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Probabilistic Safety Assessment for internal and external events on nuclear power plants and on mitigation strategies/H2020 European projects NARSIS, R2CA and BESEP

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Abstract. The NARSIS project aimed at improving assessment methodologies to be integrated into “extended Probabilistic Safety Assessment” (PSA) procedures for nuclear plants in case of single, cascade and combined external natural events. An open-access framework tool has been released to build multi-hazard scenarios, and various risk integration approaches (e.g., Bayesian Networks) have been implemented and compared, identifying their advantages and limits for further collaborative research activities. The R2CA project aims at harmonizing the safety analysis methods for best estimate evaluations of the radiological consequences, in case of Design Basis Accidents and Design Extension Conditions without significant fuel melting. It is planned to improve models and upgrade existing simulation tools and calculation chains used in safety studies. Finally, the BESEP project is to support safety margin determination, by developing best practices for safety requirement verification against external hazards, using efficient and integrated set of Safety Engineering practices and PSA. The project is carried out in a benchmark exercise based on case studies previously performed by the consortium participants. All three projects aim to improve nuclear safety within the European research and development framework. The research objectives are achieved by the development and improvement of proven and justified safety assessment methodologies for the verification of stringent safety requirements of nuclear industry.

1 Introduction

The response to the 2011 Fukushima nuclear accident has led to stringent safety requirements in many EU countries. To verify the fulfilment of the stringent safety requirements, proven and justified safety analysis methods should be developed and applied in the nuclear industry. Such methods have been studied and developed for internal and external events and on mitigation strategies in three EU projects: NARSIS, R2CA and BESEP.

The NARSIS project (see Fig. 1a) aimed at improving assessment methodologies to be integrated into “extended Probabilistic Safety Assessment” (PSA) procedures for nuclear plants in case of single, cascade and combined external natural events. An open-access framework tool has been released to build multi-hazard scenarios, keeping only hazard parameters relevant for the safety assessment of the main critical plant structures, systems and components. Various risk integration approaches

(e.g., Bayesian Networks) have been implemented and compared as well, identifying their advantages and limits. The project achievements have led to recommendations useful for further collaborative research activities.

The R2CA project (see Fig. 1b) aims at harmonizing the safety analysis methods for best estimate evaluations of the radiological consequences in case of Design Basis Accidents and Design Extension Conditions without significant fuel melting. It is planned to improve models and upgrade existing simulation tools and calculation chains used in safety studies. Among results, some guidelines to design and implement new Accident Management Procedures and safety devices are expected, as well as the development of innovative approaches (e.g., artificial intelligence) for anticipated accidental situation diagnosis.

The BESEP project (see Fig. 1c) aims to support safety margin determination, by developing best practices for safety requirements verification against external hazards, using efficient and integrated set of Safety Engineering practices and PSA. The core of the project is a benchmark exercise based on case studies previously

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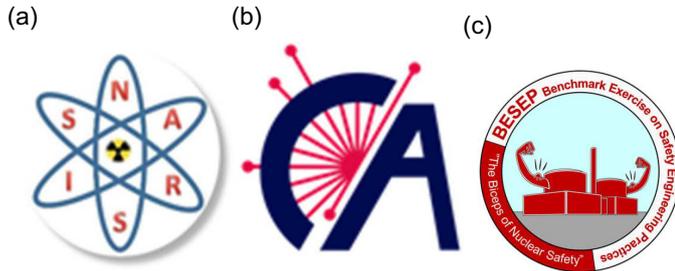


Fig. 1. The logo of (a) NARSIS; (b) R2CA; and (c) BESEP project.

performed by the consortium participants. In the benchmark, the performances of various safety analyses (i.e. Deterministic safety analysis, PSA and Human factors engineering) and Safety Engineering practices are compared to common safety requirements defined for the project. Expected project results are best practices and guidance for the verification of evolving and stringent safety requirements against external hazards.

A more detailed introduction to the NARSIS, R2CA and BESEP projects, their objectives, expected key results and dissemination activities are given in Chapters 2, 3 and 4 respectively. Finally, Chapter 5 is reserved for a short, general conclusion on the research projects.

2 The NARSIS project

2.1 Presentation of the project

In light of the Fukushima Daiichi nuclear accident in Japan (March 2011) and based on the FP7 ASAMPSEA lessons and on the outcomes from other European FP7 projects (e.g., SYNER-G, MATRIX, INFRARISK), the NARSIS project (New Approach to Reactor Safety ImprovementS, 2017–2022) aimed at investigating the possible improvements to be integrated into existing PSA procedures for NPP related to single, cascade and combined external natural hazards.

Two main interconnected components were part of the NARSIS methodology, organized in 5 main scientific work-packages WP1 to 5 (Fig. 2):

- reviewing existing methodologies and proposing some scientific improvements for multiple natural hazard assessment and their impacts, including advance in evaluation of uncertainties and reduction of subjectivity related to expert judgments (addressed in WP1, 2 & 3);
- verification and testing of the applicability and effectiveness of the improvements w.r.t. safety assessment, as well as demonstration of an improved decision supporting tool for operational and severe accident management purposes (addressed in WP4 & 5).

Thirty-six technical reports and two software tools have been produced by project partners. Most of these reports are publicly available on the project website (www.narsis.eu).

In the following sections, we give an overview of the main objectives and related achievements of the project.

2.2 The Multi-Hazard (MH) framework

One of the main objectives of NARSIS was to develop an integrated Multi-Hazard (MH) framework for safety assessment on main critical NPP Systems, Structures & Components (SSC), accounting for single, cascade and combination events at different time scales, focusing on earthquakes, flooding, tsunamis and extreme weather, as these hazards were identified as priorities by the PSA End-Users community in the European ASAMPSEA-E project. The MH framework implementation has been possible thanks to the works recalled hereafter:

- producing a complete state-of-the-art [1]:
 - many data from various sources, for the key hazards identified to affect NPPs across Europe have been collected as well as examined: earthquakes, tsunami and waves, extreme weather effects (heat/cold wave, hail, precipitation, etc.), flooding. Various methodologies for single and multi-hazard characterization and assessment have been reviewed, and various definitions of natural external events have been provided (e.g., occurrence of either simultaneous-yet-independent hazards or cascading events). Part of the work involved determining which hazards are more suitable for probabilistic or deterministic analysis and where improvements could lie in the assessment. Key input parameters and metrics have been also examined for each of the main hazards, as well as uncertainty, which forms a major part of the analysis, given the large variability of past events and simply the random nature of natural hazards. From the many historical events reviewed, more than 60 of them were identified as affecting NPPs in Europe, but with no extensive damages in most cases. Secondary effects (e.g., earthquake-triggered tsunami or landslides, storm surge/heavy rainfalls during tropical cyclones) have also been examined, as often causing more damages and fatalities than the primary hazards.
- Performing stress tests' review [2]:
 - the key design parameters for earthquake, flood, and precipitation have been derived from the review of the national and individual plant reports for each of the available NPP in Europe. This review has shown that the multi-hazard aspects (assessment for combined and/or cascading hazard events) were not addressed in most cases, thus comforting the need for a MH framework such as proposed by NARSIS.
- Improving Probabilistic Hazard Assessment (PHA) methodologies for tsunami, extreme weather and flooding hazards:
 - for tsunamis, contrary to the usual practice, NARSIS has implemented an accurate numerical tsunami propagation and inundation modelling approach, based on several nested bathymetric/topographic grids, characterized by a coarse resolution over deep-water regions and an increasingly fine resolution close to the shores and coastlines. Thus, specific

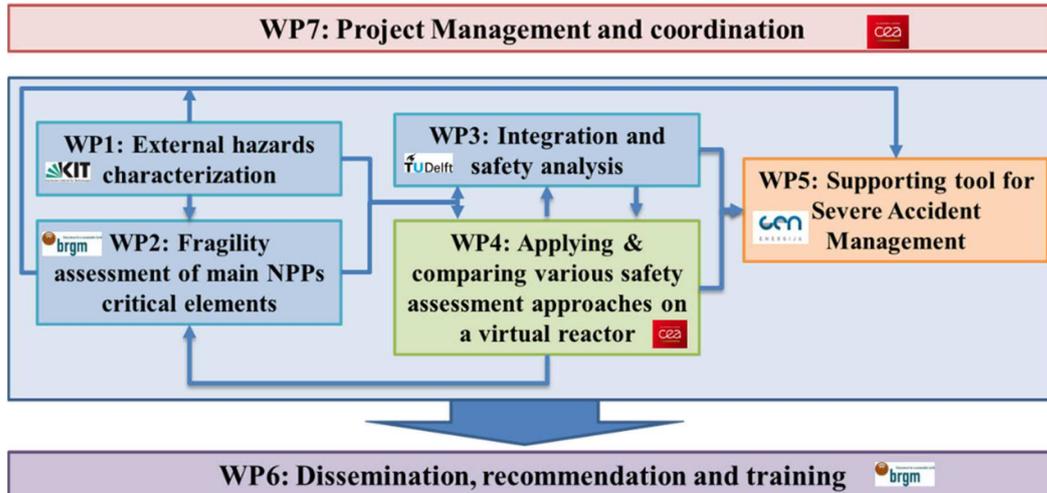


Fig. 2. NARSIS overall structure.

coastal responses, run-up and horizontal inundation could be assessed properly, together with the related uncertainties along the whole process (e.g., fault slip variability, in case of earthquakes, etc.). This approach has been applied to the French Riviera coastline and could be also applicable to other European regions, which are prone to tsunamis (e.g., Mediterranean, English Channel & South-Western Atlantic coastlines).

- For extreme weather and flooding, the current structural design codes are based on the assumption of stationary climate conditions, which are however no longer prevailing in the Climate Change context. Hence, the traditional reliability-based calibration approaches cannot be directly extended to non-stationary climate cases. For instance, the return period concept is no longer applicable due to a time distribution between event occurrences which is no longer invariant. Similarly, the annual failure probability is no longer constant in non-stationary climate conditions. These issues have been fully addressed in NARSIS using the stochastic processes and extreme value modelling-type approaches, as their sound mathematical framework allows to justify data extrapolations. In addition, some works have been dedicated to reviewing the different methods used for Uncertainty Quantification and Global Sensitivity Analysis in case of modelling input parameters considered as dependent, as most analyses rely on the assumption of independent variables of interest.
- Testing and refining the MH framework: careful site selection around Europe was important, in order to test and demonstrate the capabilities of the NARSIS framework. As a real NPP would never be located anywhere, creating a generic set of locations has been considered outside of the scope of the project. It was hence decided to analyse all NPPs in Europe including decommissioned and research plants to examine potential

sites for NARSIS analyses. Finally, three decommissioned and shutdown sites have been selected: Trino Vercellese in Italy, Mülheim-Kärlich and Biblis in Germany and their hazards' datasets have been fully examined and characterized, to produce single hazard curves. Station correlation analysis for extreme weather was undertaken as part of the study, as well as multivariate modelling, looking at correlations between various dependent parameters/station measurements.

The screening of the main NPP SSC was included in the analysis, in order to keep only relevant hazard parameters for each hazard in the final MHE software.

The final NARSIS methodology is derived from the FP7 MATRIX approach [3], which was based on 3 levels for analysis (qualitative, semi-quantitative, quantitative). In order to match the NPP specific nature, the NARSIS methodology implements five successive levels for assessment, which are part of the steps related to Initiating Events and Screening (deterministic or probabilistic) analyses in the extended PSA flowchart. Fig. 3 shows levels 1 to 4 related to the multi-hazard assessment loop. Level 0 (not shown) corresponds to a single hazard assessment through standard practice or improved methods.

This methodology has been implemented in an open-source open-access software tool, the NARSIS Multi-Hazard Explorer [4], suitable to assess not only multiple hazards but also independent single hazards. It is very plant specific, and although the methodology can screen all hazard types and scenarios, there are still some combinations, which may be missed due to specific fragility loops, and/or dynamic hazard loops.

2.3 Fragility assessment in a MH context

A second main objective in NARSIS was to develop refined fragility derivation methods in order to increase the accuracy of the estimation of SSC failure rates against external

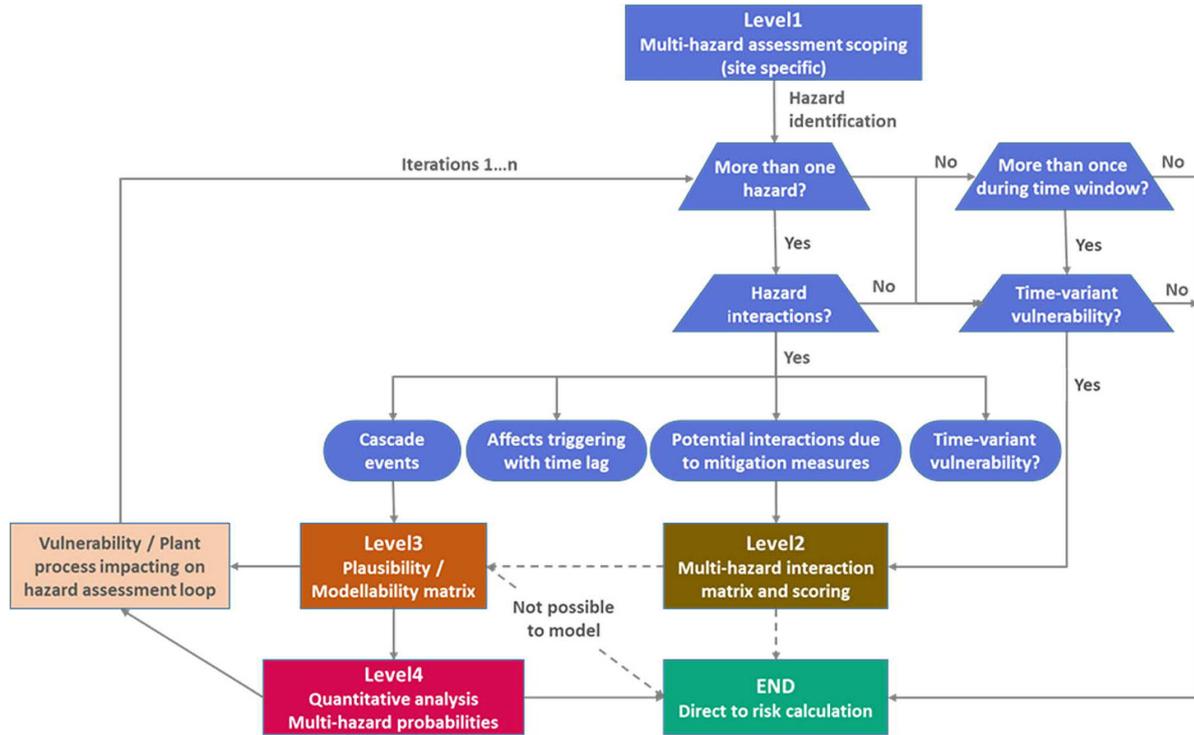


Fig. 3. The NARSIS MH framework scheme for scenarios to be used into multi-hazard PSA.

threats, thanks to current advances in quantitative hazard modelling and computational capacities.

Fragility or vulnerability curves are common in the nuclear industry as they are well suited for PSA applications, being at the interface between the probabilistic hazard assessment and event tree analyses, in order to estimate the occurrence rate of undesirable top events. They represent the probability of a given SSC to reach or exceed a predefined damage state as a function of an Intensity Measure (IM) representing the hazard loading. In the case of complex hazard loadings, a single scalar IM may not be sufficient to represent the severity of the aggression. As a result, conventional fragility curves using scalar IMs may come with a larger dispersion (i.e., uncertainty) in order to represent the imperfect relation between the IM and the loading actually applied. Such uncertainty then propagates through the PSA chain, potentially leading to unnecessary reliability margins.

In NARSIS, the concept of combining multiple IMs (also referred to as vector-valued IMs) in the formulation of fragility functions has been used, either for a single hazard event (essentially earthquakes), or for multiple hazard events (a volcanic eruption with deposit of tephra loads on a flood protection levee, followed by an earthquake), where each IM represents a loading level for a different hazard and the consideration of all IMs provides the means to quantify the probability of damage for multi-hazard scenarios. It has been shown that multiple IMs and physical failure modes can be combined in order to generate fragility models for a wide range of multi-hazard configurations. Provided that the required hazard-specific physical models are available, the following

statistical tools are able to cover most of the multi-hazard cases:

- multivariate generalized linear model regression or maximum likelihood estimation are to be used for the estimation of fragility parameters given a set of conditioning variables.
- Algorithms and procedures based on the system reliability theory (e.g., [5]) are able to combine hazard-specific failure modes in order to model the functionality states of a given SSC. Either joint probabilistic of failure or damage-state-dependent fragility functions may be derived from this framework.

Parts of the works have been first dedicated to screening and selecting the most critical SSC deserving in-depth fragility assessment. Contrary to the usual Safety Factors approach, this was done by applying a Risk-Informed methodology (NRC 2004) in which the safety significance is quantified by risk importance measures. Based on different case studies, the following SSC and safety functions have hence been identified as critical elements for PSA:

- I&C and switchgear cabinets/devices;
- fuel assembly spacer grids and, more generally, reactor pressure vessel internals;
- distributed systems (HVAC, piping, cable raceways).
- primary circuit depressurization;
- active isolation of the reactor containment building;
- passive reactor building resistance and leak-tightness in severe accident conditions (pressure and temperature);
- depressurization of the reactor building (by a filtered containment venting system);

- annulus venting system for NPP with double wall containment, auxiliary buildings filtration and venting;
- hydrogen risk management provisions.

Some works have also aimed at investigating the impact on the fragility assessment of SSC, of Soil-Structure Interactions (SSI) and of cumulative effects by succession of seismic events combined with ageing mechanisms and/or lifetime fatigue.

Regarding ageing, structural degradations due to the accelerated flow corrosion, creep and time and/or temperature material properties degradation have been considered for analysis. A methodology for performance prediction has been set and a deterministic approach has been adopted, based on several thermo-mechanical and seismic finite-element simulations performed on the NARSIS virtual NPP, used as reference for this assessment.

Regarding fatigue and earthquake combinations, a unifying framework for characterizing the probabilistic behaviour of a critical SSC (piping) has been proposed, which rely on a loading sequence made of a preliminary High-Cycle Fatigue (HCF) thermo-mechanical loading followed by some damaging seismic loadings. The aim was to derive the vector-valued fragility curves, as a function of both the duration of the nominal HCF stage and the chosen seismic IM.

Finally, the integration of human factors in the reliability analysis, as a potential source of epistemic uncertainty in the PSA, has also been explored.

2.4 The Multi-risk integration framework for safety analysis

A third key objective within NARSIS has been to improve the integration of external hazards and their consequences with existing state-of-the-art risk assessment methodologies in the industry.

Parts of the activities have been dedicated to investigating, further developing and evaluating the Bayesian Networks (BN) approach, hence delineating the advantages and challenges as compared to more conventional probabilistic safety assessment techniques (e.g., fault trees). Vector-based fragility was used in order to use multiple IMs for hazards and a novel BN-based method for human error probability was developed and connected to technical BNs. In complex (sub-)systems, BNs were shown to be able to be used as surrogate models for advanced numerical methods, in order to substantially reduce computational effort and allow their inclusion into larger systems. In addition, a new approach to the analysis of common cause failures was developed showing several advantages over existing methods in both calculation of the impact and visualization.

In addition, the Extended Best Estimate Plus Uncertainties (E-BEPU) methodology, which combines deterministic and probabilistic approaches for safety assessment, has been implemented and its behaviour evaluated regarding defence-in-depth and design extension conditions, as well as Severe Accident Management Guidelines. E-BEPU is able to introduce stricter require-

ments on possible event sequences and avoid possible cliff-edge effects. It allows relaxation for extremely unlikely sequences under certain conditions, when these sequences can be treated as “practically eliminated”. Its use has been demonstrated on the NARSIS reference plant model. However, it has required a huge computational effort.

Finally, some developments have been performed to constrain uncertainties when little data are available on failures. These developments have focused on the ability to identify the most influential sources of uncertainty and novel methods to prioritize and reduce them. In case of modelling of operator/human actions, the human failure probability for these actions can now be assessed and included in the study. Finally, a particular result is for the treatment of expert-based information using the tools of new uncertainty theories.

All these methodologies and developments can be used within a PSA. Each has advantages and disadvantages. Some methods (e.g., BNs) can be used as advanced versions of standard tools, whereas others can be used to investigate specific aspects and reduce uncertainties. Given the large variety of decision-making situations, finding a single appropriate framework appears to be debatable, and it is beneficial to take advantages of the strengths of multiple approaches to capture different types of information and knowledge important to inform decision-making.

2.5 Dissemination activities, potential impacts

Regarding education and training activities, apart from master trainings and postdocs proposed in the project, there have been 5 PhD theses covering a number of key research topics for NARSIS:

- extreme weather characterization,
- seismic fragility of ageing structures,
- vector-valued fragility functions for multi-hazard assessment,
- model reduction strategies for seismic response of structures,
- BN integration framework for probabilistic risk assessment.

Two international training workshops related to the Probabilistic Safety Assessment for Nuclear Facilities, have been held, one in Warsaw on September 2019 and the other in a fully digital format. A collaboration with the European Nuclear Education Network (ENEN), has enabled to invite 20 selected students and young researchers to participate in the first training workshop, where various lectures have been proposed in direct link with the project outcomes, as well as external invited talks on various topics.

At these occasions and all along the project duration, pedagogic materials (presentations, short videos, hands-on tutorials, notebooks) and lectures targeted towards students (e.g., masters) and young researchers or professionals have been produced.

Regarding dissemination activities, more than 20 journal papers have been published, as well as about 30 scientific conference papers (TINCE, NENE, SMIRT, FISA,

EGU, COMPDYN, ...). In addition, the project results have been presented systematically to the nuclear community during the annual SNETP/NUGENIA Forums.

Finally, apart from newsletters and the aforementioned publications and participations, the project had interactions with its International Advisory Board members, through dedicated meetings. This has led to very profiting discussions and feedback as these members are all part of international organizations (SNETP, IAEA, JRC, ...) with close links to nuclear safety issues.

3 The R2CA project

3.1 R2CA general overview and motivation

The R2CA project (Reduction of Radiological Consequences of design basis and extension Accidents, 2019–2023) is intended to harmonize the safety analysis methods through the development of generic methodologies for best estimate evaluations of the radiological consequences. The project addresses a broad scope of light water reactor designs from Gen II, III and III+ through the analyses of bounding scenarios of loss-of-coolant and steam generator tube rupture transients. It explores both design basis accidents (DBA) and design extension conditions without significant fuel melting (DEC-A).

The idea of launching such a project comes directly from (1) the consolidated evaluations of nuclear power plant severe accident progression and their associated radiological consequences and (2) the improvements of severe accident management strategies both issued from the numerous R&D programs launched after the Fukushima Daiichi Nuclear Power Plant (FDNP) accident. The integration, thereafter, of all these outcomes in level 2 Probabilistic Safety Assessments (PSA2) indeed demonstrated the effective reduction of the risks of all main categories of severe accidents and, in turn, highlighted the too conservative evaluations of design basis accidents. In addition, the importance to still strengthen the nuclear power plant safety level by considering accidental situations more severe than those currently taken into account for the plant design (the so-called design extension conditions) resulting from additional events or combination of different events and for which provisions have to be designed, have also been evidenced at that time [6,7].

With the primary objective of better estimating the radiological consequences of accidents for design basis and design extension conditions, the work undertaken in this project is then fully in-line with the European directives [8] recommending to continuously review the methodologies to establish the nuclear power plant safety margins for especially considering the changes that might have occurred in the operation conditions (e.g., increase in fuel burn-up) and the potential higher risks exhibited by knowledge improvements (e.g., clad secondary hydriding, increased fission product releases from local restructured high-burn-up fuel zone and from MOx fuels).

During the project, main efforts will be paid on the upgrading of currently used simulation tools and calcu-

lation chains in safety studies through the improvements of their models. The updated methodologies/calculation chains will help to derive some rationales for the optimization of Emergency Preparedness & Response plan in order to lower down the impact of the population protection measures. Additionally, innovative actions will also be performed where their main goals will be to provide some guidelines for the design and implementation of new accident management procedures or safety devices (incl. dedicated instrumentation) or to develop innovative approaches based on artificial intelligence capabilities for anticipated accidental situation diagnosis. Finally, the project will also take benefits of the upgrading of simulation tools and calculation methodologies to quantify the pros and cons of some concepts of Accident Tolerant Fuels promoted worldwide.

3.2 R2CA overall approach

To address the project objectives, a step-by-step methodology (Fig. 4) was implemented including the following key milestones:

1. review of the existing methodologies for radiological consequence evaluations of loss-of-coolant and steam generator tube rupture scenarios, of available experiments and/or reactor measurements relevant for design basis and design extension conditions and of the simulation tools that will be used within the project;
2. identification of reactor accidental cases of interest covering all aspects (both conditions and scenarios, different light water reactor designs) and use of the available calculation chains and methodologies for the simulation of the selected scenarios;
3. developments and/or improvements of the calculation schemes used for the simulations of loss-of-coolant and steam generator tube rupture accident phenomena, when needed, and verification/validation of upgraded models against consolidated experimental databases. Model developments and/or improvements are expected from fuel behaviour up to the fission product releases to environment;
4. quantification of the obtained gains in terms of radiological consequence reduction for the selected scenarios with the improved simulation schemes and elaboration of guidelines for the development of harmonized evaluation methodologies;
5. demonstration of the safety gains that could be achieved from innovative accident management procedures, new safety devices (i.e., dedicated instrumentation and some Accident Tolerant Fuel concepts) and early diagnosis tools.

Both integral calculation chains (dealing globally with all the processes from the initiating events up to the environmental releases) and detailed/mechanistic simulation tools (addressing part on the phenomenology only) are used and will be upgraded within the project. These latter will support the re-assessment of the experimental database, will be used to inform low fidelity model of more integral simulation tools or, in some cases, to perform numerical experiments to investigate badly understood uncertainties.

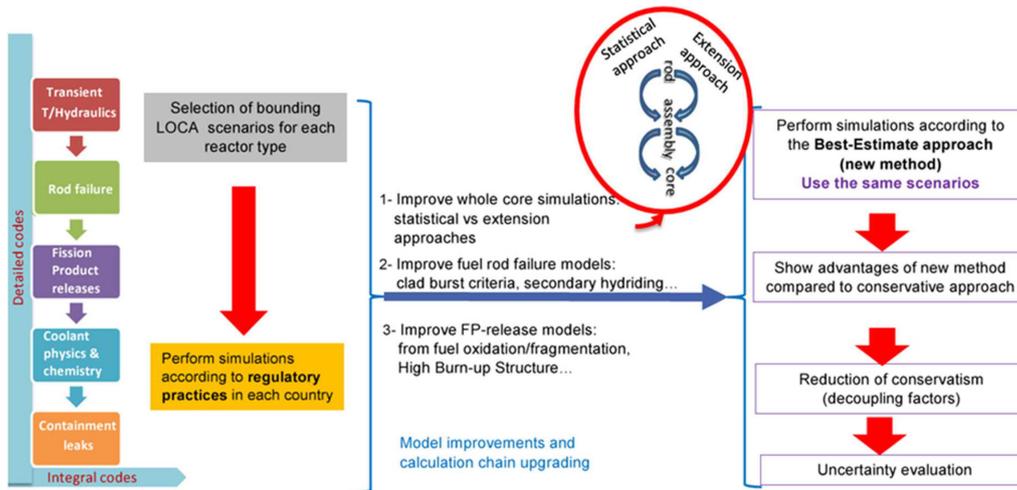


Fig. 4. R2CA methodology for best-estimate evaluation of releases during a LOCA transient.

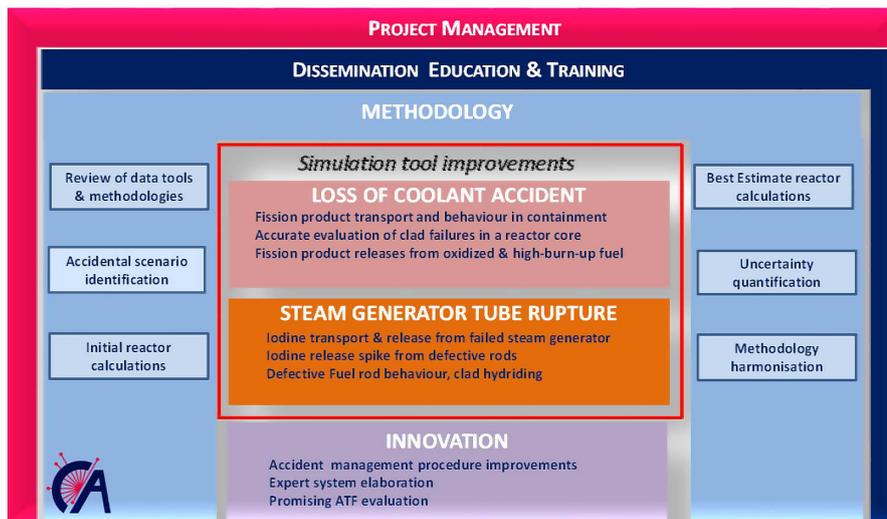


Fig. 5. R2CA overall structure.

The determination of radiological consequences to individual or group of populations will be evaluated from the fission product releases from the facility to the environment. Comparison and improvements of evaluation methodologies will be restricted to the source term from the facility whereas a simplified and unique tool for the source term dispersion in the biosphere and the associated doses to the population will be established and used by all the participants to quantify the obtained gains.

The achievement of the overall objectives is assured by a consistent and coherent work program, reflected in the four technical Work Packages (WP) defined as follows (Fig. 5) which are dedicated to:

- the reviews (methodologies, simulation tools, database) and performance of initial and upgraded reactor case simulations (WP2);
- the improvement of phenomenon modelling for loss-of-coolant (WP3) and steam generator tube rupture (WP4) transients;

- the development of new accident management actions (incl. proposals for new devices e.g., passive systems...) as well as of an accidental diagnosis tool prototype for steam generator tube rupture anticipation and the evaluation of the resistance to loss-of-coolant transients of some Accident Tolerant Fuel concepts (WP5).

The diversity of the 17 organizations included in the R2CA consortium, from industry (designers, utilities) to academic (universities, R&D centres) and including TSOs, favours the foreseen R&D work, the emergence of innovative ideas and the development of theoretical model whereas the demonstration of their effectiveness in lowering the radiological consequences will be tested on selected nuclear power plant case studies within the project. In addition, the consortium composed of 11 countries participating in the project, equally balanced between western and eastern Europe with different regulatory frameworks, offers the opportunity to cooperate on a wide variety of reactor designs from Gen II, III and III+

(BWR, PWR, VVER, EPR) and to share different safety approaches.

3.3 Main advances

Several milestones have already been reached by the project whose main ones can be summarized as follows:

1. the existing knowledge reviews have been completed. In particular, a specific database selecting past experimental data and power reactor measurements of interest for the project has been built [11]. Meanwhile, the methodology review has highlighted major differences leading for instance for loss-of-coolant transients to 1 to 2 orders of magnitude difference for some key isotopes (I and Cs). Finally, the review of simulation tools (more than 20) has helped not only characterizing their modelling capabilities but also pinpointing their required development needs and where the modelling effort should be focused on;
2. first set of reactor calculations have been performed (more than 40). A common template has been set-up for collecting the results of the reactor case simulations that will ease the comparison between initial and best-estimate calculations and the construction of the final calculation result database.

In parallel, a simple radiological tool was provided for evaluating the radiological consequences in a very generic way for the assessment or re-assessment of the safety margins. It basically consists in: (i) a Bi-Gaussian Plume dispersion model function with Briggs-equation for modelling the ambient air behaviour and (ii) the calculations of the effective doses (by inhalation & external exposure) and the equivalent thyroid dose based on formula originated from ICRP guides (e.g., ICRP 144 [9] & ICRP 71 [10]).

Finally, model revisions/developments for fission product transport/behaviour and for fuel/clad behaviour, as well as the coupling of simulation tools (e.g., fuel performance codes with fission product release ones) are ongoing and pave the path towards upgraded calculation chains that will be finalized very soon.

3.4 Key results expected and impacts

The derived guidelines to harmonize the methodologies for safety analysis of the radiological consequences for design basis and design extension condition accidents should be applicable to all existing European reactor designs (BWR, EPR, PWRs, VVERs) and foreseen concepts (incl. Small Module Reactors).

In addition, thanks to the knowledge, data provided to all participants during the project, and to the sharing of simulation tool improvements, it is foreseen to increase the competence of the contributing organizations in their evaluation of radiological consequences for loss-of-coolant and steam generator tube rupture transients. While being an opportunity for some of them to improve their safety studies, it is also expected that the upgraded simulation

tools and calculation chains will be useful beyond the consortium to both Industry (utilities, vendors), National Authorities and their TSOs.

Finally, it is expected that by fostering the cooperation between a large diversity of participants and different countries in Europe and bringing together experts from fuel safety, source term and accident consequences, the nuclear power plant safety at European level will be re-enforced.

3.5 Dissemination and training activities

Dissemination of the project results is oriented towards the widest community as possible through several different media (publications in peer-reviewed journals, presentations in international conferences, periodic newsletters, public project deliverables, workshops or side-events open to a large audience...). A public website is also available (<https://r2ca-H2020.eu>). Since the beginning of the project, 15 papers have already been produced for journals, general and specialized conferences (ERMSAR, NENE, TOPFUEL, NURETH). A special edition of Annals of Nuclear Energy is also under preparation.

The dissemination is expected to be particularly efficient within the nuclear community and Europe, due to, respectively, the variety of participants spreading the information through their own networks (e.g., industrial partners through utility groups, TSO's through ETSON...) and the different countries involved.

The final dissemination of the project results is planned to be done through:

- an End-Users Group (with researchers not participating to R2CA and external stakeholders);
- international organizations (e.g., OECD or IAEA) with the sharing of a database collecting the best estimate reactor calculation results or the edition of dedicated documents (i.e., as State Of the Art Report or as part of "Safety Guides").

Training and education will be both part of the dissemination and exploitation of the results. An important objective of R2CA project is to contribute to the European effort on nuclear education and training activities by integrating the main outcomes of the project into the program of dedicated side-events to the main international workshops or of ad-hoc workshops. The project also favours the involvement of students (masters, PhDs) and postdoc fellows. Four students and one postdoc have already been involved covering the following research topics:

- fuel behaviour (re-structuring) and associated fission product releases during loss-of-coolant transients;
- defective fuel rod behaviour and accident management optimization during steam generator tube rupture transients;
- smart tools for early diagnosis of accidents.

Mobility of students and young researchers between different partner's organizations will be also funded to encourage the transfer of knowledge and expertise. Additionally, specific training sessions on computational tools

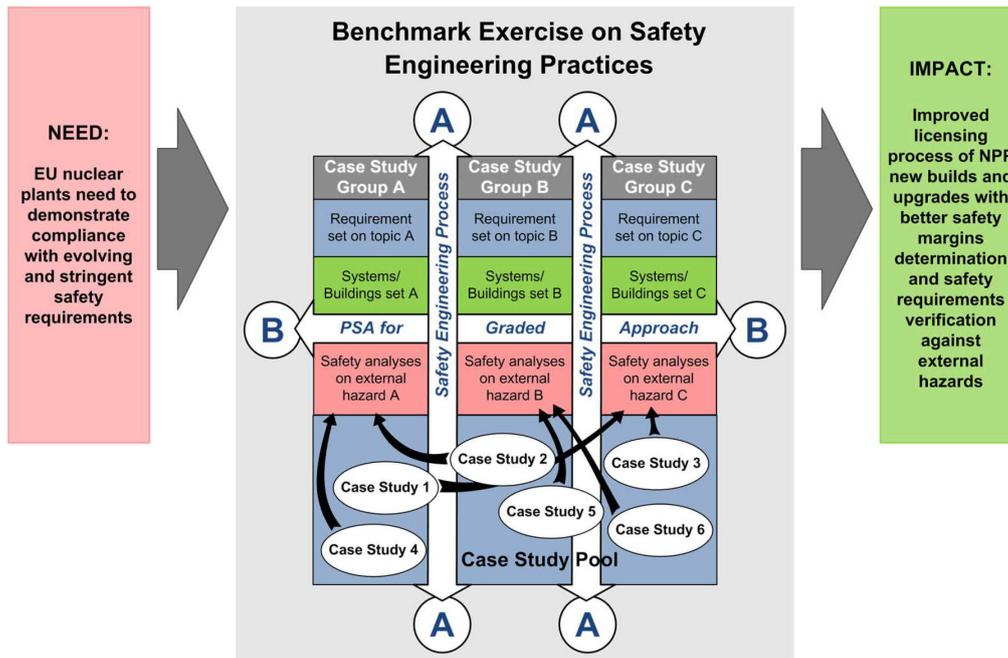


Fig. 6. The overall concept of BESEP project.

are planned to disseminate their use and their associated best-practices.

4 The BESEP project

4.1 BESEP overview

The objective of the BESEP project (Benchmark Exercise on Safety Engineering Practices, 2020–2024) is to support safety margin determination by developing best practices for safety requirement verification against external hazards, using efficient and integrated set of Safety Engineering practices and probabilistic safety assessment.

The overall concept of BESEP is illustrated in Fig. 6. The project is carried out as benchmark exercise (marked with grey background in the figure) between several EU member countries participating in the project. The benchmark exercise is based on case studies previously performed by the participants. The case studies will be further refined during the project to support the benchmarking.

The benchmark exercise focuses on the comparison of failure analyses performed in the case studies and on the inter-connections and inter-actions of different analysis methods involved in the safety assessment of different external hazards. The integration of safety analysis methods is typically handled in a Safety Engineering process.

For the selected case studies, a cross-case comparison for the case studies belonging to the same group and a cross-group comparison between generalized case studies representing each group are performed. The evaluation results of cross-case comparison focus on the safety margin determination and safety requirement verification (shown with letter A in Fig. 6). The evaluation results of cross-

group comparison focus on the identification of benefits for increasing the level of detail in the applied safety analysis methods, e.g., the benefits of applying more detailed models or additional simulations. This helps in balancing the plant safety against different external hazards (shown with letter B in Fig. 6). Together, the results of both comparisons can be used to estimate the resilience of safety margins in case of design-basis exceeding external hazards.

As a result, BESEP will answer to the need for EU nuclear power plants to demonstrate compliance with evolving and stringent safety requirements. The impact of BESEP is the improved licensing process of nuclear power plant new builds and upgrades with better safety margin determination and safety requirement verification against external hazards.

4.2 Safety design and Safety Engineering process

In the licensing process of a nuclear power plant, the safety authority will review and assess the design basis of the plant, the requirement specifications, the analyses substantiating the fulfilment of safety criteria, the implementation of defence-in-depth concept in the design as well as the implementation of redundancy, physical separation, functional isolation and diversity principles in the design and implementation of safety functions. The licensing process is endorsed by a Safety Engineering process connecting together the main elements of safety design: safety requirements, safety analyses and plant design.

In a steady-state situation, the three main elements are in balance, and there is general consensus that, based on the safety analyses, the current plant design fulfils the given safety requirements. However, in case there is a change in one of the elements the change should be

reflected in the two other elements. This is usually for the Safety Engineering process to take care of.

During the lifecycle of a plant, there can be various changes to the elements, for example:

- new design concepts and feasibility studies may give new ideas to refresh the plant design;
- international and national safety agencies may introduce new safety goals leading to changes in the safety requirements; or
- operational experience from internal and external hazards may challenge the existing safety analyses giving initiative for more stringent safety margins.

The main elements of safety design and the potential reasons to the changes in the main elements are illustrated in Fig. 7.

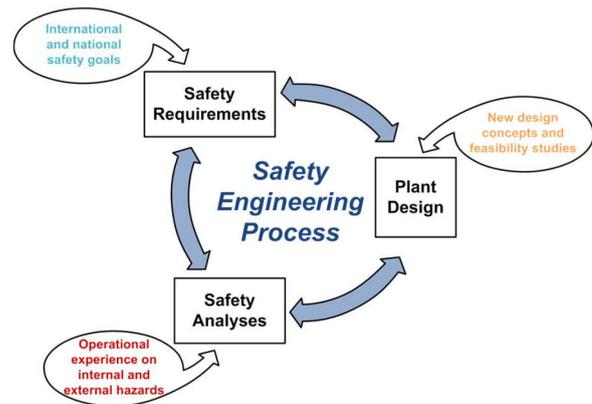


Fig. 7. Main elements of safety design and potential reasons to the changes.

4.3 Requirement baseline for BESEP

The BESEP partner countries have different nuclear safety requirements which lead to different safety engineering practices. Although, there are differences in practices, the goal is the same: Showing the fulfilment of the safety requirement in the nuclear power plant design and operation.

A requirement baseline for the benchmark exercise is created in Work Package 2. The requirement baseline is later used in Work Package 3 for cross-case comparison within case study groups and Work Package 4 for comparison between generalized case studies representing the different case study groups.

The following safety analyses and safety engineering practices are needed to ensure compliance with safety requirements for the plant:

1. deterministic safety analyses (DSA) – analyses of initiating events induced by external hazards, evaluating of plant response, plant performance or success criteria;
2. probabilistic safety analyses (PSA) – modelling of accident sequences, quantification of their risk significance;
3. human factors engineering (HFE) – scope of testing and maintenance, operator and emergency response actions on the basis of pre- and post-hazard procedures, SB EOPs and SAMGs;
4. safety engineering practices (SEP) – implementation of safety requirements to exiting plant design for fulfilling the Defence-in-Depth principle.

Based on the preliminary case studies and general experience of the BESEP partners the safety requirement topics have been defined for the above-mentioned safety analyses and safety engineering practices to be applied in the project. As an example, the topics and short descriptions on their focus in the category of safety engineering practices are listed below. The presented list is not trying to be a comprehensive representation of safety engineering practice topics. The purpose is to identify safety engineering practice topics of interest supporting the benchmark and the objectives of BESEP project.

1. safety engineering management, this topic concerns the processes and models regarding the general structured management of safety engineering activities of NPP license holders;
2. safety design and requirement management for external hazards, this topic concerns managing the balance between the plant safety design and the allocated safety requirements;
3. flow of information between safety analyses, this topic concern interactions and interconnections between the three analysis areas (DSA, PSA, HFE);
4. verification and validation (V&V) of design, this topic concerns interaction between the three main elements of safety engineering: safety requirements, plant design, and safety analyses;
5. system modification and configuration management, this topic concerns system modification configuration management;
6. validated modelling and simulation analysis tools, this topic concerns the validation and improvement of models and the tools used for the analysis of effects of external hazards.

For all these topics specific BESEP requirements were defined to support the upcoming benchmarking. The BESEP requirements were elaborated from the high-level requirements of IAEA and national requirements identified and selected by the BESEP partners. As an example, the BESEP requirements on the flow of information between safety analyses topic are shown in Table 1. The collection of BESEP requirements for all topics on safety analyses and safety engineering practices create the requirement baseline for the benchmark exercise [12].

4.4 Key results expected from BESEP

The expected key results from the benchmark exercise are:

1. best practices for the verification of evolving and stringent safety requirements against external hazards;
2. guidance on the closer connection of deterministic and probabilistic safety analysis and human factors

5 General conclusion

The focuses and interaction areas of NARSIS, R2CA and BESEP projects are illustrated in Fig. 8. As can be seen from the figure, the three projects cover a wide range of assessment methodologies in nuclear safety. The projects prove that the European research and development framework is the convenient environment for the improvement of safety assessment methodologies. The stringent safety requirements call for proven and justified safety assessment methodologies to be applied in the European nuclear industry. The European research and development programs bring together the different sides of nuclear industry, i.e. utilities, vendors, national safety authorities and technical support organisations, and benefits from their know-how and expertise.

The achieved and expected key results from the projects help improve the best practices for the safety assessment of internal and external events and for the planning of mitigation strategies. The results support the harmonization of safety assessment methodologies between European countries applicable to different existing NPP designs and foreseen concepts, such as Small Module Reactors. The projects help tighten cooperation between participants from different countries in Europe and bring together experts from the different areas of safety assessment, such as Deterministic safety analysis, Probabilistic safety analysis and Human factors engineering. Last, but not least, the projects foster new experts for the industry, who eventually take the responsibility of continuous development in nuclear safety.

Conflict of interests

The authors declare that they have no competing interests to report.

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Data availability statement

There are no special data nor any repository apart from what is provided on the projects or EU website.

Author contribution statement

This paper is an attempt to summarize the main elements regarding the three targeted projects, each

author having contributed on behalf of his/her project consortium.

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