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NARSIS

New Approach to Reactor Safety ImprovementS

Newsletter # 2

Edito

Welcome!

James Daniell - NARSIS WP1 leader

Dear reader
A warm welcome to the second issue of NARSIS Newsletter!

NARSIS coordinates the research efforts of eighteen partners encompassing leading universities, research institutes, technical support organizations (TSO), nuclear power producers and suppliers, reactor designers and operators from ten countries. The project aims at making significant scientific updates of some elements required for the Probabilistic Safety Assessment (PSA), focusing on external natural events such as earthquake, tsunami, flooding, high speed winds etc.

The NARSIS project has now been running for a year, and the first set of deliverables and milestones have been produced as part of the effort of the consortium. Datasets have been collected, methodologies tested, the state of the art has been researched, and various criteria and plans developed. The first year plenary meeting was held from the 18th to the 20th of September at the Karlsruhe Institute of Technology with over 40 members joining the 3 days of discussions, presentations, working groups and activities.

As part of the multi-hazard framework WP, a state of-the-art (D1.1) has been undertaken incorporating various facets of methodologies for single and multi-hazard, past disasters, stress test reviews as well as various definitions of natural external events (occurrence of concomitant external events, either simultaneous-yet-independent hazards or cascading events).

The definitions and inventory of the physical and operating fragility of main SSCs (Systems, Structures and Components) present in Nuclear Power Plants (NPPs) (D2.1) have been provided.

Various studies and classification rankings have been undertaken as part of this deliverable as a key first step towards the multi-hazard fragility functions and modelling to be developed in the coming years within NARSIS.

Within WP3, a key review (D3.1) and comparison of risk integration methods from high-risk industries has been undertaken with a particular emphasis on methods incorporating low probability events, multi-hazard frameworks and previous lessons learned, with insights into the potential better risk integration to be explored through Bayesian Belief Networks (BBNs).

The proposed improvements will be tested and validated on simplified and real NPP case studies. The simplified theoretical NPP representative of the European fleet has been discussed at length between partners in the Slovenia meeting, 6 months into the project, and the D4.1 has been produced with this input as well as criteria for model reduction necessary to be relevant for PSA. This has been done in conjunction and consultation with WP5 where characterization of the referential plant in terms of critical systems and structures has been set up, in the first steps toward a demonstration of a supporting decision tool for Severe Accident Management (SAM).

Although it has been a busy period, the initial deliverables of the five work packages set the foundations for the integrated approach towards multi-hazard risk assessment and studies into the integration within PSA.

With this newsletter, we would like to broaden the circle and share the outcomes of our project with larger audience. Our objective is to attract wide support from and involvement of any stakeholder interested in cooperative development of the nuclear safety. This newsletter aims to function as an information tool for disseminating results and outcomes of our project but also to become a forum for discussion, reflection and dialogue. Our conceptual strategy is anticipative, reflecting our wish to involve more researchers, professionals and interested groups in the debate including through our web site www.narsis.eu.

We will be happy to receive your comments and suggestions. Please feel free to communicate your feedback to Prof. Behrooz Bazargan Sabet (b.bazargan-sabet@brgm.fr) for inclusion in our forthcoming issues. We would also like you to help us disseminate this second newsletter to your network.

We look forward to hearing from you!

WP summaries

WP1: Characterization of potential physical threats due to different external hazards and scenarios

James Daniell
KIT

A large amount of literature and models have been reviewed as part of this attempt to define a state-of-art in multi-hazard analysis for Nuclear Power Plants (NPPs) in the first year (WP1.1). Many methodologies, software packages and datasets have been developed globally over the last decades for both probabilistic and deterministic hazard analysis of natural catastrophes. These tools have fed the production of potential external hazard scenarios and return periods for NPPs as part of PSA (Probabilistic Safety Assessments) and Screening analysis.

A huge amount of external hazards from natural catastrophes exist – over 70 as determined by the ASAMPSA-E project of geophysical, meteorological, extra-terrestrial, biological, hydrological and climatological origin with various combinations of these events possible. Each hazard type interaction (coincident, causally correlated, mutually exclusive, direct) needs to be examined in a multi-hazard assessment.

A large number of historical single and multi-hazard events have been reviewed as part of this work, including large events such as the Tohoku 2011 earthquake and tsunami which will have a long lasting impact on the nuclear industry.

Over 60 natural hazard events have been identified affecting in some ways NPPs in Europe. In most cases however, the damage was not extensive. Still, many more events not affecting NPPs have been identified from history. In fact, for earthquakes, 30% of all fatalities have not been from shaking but from secondary effects such as tsunami or landslide. Similarly, we often see for tropical cyclones that storm surge and rainfall cause more fatalities than the pure wind losses themselves.

A review of the stress tests for European NPPs including various discussions with stakeholders shows the key design parameters for earthquake, flood and precipitation, using the national and individual plant reports for each of the available NPPs in Europe. The multi-hazard aspects however, are not touched upon in nearly all cases, thus the need for this project.

NARSIS takes benefit of the recent ASAMPSA_E project, which aimed to examine in detail how far the PSA methodology is able to identify any major risk induced by the interaction between a NPP and its environment, and to derive some technical recommendations for PSA developers and users.

For the key hazards identified to affect NPPs across Europe, earthquakes, tsunami and wave, extreme weather effects (heat/cold wave, hail, precipitation etc.), and flooding, empirical data for Europe has been collected and examined as well as a discussion of empirical events collected from various scientific papers, projects and industry briefs. Methodologies have been put forward for the state-of-art assessment in deterministic or probabilistic methodologies for the perils albeit via extreme value statistics of empirical data, with Monte-Carlo simulation to produce a stochastic event set; or Probabilistic Hazard Assessment (PHA) using historical regression of disaster data via physical hazard zones and lognormal relations (or forms of it).

Key input parameters, datasets and metrics have been examined for each of the main types, as well as how uncertainty is examined as part of the analysis framework. A review of such hazard curves and combination methodologies is thus made for singular and secondary hazards. The step from single to multi-hazard analysis and the review of various frameworks suggests that this field is rapidly evolving with a significant increase in literature associated with multi-hazard in the last 5 years (in part due to the Tohoku event).

Various methodologies such as multivariate analysis and multi-hazard combinations of curves, have been undertaken by many authors at a global, regional and local scale. With respect to NPPs, it can be seen from the stress test review and some other details that correlated hazards have rarely been used as part of design, however using the frameworks found, this should allow for the initial steps for the production of a software based on upcoming WPs, and the software review as part of this analysis in the next year.

A number of workshops are planned for the coming 6-month period for work on the various single hazard methodologies as well as the setup of a test scenario on decommissioned nuclear sites using multi-hazard curves, which can be used within the other work packages as an initial cross-cutting study.

WP summaries

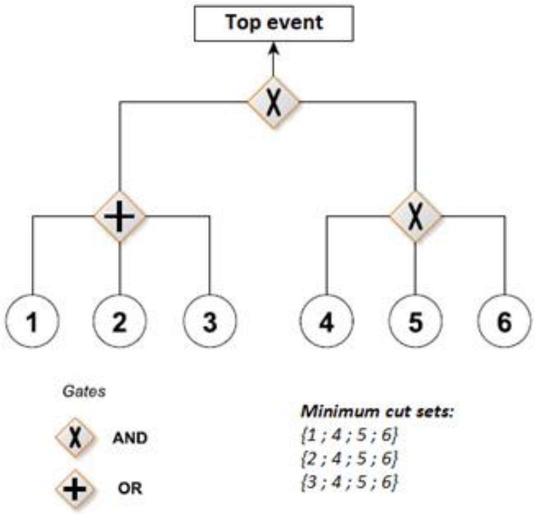
WP2: Fragility assessment of main NPPs critical elements

Pierre Gehl

BRGM

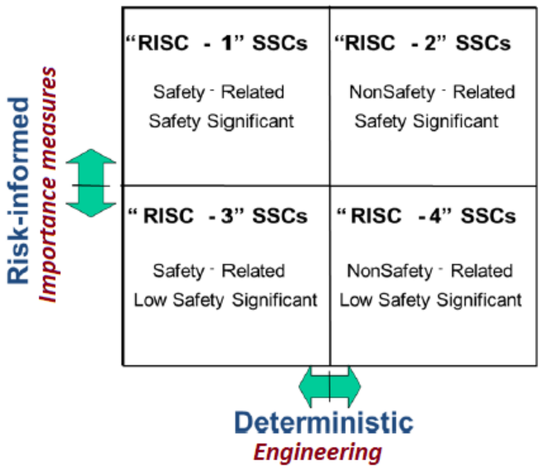
Within the first year of the project, developments in Work Package 2 have started with the completion of a first deliverable report, on the inventory and selection of the most critical SSCs within a Nuclear Power Plant (NPP).

This preliminary step constitutes a crucial milestone for the subsequent tasks, as it will condition which SSCs to consider when developing specific models for the fragility assessment, with respect to multiple natural hazards. To this end, screening approaches, based on the outcomes of Probabilistic Safety Analyses (PSA) or on a risk-informed categorisation of SSCs, have been detailed. For instance, PSA are useful to identify *critical failure paths* of SSCs leading to an undesired top event (e.g., minimum cut sets, which represent the smallest sets of SSCs that need to fail in order to induce the failure of the system): as a result, quantitative importance measures may be assigned to the different SSCs (e.g., Fussell-Vesely importance measure).



Example of a fault-tree with the identification of three minimum cut sets with respect to the top event.

A qualitative risk-informed categorisation of SSCs may then be carried out, following the NRC 10 CFR 50.69 guideline, which advocates the integration of risk-informed measures and traditional engineering insights within a single decision-making process.



Four risk-informed safety classes (RISC), according to the NRC 10 CFR 50.69 Guideline.

Safety-related SSCs are defined as the ones that must remain functional during and following design-basis events, in order to ensure the fundamental safety functions, i.e. the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and to maintain it under safe shutdown conditions, and the capability to prevent or mitigate potential offsite exposures.

Exploiting the results of previous case studies of actual NPP, various metrics based on either seismic PSA or seismic margin assessment have been used in order to derive a normalized importance indicator for the considered SSCs. As a result, the following groups of SSCs have been identified as critical elements, as far as the progression of accidents to core damage (level 1 PSA) is concerned:

- I&C and switchgear cabinets/devices;
- Reactor pressure vessel internals, and especially fuel assembly spacer grids;
- Distributed systems such as HVAC, piping or cable raceways (the risk footprint of distributed systems ranks high because of the need to adopt conservative assumptions, in order to limit analytical work).

Although the above results have been obtained with respect to seismic hazard only, similar approaches are applicable to other external hazards events, such as flooding, tsunami or high wind. However, PSA (especially the fragility assessment part) for hazards other than earthquakes have to reach a sufficient level of maturity in order for them to yield quantitative and exploitable importance measures; while seismic PSA have historically benefited from wider research efforts.

Regarding the other tasks within Work Package 2, on-going technical work is focusing on the improvement of mechanical and structural numerical models, in order to better account for effects other than wave propagation in the case of seismic hazard (i.e., soil-structure interactions, ageing mechanisms and cumulative loadings). In particular, an internship jointly conducted by *CEA* and *BRGM* has studied the feasibility of applying probabilistic fatigue methods (widely used in the reliability assessment of mechanical parts) to the case of structural elements: the number and history of loading cycles may then be taken into account, resulting in the derivation of cumulative damage curves. While preliminary results are promising on the case of a concrete beam, it is expected to pursue the efforts the following year by extending the procedure to reinforced-concrete assemblies.

Finally, methods for the derivation of fragility functions with multiple intensity measures have been studied further, with a first application on a structure/equipment assembly subjected to seismic loading. This example has led to the assessment of the adequacy of various combinations of ground-motion parameters as intensity measures: the use of different engineering demand parameters for different types of elements requires to consider vector-based intensity measures that are able to capture the complexity of a given ground-motion time history. The preliminary conclusions of this first methodological study have been presented by *BRGM* at the TINCE 2018 conference in France (www.sfen-tince2018.org). The next step will be the integration of other external hazard loadings, either as concomitant or successive occurrences, in order to derive fragility models that are able to account for various types of hazard combinations and potential cascading effects.

WP summaries

WP3: Integration and safety analysis

Phil Vardon
TU Delft

Work package 3 is related to the quantification and integration of risks. Nuclear power plants are complex systems, with many components which interact. Typically these systems are quantified only considering the technical aspects; moreover these systems may have organisation/human aspects and the systems are dynamic, i.e. they change over time. Systems in nuclear power plants are designed to be robust and demonstrably work the vast majority of the time. Additionally, serious external hazards also occur infrequently. This means that there is little information available in times where systems do not operate as intended during severe external hazards, and quantification of probabilities is difficult. However, utilising probability allows uncertainties to be characterised and by doing so, sensitivities and weaknesses identified.

The work undertaken recently in this work package has been on understanding the current state-of-the-art. There have been many deterministic and probabilistic methods of integrating risk proposed and used, each with its own advantages and disadvantages. A series of well-known case studies have been presented, from both the nuclear and non-nuclear industries. What was common in all of these case studies, is while there was a technical weakness which caused the failure, this was often caused by an organisational failure.

Two approaches which will be further developed in this work package are the “Extended Best Estimate plus Uncertainty” (E-BEPU), which uses a probabilistic safety analysis to evaluate uncertainties, with the realistic estimate of behaviour being evaluated deterministically, and the Bayesian Belief Network (BBN) where all parameters are evaluated statistically, and there interactions are explicitly included. The BBN method offers a complete representation of processes, and therefore allows identification of weaknesses through thorough computational analysis, but can become complex and limitations in available data become significant.

The work package will now focus on the major technical tasks which are:

• Build a non-parametric dynamic BBN approach for a NPP by developing and integrating sub-networks for cause and consequence of technical, organizational and human aspects;

• Develop appropriate statistical techniques for constraining the input uncertainty sources of the BBN with a particular attention paid to expert-based information;

• Develop an E-BEPU approach, combining probabilistic and deterministic analyses.

WP summaries

WP4: Applying and comparing various safety assessment approaches on a virtual reactor

Giuseppe Rastiello
CEA

One of the main tasks of work package 4, which has been achieved during the first year period of the project, has consisted in defining a simplified theoretical NPP, based on the design of typical pressurized water reactors (PWR). The proposed NPP model is generic enough to be considered as representative of generations II & III PWRs of the European fleet. Similarities with European operating plants are present, but simplifications have been introduced to generalise and keep the model as theoretical as possible. This virtual NPP will be further used for verification and inter-comparison of existing and new methods for PSA.

The simplified theoretical NPP has been defined by focusing on the reactor, containment structures and associated systems. Critical systems and components were identified based on PSAs, considering several initiating events. With this term, we designate events (e.g. components' failure) that create a disturbance in the NPP such that countermeasures are required to prevent core damage. Three main initiating events were chosen in coordination with WP5: Loss of Offsite Power (LOOP), Large Break Loss of Coolant Accident (LB-LOCA) and Loss of Ultimate Heat Sink (LUHS).

Simplified Level 1 and Level 2 PSAs have been performed to study the responses of the containment and related systems to potential loads and to evaluate radioactive releases in case of severe core damage accidents. PSA models provided are fully detailed for the NARSIS partners (event trees), in order to implement them into various PSA codes (e.g. Saphire®).

Framatome (task leader), WUT and IRSN jointly prepared the related deliverable D4.1, which contains the main data (e.g. geometrical, mechanical, functional features) for the critical systems and confinement structures of the NPP. These data will be needed to work on reduced and meta-modelling strategies planned in task 4.2, as well as to conduct the reactor safety analyses planned in task 4.3, considering the external events scenarios proposed in WP1 and using the fragility models developed in WP2. Some meta-modelling strategies have already been tested, especially for the seismic probabilistic risk assessment. During the first year of NARSIS, Wang et al. (2018) have indeed combined Artificial Neural Networks simulations and real observations for Bayesian updating of fragility curves related to low voltage switch gear equipment, located in the reactor building of the Kashiwazaki-Kariwa NPP (Japan). Application of this meta-modelling strategy on the NARSIS theoretical NPP is expected in the next year period.

WP summaries

WP5: Supporting Tool for Severe Accident Management

Luka Štrubelj
Gen Energija

The objective of WP5 is to develop a tool for supporting decision-making (DM) in the severe accident management, relying on the PSA techniques. During the first year period of the work, the referential operating NPP has been described in deliverable D5.1, to serve as a reference for developing the demonstrative supporting tool for severe accident management analyses. A plant with PWR has been selected, as the most representative of European fleet considering, also, the availability of generic Severe Accident Management Guidelines and supporting analyses, probabilistic safety analyses (PSA) and others, which are necessary pre-requisites. This referential plant for the WP5 is a PWR of Generation II, located near a river, which represents the ultimate heat sink, and is cooled by once through cooling system. The reactor coolant system consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, a steam generator and connecting piping. Additionally, the system contains a pressurizer, pressure relief and safety valves, a pressurizer relief tank and the instrumentation necessary for operational control and safeguards actuation. The reactor vessel is a cylindrical structure with a welded, hemispherical bottom head and a flanged hemispherical upper head. The vessel contains the core, core support structures, control rods, neutron pads, and other parts directly associated with the core. The control rods are operated by sealed drive mechanism mounted on the upper head. Each of the two steam generators is a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the head to the tube sheet. Engineered safety features (ESFs) are provided to manage the design basis accidents including loss of reactor coolant and different transients. The ESFs include the Emergency Core Cooling System (ECCS), consisting of the high pressure safety injection (HPSI) system, accumulators and low pressure safety injection (LPSI) system. The HPSI and LPSI contain two redundant trains with pumps and associated valves and Refuelling Water Storage Tank (RWST) representing the borated water source for the ECCS injection phase. LPSI system is the residual heat removal system (RHR) during normal plant shutdown and outages. The two accumulators represent passive engineered safety features where a cover gas forces borated water injection; without need for an external source of power or actuation signal. Another important ESF is the Auxiliary Feedwater (AFW) system consisting of the two motor-operated pumps feeding the respective two steam generators and the one turbine-driven pump feeding both steam generators. In addition to its functions during normal plant shutdown the AFW also represents a safety class emergency feedwater. Transfer of heat from the ESFs to the ultimate heat sink is implemented by the Component Cooling Water (CCW) and Essential Service Water (ESW) systems. Both of these are safety class systems consisting of two redundant trains. Other ESFs include Chemical and Volume Control System (representing, also, a normally operating system), pressurizer safety valves, steam generator safety valves and main steam isolation valves. Two safety class emergency diesel generators are provided, one for each of the two safety class trains, to feed main safety class buses in the case of a loss of power. The referential plant has large wet-cavity containment with cylindrically shaped body, spaced between nearly spherical cups. The containment heat removal during a design basis accident is performed by the containment spray system in conjunction with LPSI heat exchangers or by the Containment Fan Coolers. Passive Containment Filtered Venting (PCFV) System is provided for protection against the potential slow over-pressurization in the case of severe accident conditions involving core relocation. Steam and power conversion system include Main Feedwater and Condensate Systems, Main Condenser with associated systems, Main Steam System with steam dump to atmosphere and to the condenser, Turbine, Generator and associated systems. Emergency power supply at referential operating plant for the WP5 was, for the purpose of coping with design extension conditions, extended with third emergency diesel generator.

In addition, there are different strategies for coping with design extension conditions, which rely on a number of mobile equipment and variety of alternative water sources. The report provides general information about the referential NPP, its reactor coolant system, the equipment and SSCs for Design Basis, and Design Extension Conditions (DEC) accident management, including severe accidents. The mobile equipment and possible connections to the NPP systems are described, as well as the spent fuel pool with associated systems for normal operation and accident management.

In task 5.2, the characterization of emergency operating procedures, extensive damage management guidelines, FLEXible coping strategies (FLEX) and Severe Accident Management Guidelines (SAMG) is ongoing and related to deliverable D5.2 currently under preparation.

In task 5.3, the definition of hazard-induced damage states and the development of state-specific Accident Progression Event Trees (APETs) for demonstration purposes is ongoing.

In task 5.4, the supporting SAMG decision making tool for demonstration purposes will be developed. Some ideas have been discussed regarding the key measurements needed from simulations and information provided to decision makers.

EVENTS

Technical Meeting on Reactor Core Management and Engineering in Operating Nuclear Power Plants
04 Dec 2018 - 07 Dec 2018 • Vienna, Austria
Event website:

https://www.iaea.org/events/technical-meeting-on-reactor-core-management-and-engineering-in-operating-nuclear-power-plants

2019 Power Plant Simulation Conference
20 Jan 2019 - 23 Jan 2019 • TAMPA, FLORIDA, United States
This conference focuses on the special needs of the nuclear and fossil power plant simulation community.
Event website:

http://scs.org/powerplant/

CONTE 2019: Conference on Nuclear Training and Education: A Biennial International Forum
05 Feb 2019 - 07 Feb 2019 • St. Augustine, FL, United States
Event website:

http://www.ans.org/meetings/c_2

Structural Materials Degradation Seminar
06 Mar 2019 - 08 Mar 2019 • Palma de Mallora, Spain

The objective of the 3-day Seminar is to give an overview of the world wide operational experience with major SCCs in nuclear power plants (PWR, BWR).

Event website:

http://antinternational.com/antia-seminars/structural-materials-degradation-seminar/

Partners