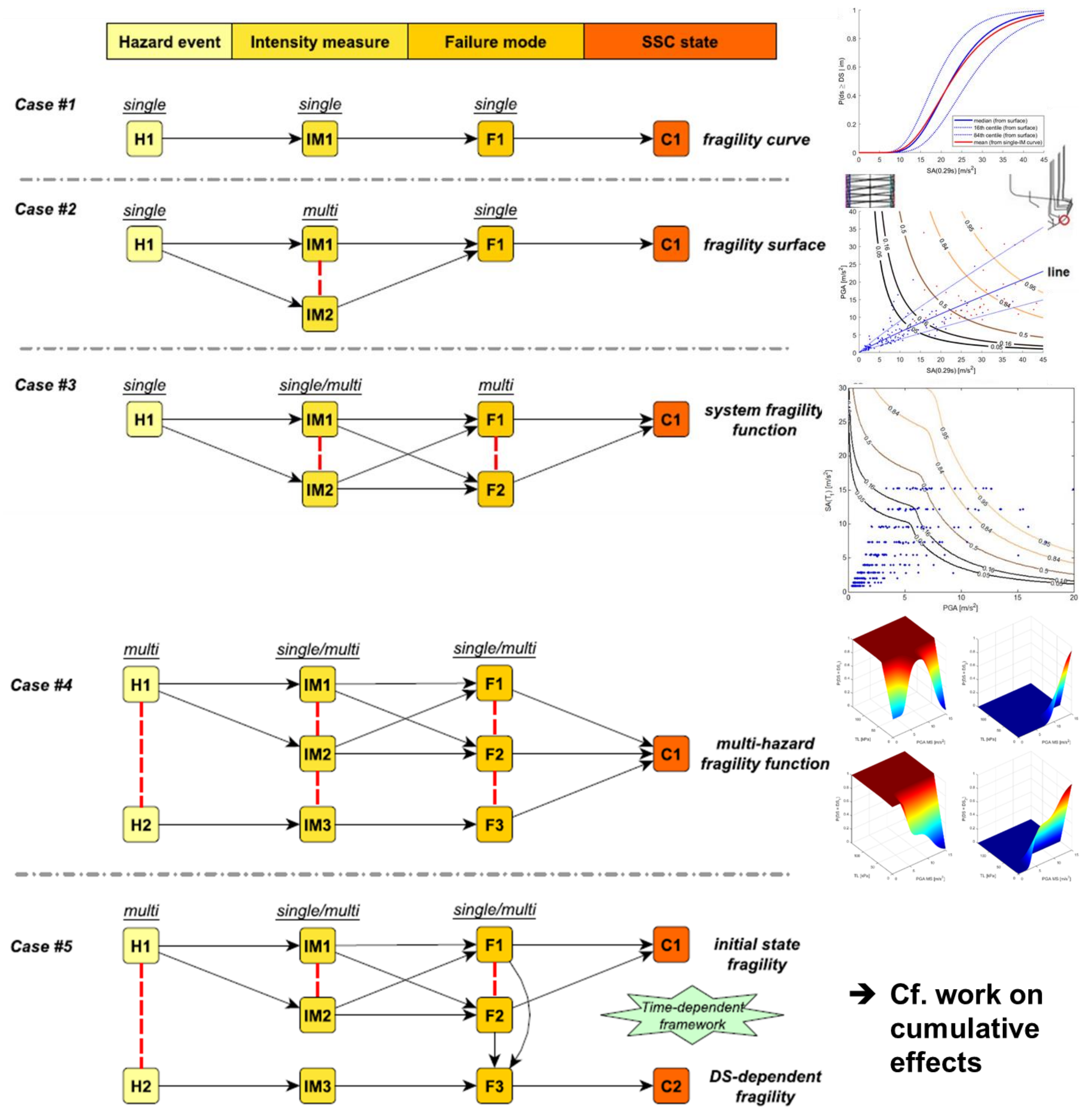
Main objective : Develop refined fragility derivation methods in order to increase the accuracy of the estimation of Systems, Structures and Components (SSC) failure rates, thanks to current advances in quantitative hazard modelling and computational capacities.

Standard practice in fragility analysis relies on a single (scalar) intensity measure (IM). The benefits of using alternative IM as well as fragility functions based on more than one IM (i.e. vector-valued IM) are discussed. General principles and theoretical framework are first presented including fragility function and major types of uncertainty. Fragility curves, which express the probability of a SSC to reach or exceed a predefined damage state as a function of an IM representing the hazard loading, are common tools in the nuclear field. A short analysis of the main features and limits of the current statistical methods for the derivation of fragility functions is presented.

Method	Added value	Main limits	Example
Separation-of-Variables	- Reuse existing design calculations (high level of quality assurance!) -> cost-effective, good enough for vast majority of components;	- Assumes linearity of demand w.rt. IM (partial correction with inelastic energy absorption factor);	- EPRI TR-103959 (1994)
Regression "on a cloud"	- Simple and intuitive approach; - Stable fragility estimates may be obtained with a few data points;	- Constrained by the functional form of the IM-EDP relationship; - Constant standard-deviation over the IM range;	- Seismic fragility of an RC structure (Seyedi et al., 2010)
MLE / GLM regression	- Applicable to empirical fragility assessment (if only damage data are available); - Ability to treat complete damage/collapse cases (where EDP values are usually inaccurate); - Compatible with multivariate regression;	- Loss of information (i.e., the true values of the EDP are not used); - More data points are required to achieve stable fragility estimates;	- Seismic fragility of a masonry structure (Gehl et al., 2013); - Empirical tsunami fragility of buildings (De Risi et al., 2017);
Bayesian updating	- Compatible with expert-judgment approaches or Experimental results;	- Influence of the prior distribution on the final fragility estimates;	- Seismic fragility of switchgear cabinets (Wang et al., 2018)

A multi-hazard fragility framework is presented in order to treat various cases of multi- and single-hazard interactions. As a result, it is found that a total of five cases may be able to describe most of the configurations that are encountered, when dealing with external hazard events:

1 Standard single-IM case, with a simple IM- Engineering Demand Parameter (EDP) relationship.
1. Vector-IM fragility function, usually with a correlation between the IMs.
2. System fragility function, resulting from the assembly of single component damage events (i.e., combination of failure modes). The correlation between the occurrences of the failure modes, given the IMs, should be taken into account.
3. Multi-hazard fragility function, where a multi-variate distribution function represents the damage probability due to the interaction of co-occurring loadings.
4. Damage-state-dependent fragility functions where a first hazard loading may degrade the resistance of the SSC or alter the conditions for when a subsequent hazard loading is applied (i.e., sequence of events). The hazards may be correlated (i.e., same source event, or one hazard event triggering another) or independent (i.e., occurrence within the same time window).



It is believed that these five cases are able to cover most of the multi-hazard cases, although cases #4 and #5 may be further refined.

The benefits of using vector-valued IMs for fragility assessment of SSC is investigated through three application cases, namely:

- Vector-IM fragility functions for seismic hazard, considering coupled model of a supporting structure and a steam line;
- System fragility functions, illustrated by a reactor building including containment building, steam generator and piping system;
- Multi-hazard fragility functions illustrated by protection dikes submitted to the combination of earthquake (mainshock-aftershock), hydraulic and tephra loadings.

Criteria for selecting seismic IM encamps:

- Efficiency: ability of an IM to induce a low dispersion in the distribution of the structural response
- Sufficiency: ability of an IM to “carry” the characteristics of the earthquake that generated the ground motion
- Practicality: strength of the link between IM and EDP
- Proficiency: combination of practicality and efficiency
- Computability or Hazard compatibility: ability to compute the selected IM accurately with current Ground Motion Prediction Equations GMPEs

Analyses show that:

- Carefully selected vector-IMs make excellent candidates in terms of IM sufficiency and efficiency, when compared to scalar IMs.
- Vector-valued fragility functions tend to generate less dispersion (i.e., aleatory uncertainty due to record-to-record variability) than scalar-IM fragility curves: this difference may be interpreted as a partial transfer from the record-to-record variability to an epistemic uncertainty component that is related to the description of the seismic loading given the hazard at the studied site.
- The conditional spectrum method for the selection of input ground-motion records appears to be compatible with the derivation of vector-based fragility functions.

Furthermore, provided that the required hazard-specific physical models are available, the statistical tools are able to cover most of the multi-hazard cases:

- Multivariate Generalized Linear Model (GLM) regression or Maximum Likelihood Estimation (MLE) are to be used for the estimation of fragility parameters given a set of conditioning variables;
- System reliability theory is able to combine hazard-specific failure modes in order to model the SSC functionality states of a given SSC.

As a complement to the NARSIS vector-valued fragility assessment presentation, the partners presented 4 study cases entitled:

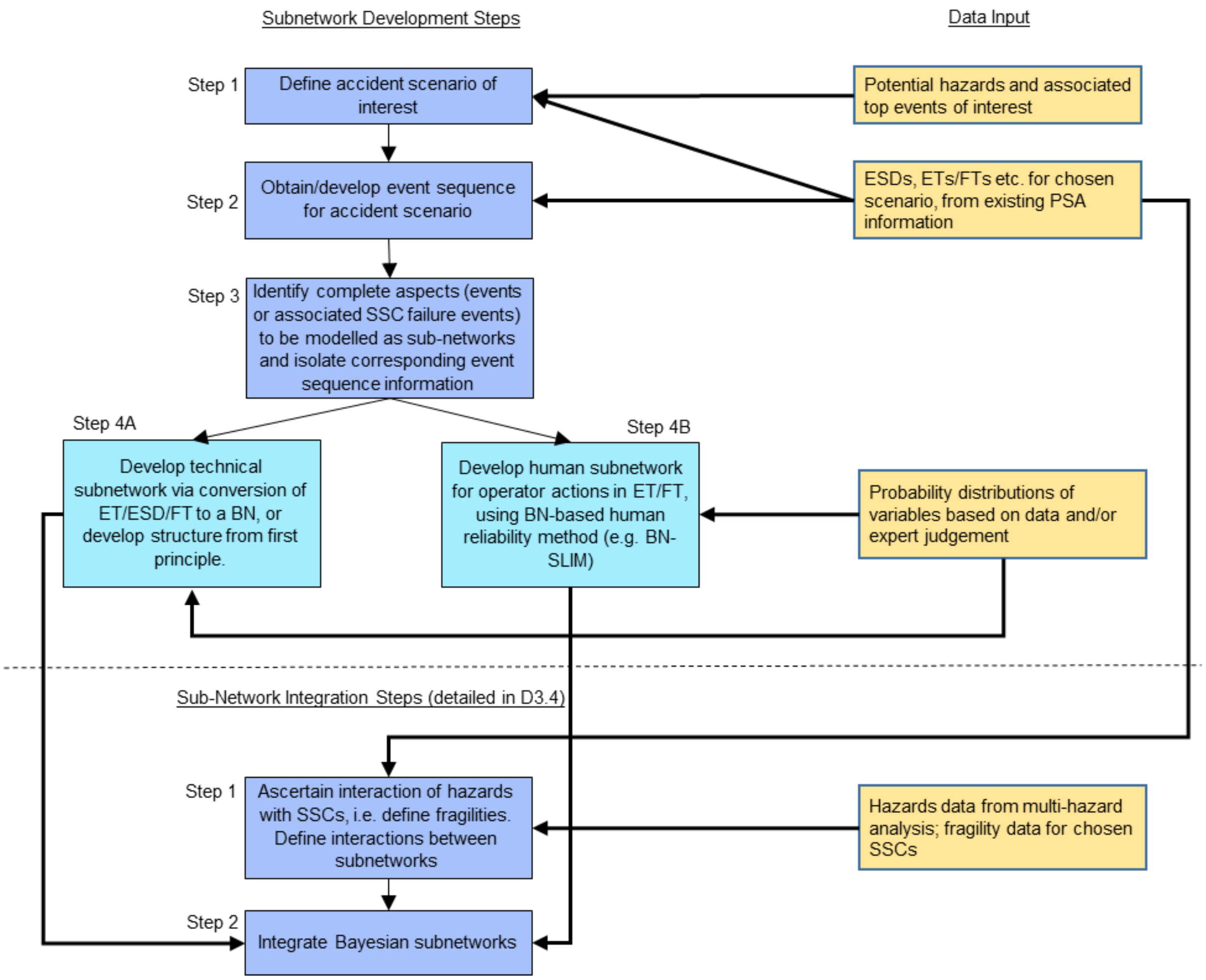
- *Fragility assessment of main NPPs critical elements, accounting for soil-structure interactions;*
- *Fragility assessment for nuclear SSC: Methodology to account for ageing mechanisms in the fragility assessment;*
- *Fragility curves including seismic input level and pre-existing damage pattern due to thermos-mechanical fatigue phenomenon;*
- *Incorporating the human factor in probabilistic safety analyses within the nuclear domain.*

Topic3: The Multi-risk integration framework for safety analysis

Main objective: Improve the integration of external hazards and their consequences with existing state-of-the-art risk assessment methodologies in the industry. The approach taken was to investigate, further develop and evaluate different methods for safety assessment of NPP.

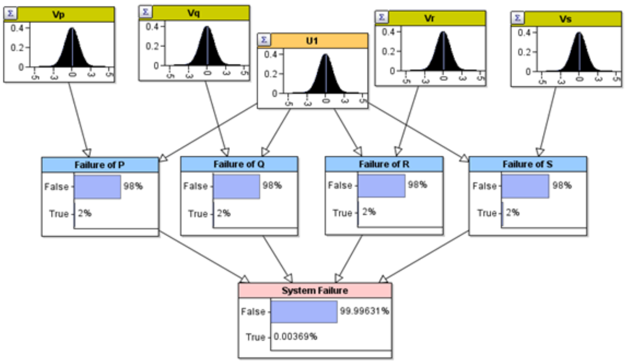
Investigations performed in the frame of NARSIS target the use of the Bayesian Networks (BNs) as an alternative to traditional methods such as fault trees, as well as the development of an Extended Best Estimate Plus Uncertainties (E-BEPU) methodology. Both activities were embedded in the efforts for constraining uncertainties.

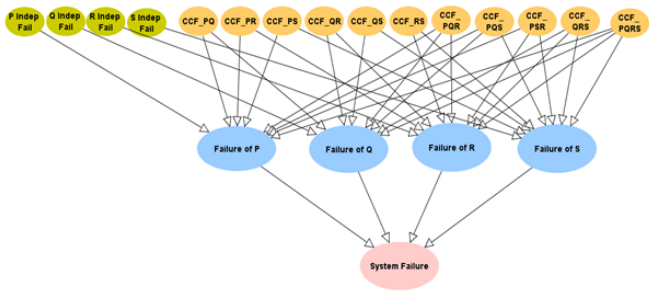
BNs fall within the strategy developed to integrate hazards, technical and human aspects. A generic methodology which can be used to develop BNs for NPP safety assessment in case of external hazard events, is presented hereafter.



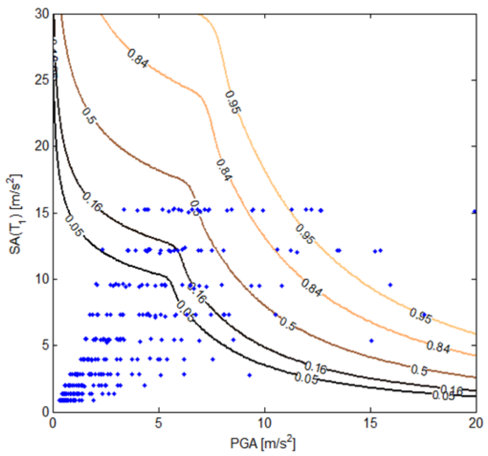
In NARSIS, BNs have been developed and used to explore different topics such as:

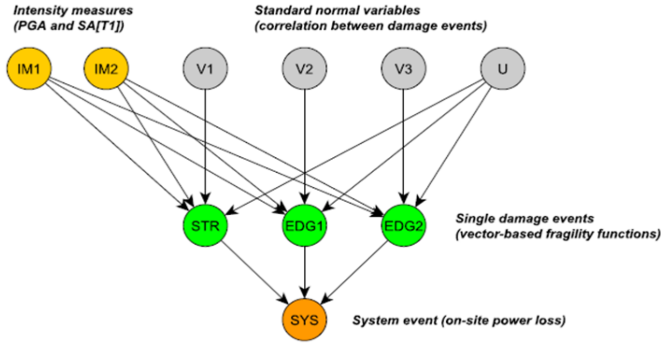
- Common Cause of Failure (CCF) modelling: the proposed new approach using BNs, based on correlation between component failures, has advantages over conventional parametric models, especially in asymmetric systems. The method can also simplify visualisation of BNs for complex systems with many redundancies.



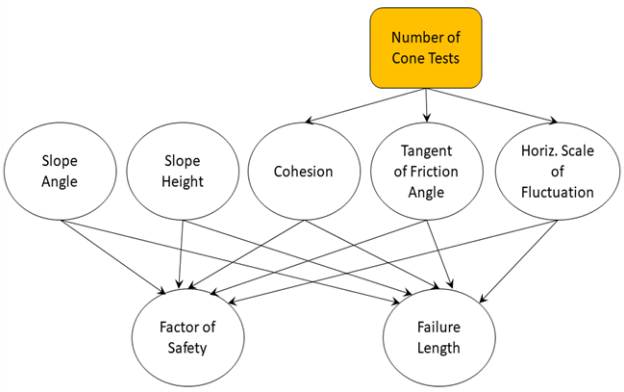


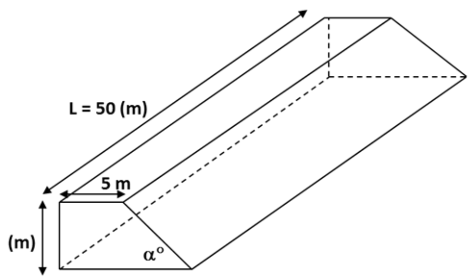
- Vector-based fragility: BNs can combine multiple hazards and vector-based fragility of SSCs, allowing for the inclusion of more than one intensity measure for each hazard.



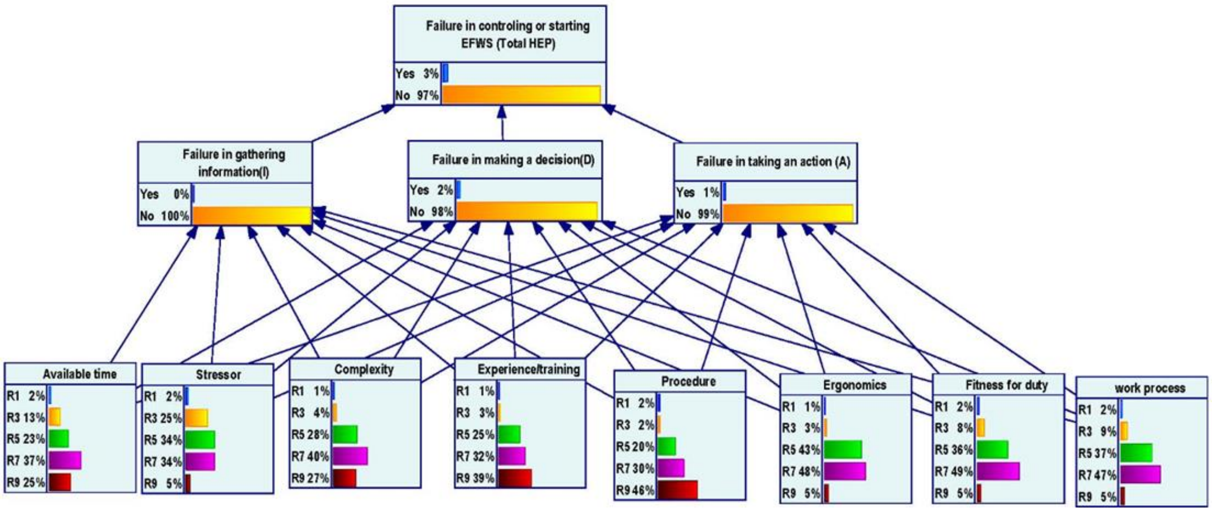


- Surrogate modelling: BNs can be used as surrogate model for advanced numerical methods in reliability assessment (e.g. flood control dikes). Such surrogate BNs can ease computational demands and provide a direct link to larger BN for overall system risk assessment.



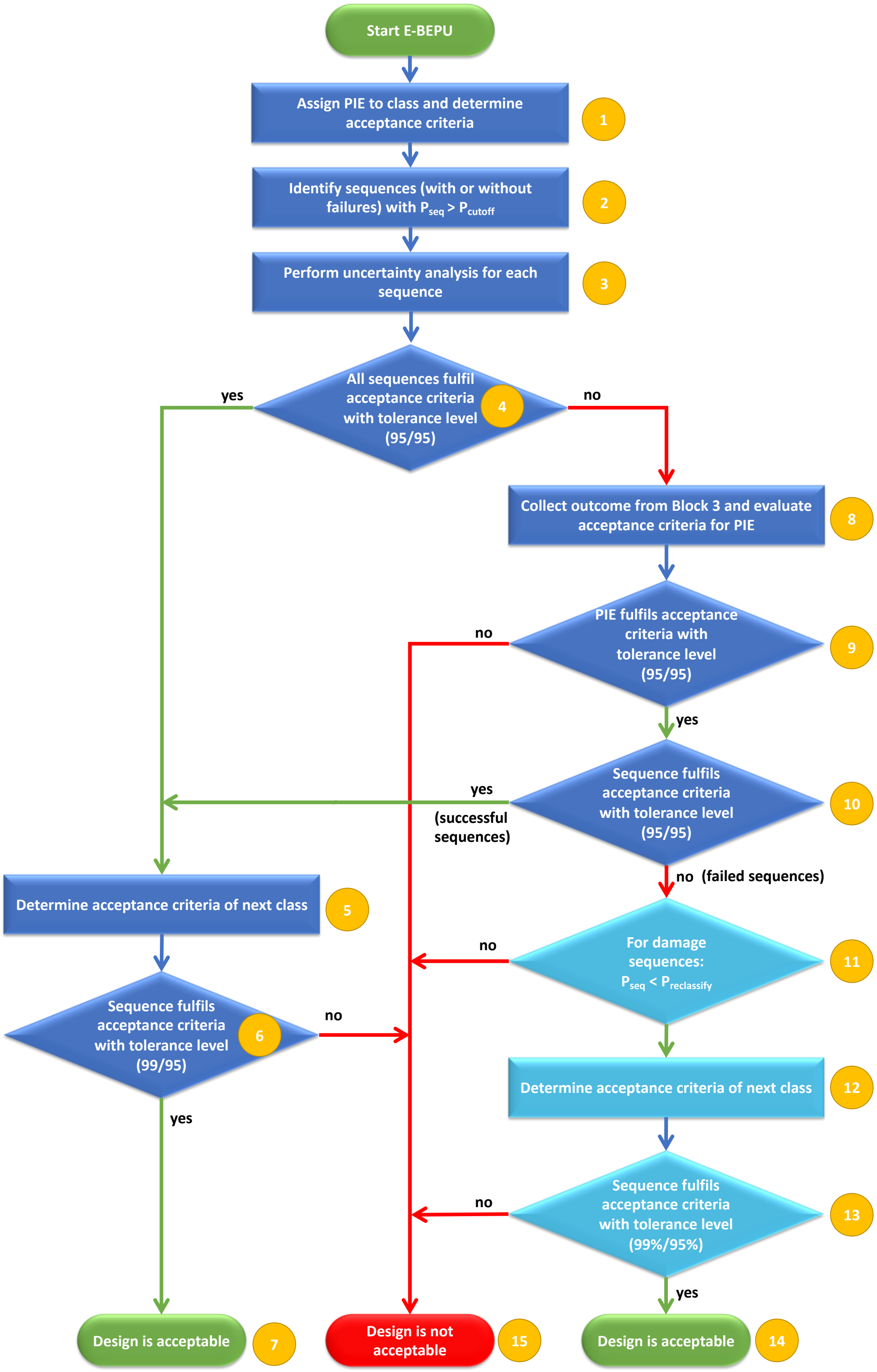


- <u>Human aspects:</u> in the frame of NARSIS, the new BN-SLIM, developed for the estimation of Human Error Probability (HEP), was coupled with structured expert judgement elicitation to power the probabilities, highlighting its applicability in data-scarce risk problems.

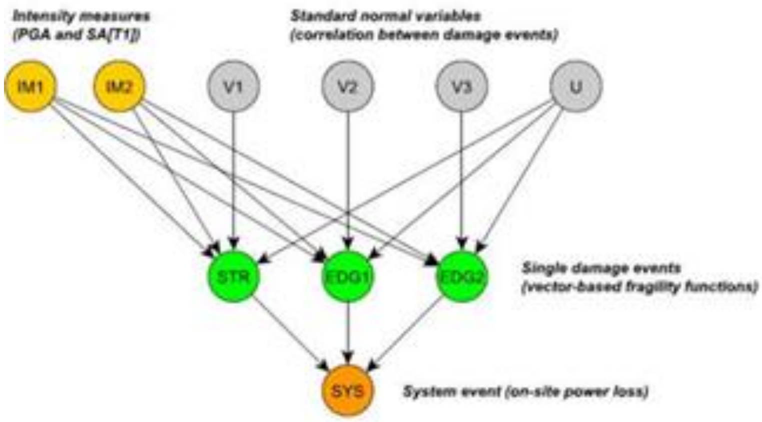


E-BEPU is a safety analysis methodology applicable to the analysis of Postulated Initiating Events (PIE) in complex facilities. Best-Estimate Plus Uncertainty (BEPU) methodologies are now widely accepted for the analysis of Design Basis Accidents (DBA). In this type of methodologies, simulation of the plant dynamics is based on best-estimate models. Uncertainties are considered in initial and boundary conditions, in properties of the system and in physical models. E-BEPU is an extension of the BEPU methodology with two important improvements:

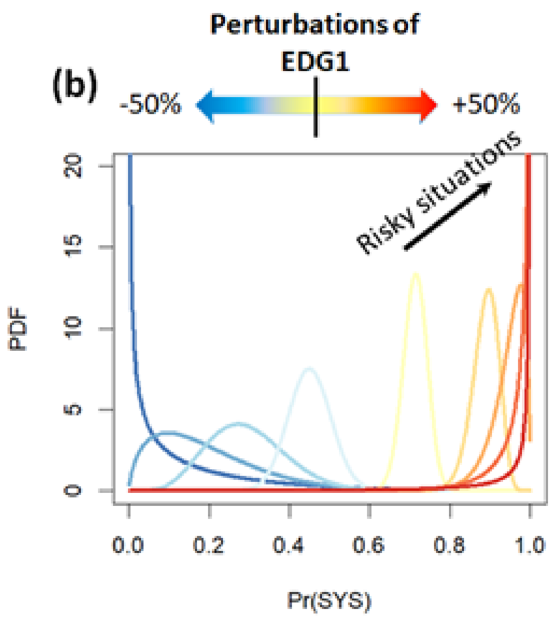
1) It incorporates uncertainty in the configuration of the safety systems involved in PIE initiated accidents, and
2) it requires compliance with additional acceptance criteria with an increased tolerance level in order to avoid possible cliff-edge effects. Both features contribute to a better implementation of Defence-in-Depth principles.


Start E-BEPU
Assign PIE to class and determine acceptance criteria
1
Identify sequences (with or without failures) with $P_{seq} > P_{cutoff}$
2
Perform uncertainty analysis for each sequence
3
All sequences fulfil acceptance criteria with tolerance level (95/95)
4
yes
no
Collect outcome from Block 3 and evaluate acceptance criteria for PIE
8
PIE fulfils acceptance criteria with tolerance level (95/95)
9
no
yes
Sequence fulfils acceptance criteria with tolerance level (95/95)
10
yes
(successful sequences)
no (failed sequences)
Determine acceptance criteria of next class
5
For damage sequences: $P_{seq} < P_{reclassify}$
11
no
Sequence fulfils acceptance criteria with tolerance level (99/95)
6
no
yes
Determine acceptance criteria of next class
12
Sequence fulfils acceptance criteria with tolerance level (99%/95%)
13
no
yes
Design is acceptable
7
Design is not acceptable
15
Design is acceptable
14

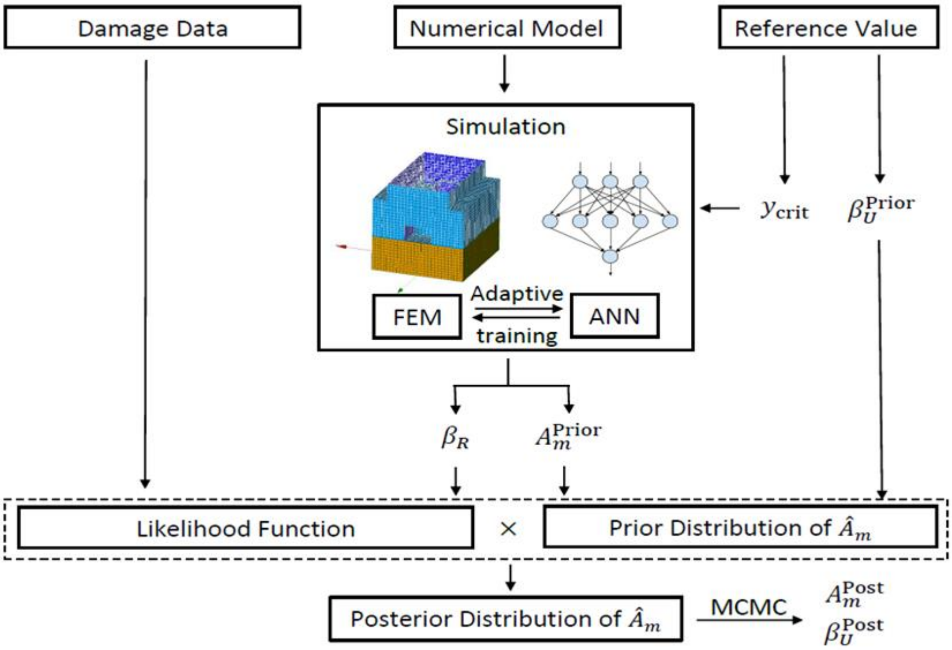
To constraint uncertainty, a new approach named "Boosted Beta Regression" has been developed. The approach has the advantage of being generic and robust. Performance assessment of this new tool has been done using two real cases including the study of station blackout following an earthquake affecting a given NPP sub-system.



An example of sensitivity analysis result is provided hereafter showing how perturbing the value of the fragility curve of an Emergency Diesel Generator (EDG) impacts the probability of occurrence of on-site power loss Pr(SYS), leading to a situation of "quasi-systematic" system failure when perturbation is varied by +50%.



To constrain uncertainty in fragility assessment, a Bayesian updating framework has been proposed, by combining an Artificial Neural Network, an adaptive training algorithm and an amplification-factor-based construction of the likelihood function. The framework allows for an improved seismic capacity estimation and reduces epistemic uncertainties in the fragility curves based on information from experience feedback.



Two technical presentations complement the multi-risk integration framework for safety analysis:

- *Comparison of Bayesian Networks with existing PSA approaches;*
- *Highlights on BN integration components: Constraining uncertainties components' modelling & expert-based information.*

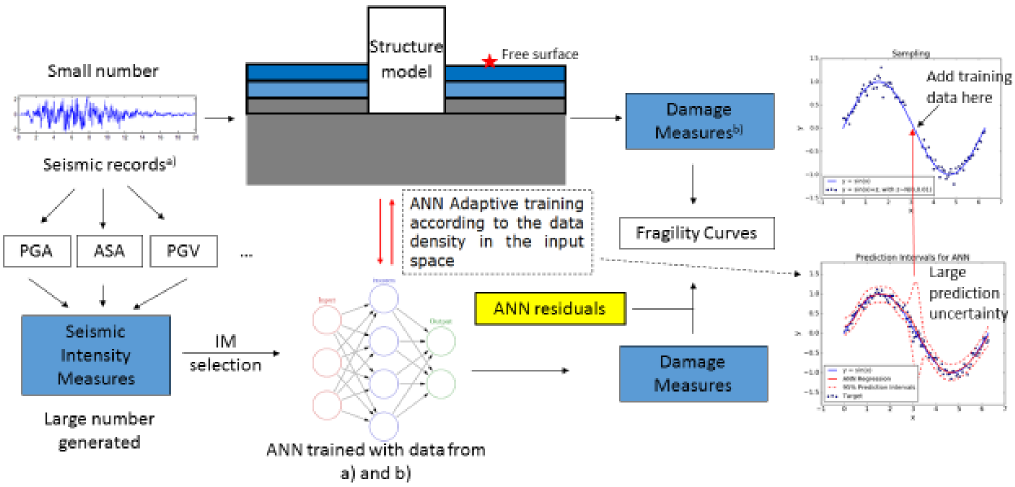
Topic4: Applying and comparing various safety assessment approaches on a virtual reactor

Main objectives: Propose possible modelling reduction strategies, which could be compatible with safety analyses and uncertainty assessment. Test the applicability of the proposed multi-risk integration methods (BNs, E-BEPU) for the safety analysis of a simplified theoretical NPP representative of the European fleet.

A simplified referential model representing a virtual PWR of the European fleet, focussing on the reactor, containment, and associated systems was proposed with selecting a number of safe shut down paths. Mainly inspired by Gen III technology, the model is driven by high-pressure failure scenarios i.e. Station Blackout (SBO) and Loss of Ultimate Heat Sink (LUHS), and low-pressure scenario i.e. Large Break Loss of Coolant Accident (LB LOCA). A detailed plant description was provided in order to be used for safety analyses including high-fidelity finite element simulations. In addition, fault trees modifications were proposed to switch from NARSIS plant to a GEN II model reactor.

Regarding modelling reduction strategies, several metamodeling approaches have been investigated. Metamodeling replaces the physical numerical model by a simplified expression linking the input to the output taking into account uncertainties in the predicted results. Running at negligible cost compared to a full model, sensitivity analysis and uncertainty propagation become computationally feasible.

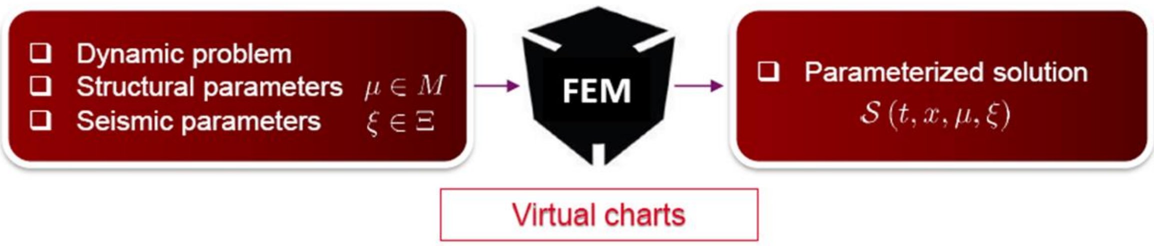
Two different methodologies have been proposed for seismic risk assessment. The first one based on Artificial Neural Networks (ANN) is used to derive the relations between seismic IMs and Engineering Demand Parameters (EDPs) of the structures, thus accelerating the fragility analysis. Fragility curves can then be evaluated using direct Monte Carlo simulations by assuming a lognormal model and applying linear regression techniques. The methodology allows for vector-valued fragility curves. The ANN prediction uncertainty was also investigated and quantified. This methodology has been successfully applied to estimate the probability of failure of an electrical cabinet in a reactor building studied in the framework of the KARISMA benchmark.



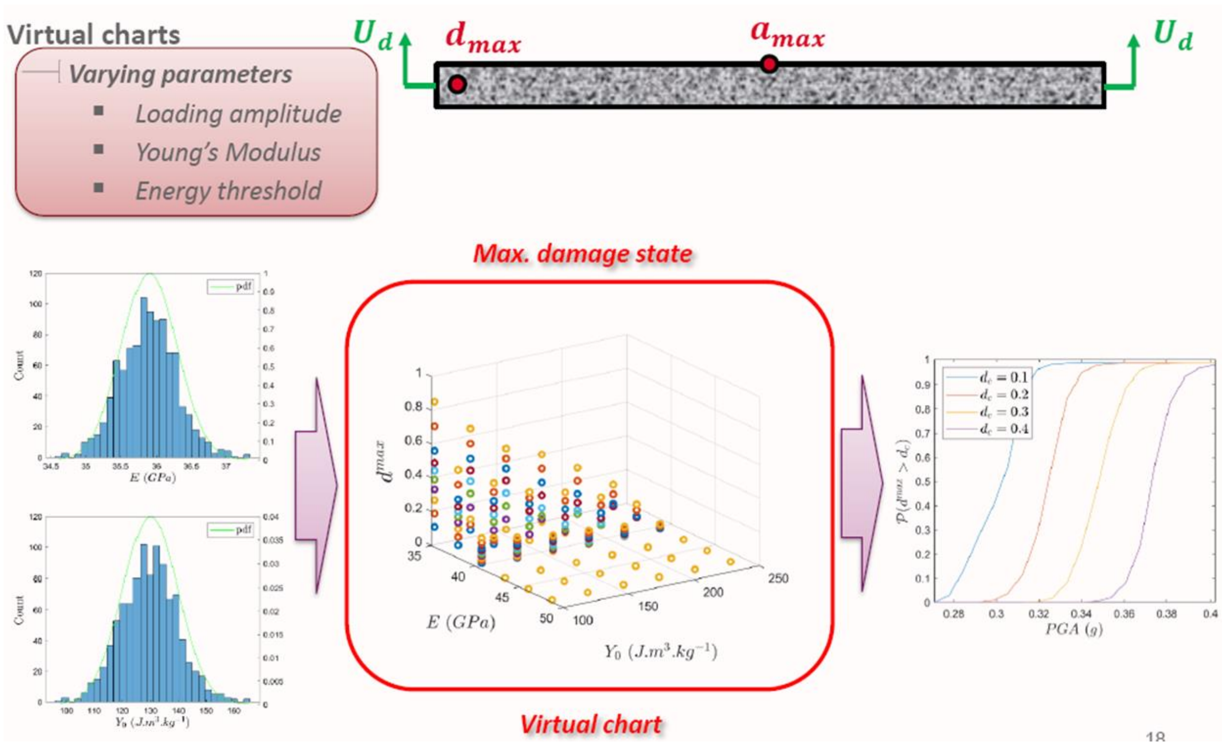
The second one is based on Support Vector Machines (SVMs) coupled with an Active Learning Algorithm. This methodology adopts SVMs to achieve a binary classification of structural responses relative to a limit threshold of exceedance. Since the SVM output is not binary, but gives a real-valued score, a probabilistic interpretation of this score is introduced to estimate fragility curves very efficiently.

A metamodeling methodology based on kriging was proposed for earthquake-induced tsunami hazard assessment, which is able to account for uncertainties on the scenario parameters (epicentre location, rupturing fault size, slip displacements). The kriging approach enables to learn in a nonparametric manner, the statistical link between the scenario parameters and the tsunami hazard IM, namely the maximum Sea Surface Elevation (SSE) at the coast. The kriging metamodels are used in place of some long-running simulations within a Monte-Carlo setting to evaluate the cumulative probability of SSE given the uncertainties on a worst-case scenario.

For highly nonlinear dynamic systems, a non incremental resolution method including novel model-order reduction technique was implemented for seismic fragility assessment. A LArge Time Increment (LATIN) solver for nonlinear problems involving an alternative sequence of nonlinear and linear stages, combined with a Proper Generalized Decomposition (PGD) which offers a conducive framework for obtaining parametric solutions in the linear range, form a suitable tool for computing virtual charts for structural response.



The numerical chart is then used for computing fragility curves by simple interpolation. The figure below presents the whole process and results obtained when simulating the seismic response of a 6m concrete beam supported by moving supports.



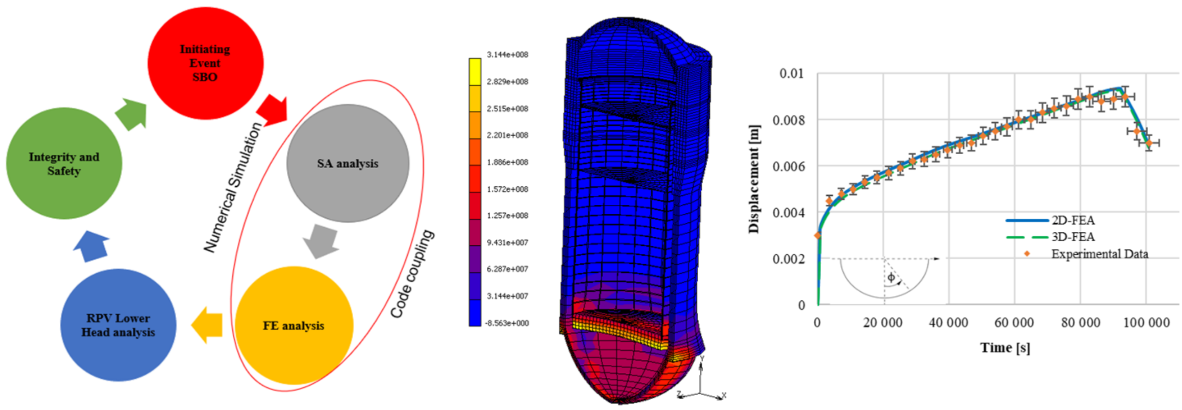
The reactor safety analysis considering deterministic and probabilistic approaches, was based on the referential Gen III NPP used as a virtual plant. It covered three main areas:

- New and existing methods for deterministic analysis in case of severe accident;
- Fully probabilistic analysis with BN application and comparison with a more traditional PSA approach based on Fault Trees and Event Trees in case of single and multiple hazard scenarios;
- Combined probabilistic-deterministic analysis (E-BEPU), which was applied for the first time in safety analysis.

Different topics were studied, including:

- sensitivity and uncertainty analyses with uncertainty quantification for severe accident;
- analysis of the impact of aged components during severe accidents;
- A severe accident analysis with source term investigation was also conducted on the reference Gen-III plant, virtually located on a Polish nuclear site.

A new methodology to study the impact of ageing, mainly material obsolescence, thermal degradation and creep phenomena, was implemented and demonstrated on a reactor pressure vessel (RPV) under station blackout (SBO). This methodology couples a severe accident analysis, based on the Melcor code, which simulates the plant behaviour and provides boundary conditions for a finite element code which can then predict the thermo-mechanical response of the selected components.



BNs were studied and developed for the accident scenario including subnetworks developed for the Loss Of Offsite Power (LOOP) with the SBO. Comparisons between fault trees and BNs methods were performed, including the new BN-based approach to CCF modeling. The more traditional PSA approach developed for multi-hazard scenarios involving earthquake and flooding events, was also compared with the BN method for the SBO scenario.

BNs provide an advantage in fault diagnostics in that new evidence can be easily incorporated into the model. However, with increased common cause effects, BNs can grow in size, making visualization and computations challenging.

Combined Deterministic and Probabilistic Methods – E-BEPU aims to demonstrate the existence of larger safety margins that could be utilized for more extensive operational flexibility or plant modifications. At the same time, E-BEPU allows for the introduction of new criteria oriented to address better other aspects of the plant safety, such as defense-in-depth.

This procedure, as presented in topic 3, was applied, tested, and demonstrated with plant scale analysis for the first time during the NARSIS project. For the sake of comparison, a more traditional BEPU-like study was also performed, showing the superiority of E-BPU methodology.

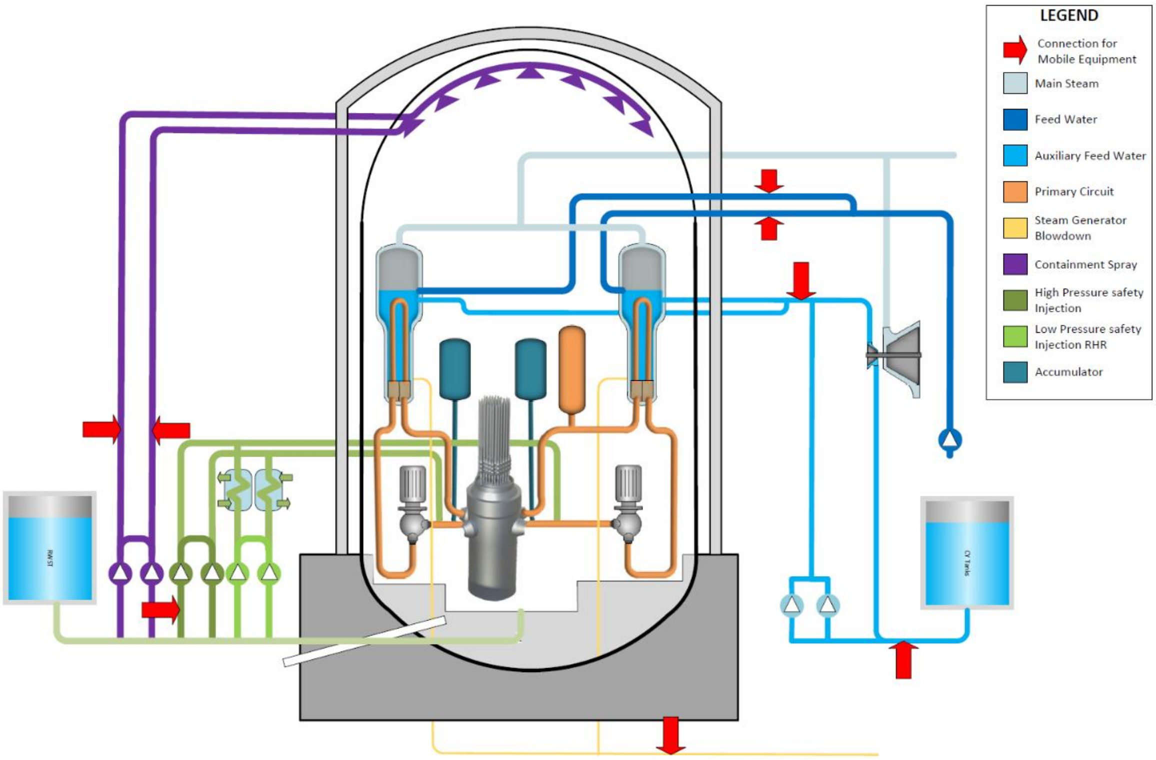
Topic 4 presentation was completed with 4 additional technical talks:

- *Metamodelling strategies for seismic and Tsunami PSA;*
- *Applying the E-BEPU methodology;*
- *PSA related to combined earthquake and flooding;*
- *Sensitivity & Uncertainty Analyses with uncertainty quantification for Severe Accident.*

Topic5: Supporting Decision-Making tool for Severe Accident Management

Main objective: Develop a demonstrative decision supporting tool for severe accident management, in order to make appropriate decisions in a timely manner. The tool should help technical support centre during management and training.

In the first step, a “referential” Gen II plant in terms of critical systems and structures was characterized. The referential NPP had two loops, large dry containment and safety systems for design basis, and design extension condition (DEC) accident management, including severe accidents.



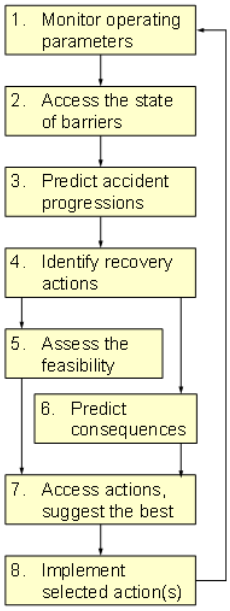
In the second step, “referential”, Emergency Operating Procedures (EOP), Extreme Damage Management Guidelines (EDMG) and SAMG (Severe Accident Management Guidelines) were characterized. Then, the hazard damage states and the logic model for accident progression, to be used as a basis for the mentioned supporting tool were stablished.

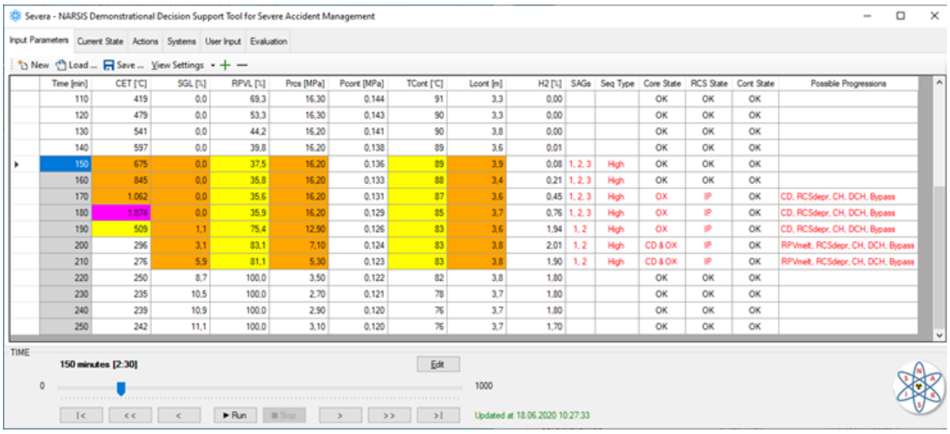
The main plant functions and corresponding systems related to hazard damage states were identified as follows:

- Reactor Coolant System Depressurization,
- Low Pressure (LP) Emergency Core Cooling Injection,
- LP Emergency Core Cooling Recirculation,
- Containment Spray Recirculation, and
- Containment Cooling.

More than 25 accident sequences with different actuation time of safety equipment activation were pre-calculated.

The decision modelling (DM) computer program is called Severa. Its flow chart is presented below.

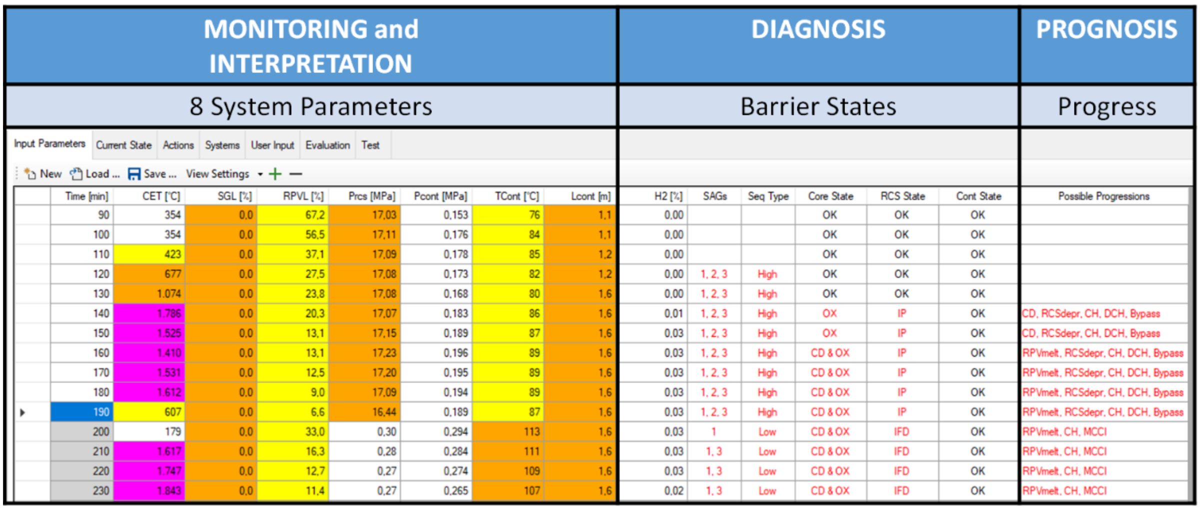




Time [min]	CET [°C]	SGL [%]	RPVL [%]	Prcs [MPa]	Pcont [MPa]	TCont [°C]	Lcont [m]	H2 [%]	SAGs	Seq Type	Core State	RCS State	Cont State	Possible Progressions
110	419	0,0	69,3	16,30	0,144	91	3,3	0,00			OK	OK	OK	
120	479	0,0	53,3	16,30	0,143	90	3,3	0,00			OK	OK	OK	
130	541	0,0	44,2	16,20	0,141	90	3,8	0,00			OK	OK	OK	
140	597	0,0	39,8	16,20	0,138	89	3,6	0,01			OK	OK	OK	
150	675	0,0	37,5	16,20	0,136	89	3,9	0,08	1, 2, 3	High	OK	OK	OK	
160	845	0,0	35,8	16,20	0,133	88	3,4	0,21	1, 2, 3	High	OK	OK	OK	
170	1.062	0,0	35,6	16,20	0,131	87	3,6	0,45	1, 2, 3	High	OX	IP	OK	CD, RCSdepr, CH, DCH, Bypass
180	1.874	0,0	35,9	16,20	0,129	85	3,7	0,76	1, 2, 3	High	OX	IP	OK	CD, RCSdepr, CH, DCH, Bypass
190	509	1,1	75,4	12,90	0,126	83	3,6	1,94	1, 2	High	OX	IP	OK	CD, RCSdepr, CH, DCH, Bypass
200	296	3,1	83,1	7,10	0,124	83	3,8	2,01	1, 2	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
210	276	5,9	81,1	5,30	0,123	83	3,8	1,90	1, 2	High	CD & OX	IP	OK	RPVmelt, RCSdepr, CH, DCH, Bypass
220	250	8,7	100,0	3,50	0,122	82	3,8	1,80			OK	OK	OK	
230	235	10,5	100,0	2,70	0,121	78	3,7	1,80			OK	OK	OK	
240	239	10,9	100,0	2,90	0,120	76	3,7	1,80			OK	OK	OK	
250	242	11,1	100,0	3,10	0,120	76	3,7	1,70			OK	OK	OK	

The Severa decision-support tool consists of two main parts: diagnostic and prognostic. The purpose of the diagnostic is to establish basic facts about the status of the severe accident sequence, based on the feedback in the form of a set of pre-selected parameters. The prognostic part supports the user in evaluating existing options / alternatives for accident management and mitigation, depending on the available means by identifying those actions that can be implemented, under their predefined priorities, resulting in the lowest risk from radioactive release.

Severa incorporates variety of modelling techniques for decision support and can be used for training of technical support teams.



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
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
210	1.617	0,0	16,3	0,28	0,284	111	1,6	0,03	1, 3	Low	CD & OX	IFD	OK	RPVmelt, CH, MCCI
220	1.747	0,0	12,7	0,27	0,274	109	1,6	0,03	1, 3	Low	CD & OX	IFD	OK	RPVmelt, CH, MCCI
230	1.843	0,0	11,4	0,27	0,265	107	1,6	0,02	1, 3	Low	CD & OX	IFD	OK	RPVmelt, CH, MCCI

Invited presentations

During the workshop, experts from outside the NARSIS consortium were invited to give four talks. Below is a summary of their content.

Surrogate models for uncertainty quantification and structural reliability

by Prof. Moustapha Maliki from ETH Zurich

Computational models are nowadays a standard tool for the design of manufactured products and structures. Algorithmic advances and a larger availability of computational resources have indeed favoured the systematic use of computational models to simulate with high fidelity the behaviour of engineering systems.

In practice, faithfully predicting the performance of a system requires that its input parameters are well known. However, there are numerous uncertainties related to either the parameters themselves (*e.g.,* manufacturing tolerances) or the system's environment (*e.g.,* loading) that need to be accounted for. This is generally done in the context of uncertainty quantification. More precisely, uncertainty quantification aims at identifying the sources of uncertainty and assessing their impact onto the system's performance. Its scope includes reliability analysis, Bayesian inversion/calibration, design optimization or sensitivity analysis.

Various methods have been developed for solving each of these problems. They however all share the common trait of being computationally intensive. Surrogate models have been increasingly used in the past two decades to propose computationally tractable solutions. They consist in building an inexpensive proxy of the computational model that can be used in subsequent analyses.

The importance of surrogate models in uncertainty quantification is discussed with a focus on two surrogate model types, namely polynomial chaos expansions and Kriging. Finally, how surrogate models can be used in the so-called active learning framework to efficiently solve complex structural reliability problems is presented.

The State-of-the-Art for Multi-Unit and Multi-Hazards PSA in the U.S.

by Dr. Dennis Henneke from GE-Hitachi Nuclear Energy

A growing concern regarding probabilistic safety assessments (PSA) is the impact of dependencies among reactor units co-located at a nuclear power site, especially after Fukushima accident. To address these dependencies and identify the critical contributors to the entire site risk, multi-unit probabilistic safety assessment (MUPSA) has been actively developed by various research and regulatory agencies. Experts involved in MUPSA recognize that External Hazards, such as seismic, flooding, and weather events, result in highly correlated core damage frequencies (CDFs) and Releases. CDF on one unit can greatly impact the operator response on another unit.

Most Small Modular Reactors (SMRs) such as PRISM are being designed to have numerous plants on a single site. MU Risk is likely the dominant public risk. A two-year US Department of Energy (DOE) project developed the next generation Probabilistic Risk Assessment (PRA) methodologies for the modernization of advanced non Light Water Reactor (non-LWR) PRA. This effort built upon a PRA developed in the early 1990s for GE Hitachi's (GEH's) PRISM Sodium Fast Reactor (SFR). The IAEA developed Safety Report on MUPSA methodology bringing together the extensive work performed in Phase I (methodology for implementation of MUPSA with practical PSA modelling guidance) and Phase II (development of a case study) and provides a comprehensive and tested methodology, which can be used by LWRs and non-LWRS to perform a MUPSA. The US NRC are working on improved approaches for hazard analysis including hazard combinations. Significant work is occurring in all hazard areas including external flooding, high winds, seismic PRA, Multi-Unit PRA, and other external hazards, generally in support of PRA efforts for the upgrade of the PRA standard for the post-Fukushima improvements.

Expanding PSA horizons - IAEA perspective

by Dr. Shahen Poghosian from IAEA

Probabilistic Safety Assessments (PSA), which was introduced as complement to deterministic safety analysis methods, is now a well-established tool that supports design and safety assessment of nuclear facilities, and risk-informed decision-making. However, traditional PSA approaches also have their limitations, specifically in meeting the challenge of identifying realistic scenarios in complex and dynamic systems. Therefore, the IAEA is encouraging the use of Advanced Probabilistic Safety Assessment to get a deeper view of a nuclear power plant's risk profile than that offered by traditional PSAs, thereby contributing to decision-making that can strengthen safety. The Agency is offering trainings, workshops, events and forthcoming publications to highlight the potential of Advanced PSAs.

The development of Advanced PSAs began decades ago to broaden the perspective gained from the static PSAs. Advanced probabilistic approaches provide a more accurate view of plant behavior in response to initiating events. Compared to conventional assessments, advanced PSAs include more realistic modelling of safety systems and structures, as well as the human-machine interactions.

Results of recent IAEA project on Multi-unit (MU) Probabilistic Safety Assessment is presented. The challenges and benefits of multi-unit risk assessment including risk-informed decision-making and risk management in a multi-unit context are discussed.

The talk also addresses the sustainable effort for enhancing nuclear safety addressing new technologies such as Small Modular Reactors and the use of PSA for security purposes.

Probabilities in regulatory hazard studies for the Chemical industry in France

by Dr. François Massé from INERIS

Accidental events in chemical industry can cause damages to human health, environment and economy. To prevent such events in industries, it is essential to identify and analyze the past events. Since 1992, the French database ARIA (Analysis, Research and Information on Accidents) collects information on accidents and incidents involving dangerous chemicals in classified installation transport of hazardous materials and other areas such as pressure equipment, mines and quarries, underground storage, as well as dams and dykes. Since the series of chemical disasters in recent decades, the French regulation has been reinforced. The policy to reduce accidental risk relies on four pillars, namely Safety Report, Land Use Planning, Emergency Planning and Public Information. The presentation describes the principles and processes related to each pillar. Risk analysis and safety barriers required for minimizing risks through calculation of the average annual probability of occurrence of hazardous phenomena and the use of semi-quantitative approach for the rating of Risk Reduction Factors are discussed.

Partners