Spanish Nuclear Safety Research under International Frameworks

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The Nuclear Safety research requires a wide international collaboration of several involved groups. In this sense this paper pretends to show several examples of the Nuclear Safety research under international frameworks that is being performed in different Universities and Research Institutions like CIEMAT, Universitat Politècnica de Catalunya (UPC), Universidad Politécnica de Madrid (UPM) and Universitat Politècnica de València (UPV).

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CIEMAT

The international nature of the investigation conducted by the Unit of Nuclear Safety Research (UNSR) of CIEMAT, responds to the belief that international cooperation is the only efficient way for addressing a so complex area. Namely, international involvement is seen by CIEMAT as an indispensable instrument to meet its generic objectives:

- To develop, validate and assimilate methodologies of risk assessment.
- To reduce existing uncertainties in postulated accidents.

However this heavy international component, UNSR activities build up a bridge between worldwide forefront research and its results application in a domestic environment. This is the major driver for CIEMAT's collaboration with the Spanish nuclear industry and institutions.

Three are the areas of investigation CIEMAT is presently addressing: severe accidents, nuclear fuel thermo-mechanics and safety of innovative nuclear systems.

RECENT UNSR RESULTS UNDER INTERNATIONAL PROJECTS

During the last 5 years CIEMAT has been participating in a good number of international projects. Besides, the just born projects in which UNSR is participating will be introduced.

Severe Accidents

The main projects shaping CIEMAT's research in the area of severe accidents have been: PHEBUS-FP, SARNET, OECD-SFP, OECD-BIP, OECD-CCVM and ARTIST. The first two were launched under the EURATOM framework program, whereas the next three have been framed under the OECD-NEA; the last one was an international project led by PSI.

The PHEBUS-FP (Clément & Zeyen, 2013) used a scaled down facility (1:5000 with respect to a 900 MW French PWR) to conduct a total of 5 experiments (4 of them with spent fuel). Outstanding insights have been gained into fuel degradation, fission products transport and iodine chemistry. CIEMAT has been contributing in the pre- and post-test analysis of containment phenomena with codes like CONTAIN 2.0, IODE 5.1, MELCOR 1.8.6 and ASTEC 2.0. Based on this work an extensive validation has been carried out and major observations, like the dominant contribution of sedimentation in aerosol deposition or the effective trapping of iodine by oxidized silver in the aqueous pond of containment, have been consistently estimated. It is worth mentioning the comparison set between the major findings concerning containment iodine behavior within the PHEBUS-FP tests and the NUREG-1465 (Herranz & Clément, 2010b). The experience gained within the project was the basis for an assessment of how significant iodine chemistry might be in NPP source term estimates (Herranz et al., 2009). Complementary to PHEBUS-FP experience, the International Source Term Project was launched to explain and deepen into some of the PHEBUS observations.

Since 2004 most severe accident investigation has been articulated through the SARNET project (Van-Dorsselaere et al., 2012). The major issues addressed within this network of excellence were: corium coolability, core-concrete interaction, steam explosions and hydrogen combustion in the containment and source term. Most of data obtained through experiments have been stored in a database called DATANET and the models developed and validated are being implemented in the next version of the ASTEC code (2.0 r3). CIEMAT has been involved in several activities with particular emphasis in those related to source term. Significant contributions have been already published as for aerosol evolution (Herranz et al., 2010a) and iodine chemistry (Dickinson et al., 2010). As a consequence of these SARNET analytical activities, CIEMAT has conducted an extensive validation of the ASTEC and the MELCOR codes (Kljenak et al., 2010; Girault et al., 2012).

The OECD-NEA projects have addressed different issues: SFP, fuel degradation under anticipated conditions during a complete LOCA (Loss Of Coolant Accident) in spent fuel pools; BIP, organic iodides generation and iodine-paints interactions; CCVM, validation matrix for containment codes. Within the OECD-SFP project, the UNSR built-up MELCOR models for BWR and PWR fuels for two fuel configurations, "hot neighbor" (i.e., no radial thermal radiation from the fuel cell) and "cold neighbor" (i.e., periphery fuel assemblies heated up by the central one). This work allowed identifying strengths and weaknesses of the code when addressing these scenarios (Herranz et al., 2013). Key elements of the modeling were the hydraulics of air going upward through the fuel assembly, the zirconium oxidation by air and the thermal radiation heat transfer. Under the OECD-BIP framework, a sound database has been obtained and CIEMAT has proposed an empirical approach accounting for the different kinetics observed of the organic iodides production (Herranz et al., 2010). The OECD-CCVM project allowed gathering relevant experimental work that might be the bases for validation of containment codes and/or modules of integral codes; specific contributions were made from CIEMAT as for the identification and definition of key in-containment phenomena and the description of relevant experimental data in areas such as aerosol removal in leakage paths and aerosol resuspension and reentrainment.

The ARTIST projects have investigated the potential retention of fission products within the secondary side of the steam generator during a meltdown SGTR sequence (Güntay et al., 2004). CIEMAT's activities have been experimental and analytical. Experimentally, CIEMAT has complemented the project database by measuring particle retention in the break stage of the steam generator. More than 30 tests have been conducted in the LASS (Laboratory for Analysis of Safety Systems) of CIEMAT, in which the effects of particle nature, breach type and size and gas mass flow rate have been explored. Consistently with other partners' findings, major insights have been gained: even in the worst scenario, some mass retention should be expected in the break stage; particle nature largely determines mass trapping; largely agglomerated particles may undergo fragmentation due to tangential fluid stresses and collisions against tubes; etc. These and many other results have been reported by Herranz et al. (2006), Sánchez et al. (2010) and Delgado et al. (2013). On the analytical side, the main achievement has been the development and validation of a semi-empirical model, called ARI3SG, capable of predicting in-break stage aerosol deposition (Herranz et al., 2007; Herranz et al., 2012; López et al., 2012).

In addition to the above projects, CIEMAT is participating in some other international projects recently launched under the frame of the 7th FWP of EURATOM. This is the case of PASSAM and CESAM. Even though research on severe accident management is one of the main pillars of both projects, PASSAM is mostly empirical while CESAM is entirely analytical. PASSAM main goal is to set up a data base on the performance of existing and innovative systems for source term mitigation. Fission products retention in aqueous ponds and in sand filters will be investigated under realistic challenging conditions still unexplored or poorly characterized, like water saturation or jet injection regime. Also, performance of innovative systems as pre-filter acoustic agglomerators, electrostatic precipitators, high pressure sprays and improved performance zeolites are to be tested under prevailing conditions in case of containment venting. The intention is to gather a sound database useful to enhance the potential performance of filtered containment venting systems. CIEMAT has focused their activities on aqueous ponds retention under jet injection regime and on the particle growth resulting from an acoustic agglomerator.

CESAM is entirely focused on development of the ASTEC code models and extension of its analytical capabilities to different reactor designs. CIEMAT is contributing through preparation of a generic input deck for a BWR plant, modeling the BWR fuel degradation tests conducted by Sandia simulating a complete LOCA accident in a spent fuel pool and assessing and improving models dealing with pool scrubbing, air oxidation of Zr alloys and thermal radiation in spent fuel pool configurations.

Finally, less than one year ago, CIEMAT joined the OECD-BSAF project (Benchmark Study of Accident at the Fukushima Daiichi Nuclear Power Station), through the bilateral project with CSN for severe accidents. The project aim is to reach a deep understanding of the scenarios, with particular emphasis on identifying the governing phenomena and the final status of the units 1-3. CIEMAT is building up MELCOR 2.1 models for each of the units. The results of each participant will be discussed and the conclusions are planned to be used for the units decommissioning.

Thermo-mechanics of nuclear fuel

CIEMAT has two major national collaboration agreements on nuclear fuel thermo-mechanics, one with CSN and the other with ENRESA. Some of their activities have a domestic scope, but there are others of an international nature related to projects like OECD-HALDEN, OECD-CABRI and OECD-SCIP (focused on fuel behavior under steady and transient conditions); additionally, a close follow-up is being conducted of the ESCP (focused on fuel extended storage) program. Access to the data resulting from these frameworks comes from the CSN involvement in the international projects.

Data from HALDEN experiments have been used for many years to improve and validate codes like FRAPCON-3. Since 2012 CIEMAT has launched an activity to assess the FRAPTRAN-1.4 predictability of the fuel thermo-mechanics under LOCA conditions. Through the analyses of the LOCA experiments, major code drawbacks and needs will be identified and, potential improvements to be made in the short and medium term will be defined.

CIEMAT's experience on fuel transient behavior under RIA (Reactivity Insertion Accident) conditions has been gained within the CABRI project. Post-test analysis and preliminary analysis of the upcoming LWR-RIA tests has fostered technical exchange among partners. The best example is the RIA benchmark organized under the frame of the international project, which has been articulated in 9 RIA scenarios based on tests already conducted or to be conducted of the CABRI and the Japanese NSRR program. CIEMAT contributed to the benchmark by analyzing those scenarios with FRAPTRAN-1.4 and SCANAIR-7.1 (Sagrado et al., 2013). As a result of this work an extensive and deep comparison of thermal, mechanical and fission product transport models responses has been made.

SCIP has been supplying data for years on irradiated BWR and PWR mechanical behavior. Through them a better understanding of the potential cladding failures by pellet-cladding interaction and hydrogen-related phenomena has been gained. Additionally, a sound database on clad ramping has been set up and two benchmarks have been recently organized. In addition to technical contributions with FRAPCON-3 and FRAPTRAN codes, CIEMAT has been deeply involved in the coordination of both exercises. Some analyses have been conducted with FRAPCON-3.3 and some others with FRAPTRAN-1.4; this way the capability of these two codes to capture phenomena in the time range of ramps has been checked (ramp timing is right in between steady state and RIA, which are the natural domains of FRAPCON-3 and FRAPTRAN -1, respectively). Herranz et al. (2011) reported in detail results and major outcomes of the first exercise.

Safety of innovative nuclear systems

Since the early 90's, CIEMAT has been working on innovative aspects of safety systems of new reactor designs. The studies evolved from modeling of passive systems performance of Generation III reactors to accident analysis of Generation IV reactors. From 2005 to 2009, CIEMAT focused on High Temperature Reactors (HTRs) through 6th EURATOM projects, like RAPHAEL, and through bilateral contracts with PBMR (Fontanet et al., 2009). However, last years the interest has moved to safety of Sodium Fast Reactors (SFRs). In that environment, CIEMAT has been working on source term studies under the frame of the CP-ESFR project of the 7th EURATOM FWP. The project addressed key design aspects of an SFR, from the configuration (pool vs. loop) to the most suitable power cycle to be coupled (Rankine vs. Brayton), going through safety aspects. In this specific regard, a critical assessment of what was known and unknown and an identification of the major needs in terms of model development were the first task undertaken by Herranz et al. (2012b).

Presently, CIEMAT is heavily involved in developing and validating models considered fundamental for source term prediction. That is the case of sodium vapor nucleation under anticipated conditions of SFR containments in case of a BDBA (Beyond Design Basis Accident). This work together with the development of a semi-empirical law for SFR cladding creep and the assessment of a modified version of RELAP-5 against sodium thermal-hydraulic tests (conducted in collaboration with UPV), are the major tasks of CIEMAT within the JASMIN project of the 7th EURATOM FWP. The final aim of the project as a whole is to enable the ASTEC platform to also address SFR accident scenarios.

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GROUP OF THERMALHYDRAULIC STUDIES (GET) OF TECHNICAL UNIVERSITY OF CATALONIA (UPC)

The group of thermal-hydraulic studies (GET) of the Technical University of Catalonia (UPC) is currently participating in different international projects in the field of dynamic analysis of nuclear systems behaviour under accidental scenarios. Such initiatives could be classified in two groups: those developed in the framework of a common effort of different Spanish universities and those build up by the group itself with other sponsorship. In the former group one can find projects like CAMP (under the leadership of the NRC), OECD-ROSA project dealing with experiments performed at the LSTF facility (JAEA-Japan) and OECD-PKL2 project dealing with experiments performed at the PKL facility (Areva-Germany). In the latter group one can find three projects: BEMUSE, UAM and PREMIUM. Both groups of projects are strongly connected with a general strategy established and sponsored by the Spanish Nuclear Safety Council (CSN). Since the former group is well known by the Spanish nuclear community, the purpose of this text is to describe the participation of UPC-GET in the latter group of projects. UPC-GET members are the authors of the work done in these projects. The names of the researchers are the following: E. de Alfonso, C. Arenas, L. Batet, J. Freixa, E. Mas de les Valls, V. Martínez, M. Pérez, R. Pericas, C. Pretel and F. Reventós.

BEMUSE is the first of these projects and it has been already completed. BEMUSE stands for **B**est **E**stimate **M**ethods for **U**ncertainty and **S**ensitivity **E**valuation. It has been a programme promoted by the working Group on Accident Management and Analysis (GAMA) and endorsed by the Committee on the Safety of Nuclear Installations (CSNI). BEMUSE represents an important step towards a reliable application of high-quality best-estimate and uncertainty and sensitivity evaluation methods. The application of these methods to a Large-Break Loss of Coolant Accident (LB-LOCA) constitutes the main activity of the programme, structured into two main stages:

- Step 1: Best-estimate and uncertainty and sensitivity evaluations of the LOFT L2-5 test. This step includes Phases II and III (A. De Crécy et al. 2008). LOFT is the only integral test facility with a nuclear core where thermal-hydraulic safety experiments have been performed.
- Step 2: Best-estimate and uncertainty and sensitivity evaluations of a nuclear power plant. This steps includes Phases IV (M.Pérez et al. 2010) and V (M.Pérez et al. 2011).

A presentation of the uncertainty methodologies to be used by the participants (Phase I) was included in the first step. The final phase (Phase VI) consisted of the synthesis conclusions and recommendations.

UPC-GET participated in all the phases of the activity being also the coordinator of phases IV and V (devoted to the exercise related to the nuclear power plant). The team took advantage of the project and managed to come to a common understanding with all the participants on the general aspects involved in the different steps of the calculating exercises. In parallel with its participation in BEMUSE, UPC managed to develop the so-called UPC-CSN methodology of Best Estimate Plus Uncertainty (BEPU) evaluation, which is a statistical method based on Wilks' theory with some specific features related to the validation of the basic plant nodalization and also to the reduction of the number of input uncertain parameters. Figure 1 shows UPC's prediction of peak cladding temperature (PCT) for large break LOCA comparative exercise along with its upper and lower bands.

UAM is the second project. UAM stands for **U**ncertainty in **A**nalysis **M**odelling. The objective of this project is to conduct an OECD benchmark for uncertainty analysis in best-estimate coupled code calculations for design, operation, and safety analysis of LWRs. The proposed technical approach is to establish a benchmark for uncertainty analysis in best-estimate modelling and coupled multi-physics and multi-scale LWR analysis, using as bases a series of well defined problems with complete sets of input specifications and reference experimental data. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) are tested on a number of benchmark exercises for which experimental data are available and for which the power plant details have been released.

Several steps or exercises, each of which can contribute to the total uncertainty of the final coupled system calculation, are prepared by coordinators. Participants have to identify relevant aspects of each step and propagate the uncertainties in an integral system simulation for which high quality plant experimental data exist.

The main scope covers uncertainty analysis in best estimate modelling for design and operation of LWRs, including methods that are used for safety evaluations.

UPC-GET is participating in selected phases of this project (C. Arenas et al. 2013). The team is gaining experience in the selected steps of the calculating exercises, bearing in mind its own specific goal of establishing a neutron-kinetics-thermal-hydraulic coupled model for a Spanish Nuclear Power Plant having the capability of being used for Best Estimate Plus Uncertainty (BEPU) evaluation in the licensing context.

The third project is **PREMIUM**, Post BEMUSE REflood Models Input Uncertainty Methods, a project endor-

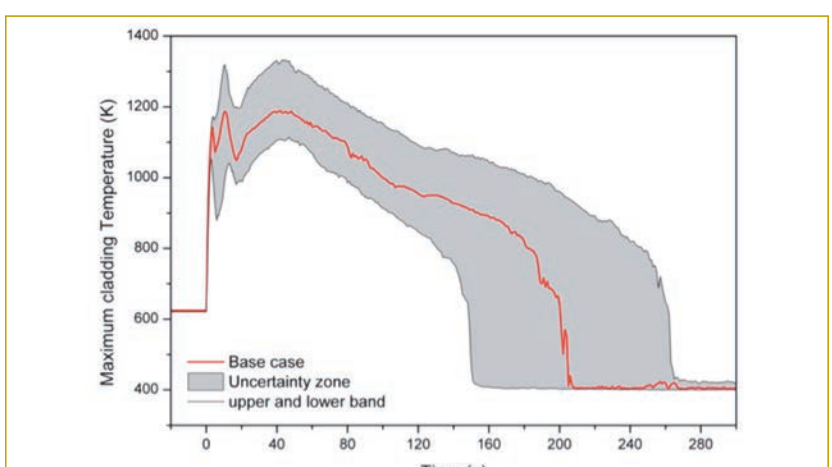

Figure 1. Base case and uncertainty upper and lower bands results obtained at the UPC for a large break LOCA in the BEMUSE project.

sed by the OECD/NEA/CSNI/WGAMA group. The objective of PREMIUM is to progress on the issue of the quantification of the uncertainty of the physical models in system thermal-hydraulic codes, by considering a particular case: the physical models involved in the prediction of core reflooding. The final goals of the project are: the assessment of advanced methods and tools used for event/accident analysis and the review of current analytical tools as well as risk assessment approaches regarding their applicability to safety assessments of new designs, and their further development and validation.

PREMIUM is structured in 5 phases to deal with the quantification of the uncertainties for the influential physical models in the reflooding. The participants will:

- Determine the uncertain parameters of their code associated with these physical models (Phase 1 and 2).
- Quantify the uncertainties of these parameters using FEBA/SEFLEX experimental result or own reflood experiment (Phases 3)
- Confirm the found uncertainties by uncertainty propagation in the case of the 2-D reflood PERICLES experiment. This part will be performed as a blind analysis. (Phases 4 and 5)

UPC-GET intends to participate in all the phases of the activity being also the coordinator of Phase 1. The team will try to take advantage of the project to develop a methodology to quantify the uncertainties for different physical models. Figure 2 shows UPC's prediction of Quench Front in FEBA experiment compared with the associated experimental data.

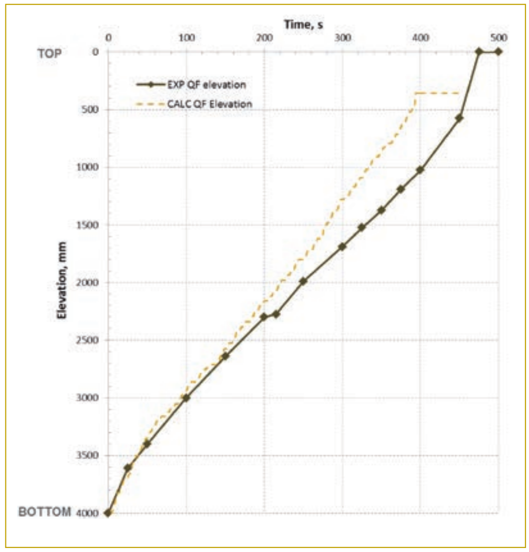

Figure 2. Prediction of the quench front propagation in the FEBA experiment of the UPC group (PREMIUM project).

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UNIVERSIDAD POLITÉCNICA DE MADRID (UPM)

The UPM participates in several international projects related with Nuclear Safety which can be split in two groups:

- those which have been funded by EURATOM, involving four projects in nuclear safety (NURESAFE, SARGEN-IV, ESFR, ESNII+) and in three projects about education in nuclear safety (ENEN-III, TRASNUSAFE, NUSHARE), and
- those corresponding to other international projects which have been funded in Spain by CSN and UNESA: CAMP, NEA/OECD PKL, NEA/OECD ROSA, NEA/OECD ROSA-2, NEA/OECD SM2A, IDPSA network.

The following description is mainly focused on the first group because there are other projects like CAMP, NEA/OECD PKL or NEA/OECD ROSA which are well-known or are described with more detail by other groups inside this paper.

NURESAFE

The NURESAFE project addresses safety of light water reactors which will represent the major part of fleets in the world along the whole 21st century. The first objective of NURESAFE is to deliver to European stakeholders a reliable software capacity usable for safety analysis needs and to develop a high level of expertise in the proper use of the most recent simulation tools. Nuclear reactor simulation tools are of course already widely used for this purpose but more accurate and predictive software including uncertainty assessment must allow quantifying the margins toward feared phenomena occurring during an accident and they must be able to model innovative and more complex design features.

This software capacity will be based on the NURESIM simulation platform created during FP6 NURESIM project and developed during FP7 NURISP project which achieved its goal by making available an integrated set of software at the state of the art. The objectives under the work-program are to develop practical applications usable for safety analysis or operation and design and to expand the use of the NURESIM platform. Therefore, the NURESAFE project concentrates its activities on some safety relevant "situation targets". The main outcome of NURESAFE will be the delivery of multiphysics and fully integrated applications.

The platform will achieve the coupling neutronics, thermal-hydraulics, and fuel performance codes, at various physical and time scales. In particular it should incorporate new models addressing recent findings from safety research as well as demands from the current plant operation, as new fuel designs, higher resolution in energy, time and space. Full time dependent solutions of stochastic and deterministic 3D neutron transport should be developed to model heterogeneous core configurations.

In the NURESIM Platform is included the participation of twenty two European organizations ASCOMP, CEA, CHALMERS, EDF, HZDR, KIT/FZK, GRS, IMPERIAL COLLEGE, INRNE, IRSN, JSI, KFKI, KTH, LUT, NRI, PSI, TUDELFT, UCL, KIT/UNIKA, UPISA, UPM, VTT. The UPM is involved in the multiphysics and multiscale simulation capacity as a main objective in the nuclear reactor modelling nowadays, joint with the high performance computing power, in order to take advantage of the advanced hardware features as parallel computation. The multiphysics simulation capability may be composed of individual physics capabilities and common support services separately, and a software tool capable of integrating the separate pieces.

CP-ESFR: COLLABORATIVE PROJECT ON EUROPEAN SODIUM FAST REACTOR

Besides the Generation IV International Forum (GIF), which focuses on the medium-long term (> 2040), the international nuclear community are defining a new framework to fit with the short - medium term. Among them, the European Technology Platform on Sustainable Nuclear Energy (SNE-TP) and its European Strategic Research Agenda (SRA) propose a vision for the short, medium and long-term where the sodium technology as a reactor coolant plays a key role.

The roadmap for a European Innovative Sodium cooled Fast Reactor defines its specific R&D strategic objectives for a fourth generation European Sodium cooled Fast Reactor (ESFR). The roadmap addresses the needs for research and development, as well as for technology demonstration. The project ESFR follows this action identifying, organizing and implementing a significant part of the needed R&D effort.

The UPM contribution has been a Doctoral dissertation for development and verification of the European NURESIM platform for pin-by-pin and nodal coupled NK-TH simulation codes with application for the ESFR core physics and safety analysis. The analysis of the impact of the minor actinides concentrations on the transients results. The UPM has collaborated also in the analysis of the simulation results obtained with different TH codes for representative design basis accidents.

SARGEN-IV PROJECT

The ESNII was launched in November 2010 to anticipate the development a fleet of fast reactors with closed cycle. Three fast neutron technologies have been selected:

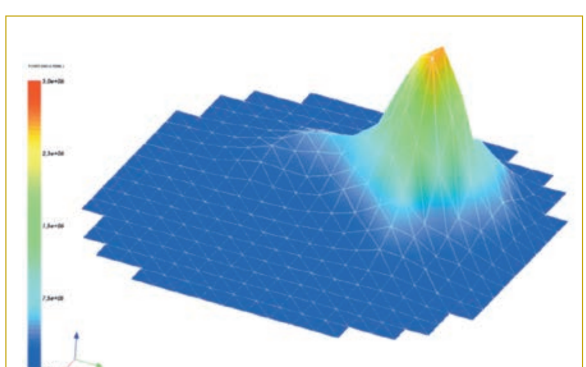
- the Sodium cooled Fast Reactor with the ASTRID prototype.
- the Lead cooled Fast Reactor with the ALFRED demonstrator which will be preceded by a pilot plan MYRRHA.
- the Gas cooled Fast Reactor with the ALLEGRO demonstrator.

With the objective of future assessment of these advanced reactor concepts, the SARGEN_IV Project is intended to gather safety experts from recognized European Technical Safety Organizations from Designers and Vendors as well as from Research Institutes and Universities to:

- develop and provide a tentative commonly agreed methodology for the safety assessment,
- identify open issues in the safety area, mainly addressing and focusing on assessment relevant ones,
- detect and underline new fields for R&D in the safety area
- provide a roadmap and preliminary deployment plan for safety-related R&D, including cost estimation.

With the aim of preparing the future assessment of these advanced reactor concepts, the SARGEN_IV Project is intended to gather safety experts from 22 partners from 12 Member States in order to:

- identify the critical safety features of the selected Generation IV concepts, relying on the outcomes from existing projects from the 7th Framework Programme (FP7),
- develop and provide a tentative commonly agreed methodology for the safety assessment, relying on the outcomes of the investigations carried out within international organizations (such as IAEA, WENRA, AEN), on national practices presently in use and on practices proposed within other European Framework Programs projects,
- identify open issues in the safety area, mainly addressing and focusing on assessment relevant ones, detect and underline new fields for R&D in the safety area (addressing methodological, theoretical and experimental issues, as well) in order to provide a roadmap and preliminary deployment plan for the fast reactor safety-related R&D.



Boron dilution transient power peak calculation (MW) in 3D with COBAYA3/FLICA coupled codes by UPM in the NURISP project.

IDPSA NETWORK

UPM, together with CSN and Indizen Technologies, participates in IDPSA network which stands for Integrated Deterministic/Probabilistic Safety Analysis. IDPSA is considered as a complementary to PSA and DSA approaches intended to help in: Resolving time dependent interactions between physical phenomena, equipment failures, control logic, operator actions in analysis of complex scenarios; Identification and characterization of a-priori unknown vulnerable scenarios, or "sleeping threats"; Consistent treatment of different sources of uncertainties; Reduction of reliance on expert judgment and assumptions about interdependencies; Potential reduction of the cost of safety analysis due to larger involvement of computers in what they can do better: multi-parameter, combinatorial exploration of the plant scenarios space.

IDPSA network provides a platform for: regular exchange of information and experience between IDPSA research and developers, and Potential users: PSA/DSA practitioners in Utilities, Vendors and Regulators.

ESNII+

The European Sustainable Industrial Initiative ESNII+ project (2013-2017) aims to define strategic orientations for the Horizon 2020 period, with a vision to 2050. To achieve the objectives of ESNII, the project will coordinate and support the preparatory phase of legal, administrative, financial and governance structuration, and ensure the review of the different advanced reactor solutions.

The UPM is contributing in two different tasks within the project. The first task is the assessment of core safety parameters for high fidelity transient safety analysis of the ASTRID sodium-cooled fast reactor. The second task is related with the evaluation of the sensitivity coefficients with respect to nuclear data, and the uncertainties on the reactivity coefficients using TSUNAMI-3D/SCALE6.1 module of the ALFRED lead cooled reactor.

EDUCATION AND TRAINING IN NUCLEAR SAFETY EUROPEAN PROJECTS: ENEN-III, NUSHARE, TRASNUSAFE

The ENEN-III project (2009-2012) covered the structuring, organization, coordination and implementation of training schemes in cooperation with local, national and international training organizations, to provide training to professionals active in nuclear organizations or their contractors and subcontractors. The training schemes provide a portfolio of courses, training sessions, seminars and workshops for continuous learning, for upgrading knowledge and developing skills in Nuclear Engineering.

The UPM has collaborated in the development of the training scheme for non-nuclear engineers and personnel of nuclear facilities, contractors and subcontractors. Two UPM alumni were also trained in the scheme for design challenges of GEN III plants.

TRASNUSAFE (2010-2014), a Project supported by the European Commission, aims at designing, developing and validating two training schemes on nuclear safety culture for professionals operating at a high level of managerial responsibilities in nuclear installations. One of the training schemes is related to the nuclear industry, while the other is related to the other installations making use of ionizing radiation based technology, mainly the medical sector. Both training schemes will have a common basis reflecting the challenging approach to risk management, followed by sector-specific specialized modules. The final product will consist in a package of five training modules for managers of both industrial and medical sectors, ready for use after validation through pilot sessions.

The objective of NUSHARE (2013-2016) is to develop and implement training and informing activities with the aim to share and grow, across EU Member States, the safety culture in nuclear installations. Security aspects (in particular, proliferation resistance and physical protection) will also be treated.

SENUBIO. GROUP OF TECHNICAL UNIVERSITY OF VALENCIA

The research group SENUBIO (Nuclear Safety and Ionizing Radiations Bio-engineering) forms part of the Environmental, Radiophysics and Industrial Safety Institute of the Universitat Politècnica de València (UPV). This group has strong relations with several universities and institutions abroad the world. We mention, Università di Pisa, particularly with the Professor D'Auria in the area of coupled codes, University of Dresden, particularly with the Professor Henning in the area of Boiling Water Reactor (BWR) instabilities, Technische Universität München, particularly with the Professor Macián in the area of uncertainties, Karlsruhe Institute of Technology, particularly with the Professor Sánchez in the area of coupled codes, Università de Milano, particularly with the Professor Zio in the area of Reliability analysis, University of Chalmers, particularly with the professors Demàziere and Pazsit in the area of neutron noise analysis, with the institute INSTN of the CEA of France, particularly with the professor Eric Royer in the area of coupled codes, with the Paul Scherer Institute and the nuclear plant KKL of Switzerland in the area of BWR instabilities, and with University KTH in the area of uncertainties. Regarding universities and institutions of America, the research group has relations with the Federal Universities of Rio de Janeiro and Minas Gerais in the area of coupled codes, with Pennsylvania State University, particularly with the Professors Ivanov and Avramova in the areas of coupled codes and uncertainties, with the University of Urbana-Campaign, particularly with the professors Kozlowsky and Udinn in the area of BWR instabilities, with the University of North Carolina at Rayleigh, particularly with the professor Azmy in the area of transport equations, and with the Tennessee University and Oak Ridge National Laboratory in the area of BWR instabilities.

Furthermore, we must mention that the research group has relations with the companies, AREVA, SIEMENS, WESTHINGHOUSE, EDF, STUDVISK and the Spanish companies IBERDROLA, TECNATOM… The research areas are neutronic and thermal-hydraulic coupled codes, BWR instabilities, neutron noise fluctuations and signal analysis.

Furthermore, SENUBIO Research group has participated in the USNRC CAMP program (Code Applications and Maintenance Program), which represents the international framework for verification and validation of NRC Thermal Hydraulic codes.

In this frame, SENUBIO has actively worked in the international projects OECD/NEA ROSA and OECD/NEA ROSA II, simulating different accidental scenarios using the thermal hydraulic code TRACE. ROSA (Rig of Safety Analysis) comprises a series of experiments performed in the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA). The OECD/NEA ROSA project has investigated issues in thermal-hydraulics analyses relevant to water reactor safety, focusing on the verification of models and simulation methods for complex phenomena that can occur during reactor transients.

The SENUBIO group has focused efforts in the simulation of the following experiments:

- Test 6.1: pressure vessel upper head Small Break Loss-Of-Coolant Accident (SBLOCA) under the assumption of total failure of HPI System.
- Test 6.2: pressure vessel lower plenum SBLOCA under the assumption of total failure of HPI system.
- Test 3.1: cold leg SBLOCA under the assumption of total failure of HPI system.
- Test 3.2: high-power natural circulation due to failure of scram during a Loss-Of-Feed Water (LOFW) transient under the assumption of total failure of HPI system, but an actuation of Auxiliary Feed Water (AFW).
- Test 5: condensation-induced water hammer tests.

Regarding to OECD/NEA ROSA II project, we have used the TRACE code to simulate the following experiments:

- Test 2: 17% cold leg intermediate LOCA.
- Test 3: 1.5% hot leg SBLOCA with an assumption of total failure of HPI system, under two different pressure conditions as a counterpart to PKL-2 Project test. Major test objectives were to clarify responses of core exit thermocouples (CETs) vs. fuel rod surface temperature at both of high-and low-pressure conditions corresponding to the pressure range of LSTF and PKL facilities.
- Test 5: thermal-hydraulic responses after a PWR Steam Generator Tube Rupture (SGTR) induced by Main Steam Line Break (MSLB).

Main results and conclusions of these works are summarized in different NUREG-IA reports (4 corresponding to ROSA project and 3 corresponding to ROSA II project).

Organizers of ROSA projects, OECD/NEA and the operating agent (JAEA) have encouraged the participation of all involved groups by means of different meetings (at least two per year) and a final Workshop "Joint PKL2-ROSA2 workshop on analytical activities related to PKL/OECD and ROSA/OECD projects" held in Paris (2012). Discussions and information exchange have been fruitful and allowed us to improve our knowledge of thermal hydraulic phenomena involved in important transients in Nuclear Safety.