POSITION PAPER
OF THE TECHNICAL
SAFETY
ORGANISATIONS:

RESEARCH NEEDS
IN NUCLEAR SAFETY
FOR GEN 2
AND GEN 3 NPPs



# FORFWORD

The "Position paper of the Technical Safety Organisations: Research needs in nuclear safety for GEN 2 and GEN 3 NPPs", so called "TSO paper", has been written by the European Technical Safety Organisations Network (ETSON).

This paper presents the views of the ETSON members:

- Bel V-Belgium
- GRS-Germany
- IRSN-France
- VTT-Finland
- UJV-Czech Republic
- LEI-Lithuania
- VUJE-Slovakia

with the contribution of ETSON associated members (SSTC-Ukraine, JNES-Japan) and safety authorities directly involved in defining safety research programmes (CSN-Spain, SSM-Sweden and KFD-Netherlands) in the EUROSAFE working group 2 dedicated to nuclear safety research.

Many research projects are currently implemented, either in the frame of the European Research Framework Programme, at an international or at a national level. The research topics listed in this paper are in compliance with the strategic research agenda of SNETP (Sustainable Nuclear Energy Technical Platform) and with national research programmes, each TSO bringing its experience feedback in the field of nuclear safety expertise. This initiative of ETSON members aims at enhancing nuclear safety and supporting research programmes at the European level and through SNETP.

This "TSO paper" presents a first list of identified research topics which may be not exhaustive. This list may be updated considering evolution of the safety needs (reordering and/or adding of research topics).

# CONTENTS

1 INTRODUCTION AND OBJECTIVES  1.1 Introduction	2
1.2 Position and objectives of the technical safety organisations in establishing research priorities	3
2 PRIORITIES IN R&D FOR NUCLEAR SAFETY	6
2.0 Safety priorities exercise	6
2.1 Safety assessment methods	7
2.1.1 Combination of deterministic and probabilistic approaches	7
<ul><li>2.1.2 The safety margins</li><li>2.2 Relevant and reliable material properties for extended service</li></ul>	8
(inspection, maintenance and repair)	9
2.3 Ageing	10
2.4 Safety related to coupling multi-physics	11
2.5 Fuel	13
2.5.1 Fuel behaviour during operation	13
2.5.2 Fuel management	15
2.6 Human and organisational factors	17
2.6.1 Human factors engineering (HFE) – Modifications and new builds	17
2.6.2 Human and organisational factors in operation	18
2.6.3 Safety culture	19
2.6.4 HOF in experience feedback	19
2.6.5 Management of competences	20
2.6.6 Other "emergent" research topics  2.7 Instrumentation and Control (I&C) systems	20 20
2.8 Internal and external loads and malicious acts	22
2.8.1 The integrity of equipment and structures	22
2.8.2 Malicious acts	23
2.8.3 Fire and smoke	23
2.8.4 Containment	24
2.8.5 Capabilities of TSO for measurements and test facilities	24
2.9 Severe accident, phenomenology, prevention and management	24
2.10 Emergency preparedness, crisis management,	
short terms consequences	27
3 CONCLUSION	29
4 REFERENCES	30
APPENDIX - Short synthesis on Fukushima Dai-ichi NPP accident lessons	31

# INTRODUCTION AND OBJECTIVES

## 1.1 Introduction

The European Technical Safety Organisations Network (ETSON) was invited by the Executive Committee of SNETP (Sustainable Nuclear Energy Technology Platform) to identify research priorities in order to participate in the SNETP project definition and launching.

This paper presents the views of the ETSON and associated members (Bel V-Belgium, GRS-Germany, IRSN-France, VTT-Finland, UJV-Czech Republic, LEI-Lithuania, VUJE-Slovakia, SSTC-Ukraine and JNES-Japan) and of some safety authorities directly involved in defining safety research programmes (CSN-Spain, SSM-Sweden and KFD-Netherlands) which share reflexion in the EUROSAFE working group n°2.

Many research projects are currently running, either in the frame of the European Framework Programme (FP), or through other international organisations, or directly at an international or at a national level; the results

of some of those contribute developing expertise in nuclear reactor safety. An important characteristic of the Technical Safety Organisations (TSO) is the duty and the capability of a constant questioning in order to enhancing nuclear safety. This questioning applies evidently to the Fukushima Nuclear Power Plant (NPP) accident and will affect not only the philosophy of expertise but also influence the priorities of the already planned safety research and, in a second time frame, put in light new research needs. A preliminary summary of the main lessons drawn from the accident is presented in the appendix, and it influences the research needs presented in the chapter 2.

In parallel to those proposals, a set of provisional subjects proposed by JNES (Japanese TSO) to his national authorities is summarized in the list below:

- investigation, analysis and evaluation on the Fukushima Dai-ichi NPP accident:
- investigation, analysis and evaluation on environmental impact by the Fukushima Dai-ichi NPP accident;

- evaluation of seismic source model, ground motion and tsunami taken in consideration in "The Tohoku District due to the Pacific Ocean Earthquake";
- research on evaluation of effectiveness of conventional seismic and tsunami countermeasures taken for nuclear reactor facilities;
- durability evaluation on nuclear equipment exposed to sea water;
- investigation on (radioactive) waste disposal due to disaster around NPP site;
- enhancement of safety of the fuel cycle facilities.

This accident will go on influencing the medium and long term research when more in depth analysis will be available. Safety organisations are working on these themes and will participate in developing their own lessons and corresponding research orientations, for example through the EUROSAFE working group and other international organisations.

#### 1.2

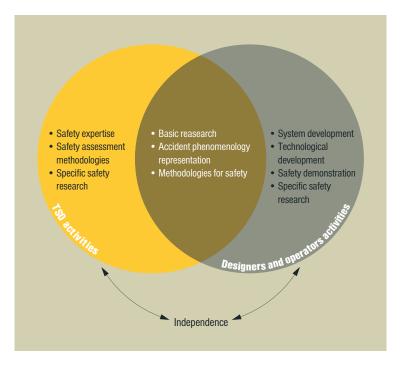
# Position and objectives of the technical safety organisations in establishing research priorities

While designers and operators are primarily interested in energy systems development with a due involvement in safety aspects, the safety organisations are primarily committed in evaluating any safety feature of these systems.

As a consequence, research supporting safety evaluation is devoted to understanding any phenomenon that would jeopardize safe and secure implementation of nuclear

energy systems and to developing reliable and accurate enough computing tools to represent them; research realised by the safety organisations is also devoted to investigate methodologies and phenomena that might influence the validity of safety demonstration supported by the designers and operators. Therefore TSOs have to provide the capability to maintain state of the art knowledge and facilities for safety assessment and to provide, beyond the aggregation of specialized expert findings, a comprehensive and holistic view of the safety issues.

The following diagram presents the fields of activities of safety organisations (TSO) in yellow and of designers and operators in grey. The common domain of activities is in brown



This diagram reflects that common research between TSOs and utilities belongs to "safety oriented" research, *i.e.* accident phenomenology and assessment methodology. This research allows optimizing the cost sharing without jeopardising the capacity for each actor in the safety to use this knowledge without compromising his independence, as far as each safety stakeholder has the capacity to performing its own safety research.

Figure 1: R&D activities of TSO and designers & operators.

Moreover, discussing the results of research and sharing this basic knowledge among all stakeholders allows progressively defining and increasing the domain in which phenomena and their numerical modelling are understood in the same way by all the actors in the safety debate. This justified increase in the safety science consensus liberates means to concentrate the safety debate on the remaining issues.

Schematically, on the same "object" of research, while the operator objective is the performance improvement with safety as one of the performance criteria, the TSOs objective will be the safety enhancement. Other aspects of TSOs researches may be characterized by:

- the necessity to examine all defence-indepth aspects, including for example with respect to severe accidents;
- the necessity to assess the utilities' proposals and so to develop the adequate competences through some R&D programmes in parallel with the utilities;
- the mission to worn their national authorities in case of detection of any danger for the public.

The second characteristic shows that at any time a methodological evaluation may result in identifying new research subjects. Beyond robust and simple methods for the safety demonstration that were developed for decades, this may lead TSOs, at the time being, to investigate complex methods (i.e. for example involving 3D thermalhydraulics coupled with neutronics calculation) because of needs in evaluating more precisely complex phenomena. The second and the third aspect may also lead TSOs to investigate specific subjects in the field of material and components ageing (which is not specifically a safety oriented perspective) to be sure that safety is preserved in case of long term operation. This may also lead TSOs to develop understanding the human behaviour and to consider other similar aspects that cannot be described with equations.

For those research subjects that pertain to the self commitment of TSOs, it is sure that the safety organisations will do their best, after preliminary convincing investigation, to persuade designers and utilities to have them included in the common research domain.

Consequently, potential subjects of research linked to safety cover a quite extended domain. Nevertheless, choices must be done and priorities have to be defined.

Elements of choice and of hierarchy considered by the TSOs include:

- the safety relevance. This aspect may be estimated considering, for NPP, the breach of the barriers and the associated safety margins;
- the level of knowledge of the subject and/ or the uncertainty associated to phenomena or event uncertainty;
- the importance granted to methods to give prominence to the risks like PSA (Probabilistic Safety Assessment) methods to assess safety margins;
- the prevention and mitigation of accidents. Though particular phenomena or events may be considered as very low probability events, their consequences have nevertheless to be studied and mitigated. The Fukushima Dai-ichi NPP accident clearly illustrates this aspect;
- the degree of importance and urgency for the challenges faced by the regulatory bodies;
- the degree of importance based on the lessons learned through operating experiences such as incidents and accidents;
- the "no later than" date which is calculated considering the milestones, the required duration and potential external supports (for instance projects supported by utilities).

As a consequence of the specific mission of technical safety organisations, the priorities for research subjects may be different from the one's identified by system designers and utilities. Technical safety organisations do hope to convince the SNETP sponsors, also considering their strong commitment and duty to operate existing and new built plants with the highest safety performance.

Failing to involve them, national authorities and international organisations should however consider that these subjects do correspond to existing risks identified by organisations that belong to the independent redundancy mean to protect the civil society.

#### 2.0 **Safety priorities** exercise

The safety organisations involved in this prioritization exercise have systematically addressed and ranked the different fission R&D topics adopting relevance-to-safety selection criteria. Even if the main todayknown outcomes of the Fukushima Dai-ichi NPP accident have been accounted for, no new explicitly-connected R&D topics have been considered.

The following diagram illustrates the practical process followed by TSOs to identify the priority research for them. At the different steps of this process, the different considerations

Figure 2: Prioritization process of TSOs.

#### **TSO position - prioritization process Setting priorities table of SNETP Extended paper** for the 2008 List of topics cross-cutting and safety from sept. 2010 file, adding LTO aspects updated and detailed ageing, severe accident, fuel... by current TSO concerns **Identification of common priorities - Feb 11 Paris - March 6** ETSON partners (IRSN, Bel V, GRS, LEI, UJV, VUJE, VTT) including authorities: CSN, SSM, KFD Optimization of the list and TSO paper on R&D explaining the choice of priorities and philosophy of safety assessment needs - March 17th Consolidated TSO position explicited in a paper

above have been applied to select the most relevant priorities.

The identified subjects were arbitrarily limited to 10. The next step which is already under implementation, is the identification of the existing programmes to feel the knowledge gaps and the available means (experiments or computing tools) to run research programmes in the corresponding field. The TSO priority ranking is also cross referenced, as far as there is, with the SNETP strategic agenda issued in the early 2009 and with the NULIFE OPERA road map. Various researches conducted within EC FP programs or under OECD/NEA coordination for a long time are also mentioned to clearly stress their complementarities with this updated position paper.

The rationale of the prioritization is described in the following paragraphs.

#### 2.1

## Safety assessment methods

In the traditional safety analysis, the regulatory acceptance of the limit values of the safety variables was set sufficiently stringent to ensure conservatism with respect to the onset of damage: margins were thus enforced in the analysis of the transients belonging to a safety set-case, quoted as the Design Basis – DB –.

Suitable methodologies, including the multi-dimensional computation capability in multi-physics, aimed at taking properly into account all the physical phenomena and the different uncertainty sources so as to evaluate the *safety margins* and their sensitivity to the initiators as well to the system status and operation.

#### 2.1.1

## COMBINATION OF DETERMINISTIC AND PROBABILISTIC APPROACHES

There is now a growing demand worldwide for a better quantification of the impact of the uncertainties affecting the physical parameters and the assessment of the reliability of the safety systems and the operators' action, as well.

Accordingly, the PSA are more comprehensively and extensively adopted both in the design, in the safety demonstration and continuous plant improvement during operation.

While the deterministic approach relies on a number of relevant transients and on the parameter values and the system reliability of which conservative rules are enforced to ensure the robustness of the design through the satisfaction of safety criteria, the probabilistic approach is aimed at considering the occurrence of all known events and their combinations, and thus privileges the completeness of the set of scenarios and initiating events.

The current trend is adopting a suitable combination of both approaches, the probabilistic one being devoted to the investigation phase and the deterministic one to the actual definition of parameter values for safety purposes.

To enhance safety in a perspective of long term operation of the GEN 2 fleet and to modify the design of GEN 3+ reactors, both probabilistic and deterministic methods have to be improved on subjects, such as:

- for the probabilistic methodology:
  - ☐ generalizing the *risk-informed approach* within the European countries to all existing reactors and for new design and operation modes;
  - □ formalizing the definition and use of the reliability data used in PSA in the context of LTO;
  - □ supplementing methodologies to support the severe accident practical elimination principle;
  - investigating of the likelihood of a common-mode origin for initiating events and transient propagation;
  - developing of suitable methodologies to investigate the reliability of both active and passive safety systems;

- ☐ formalizing the treatment of uncertainties associated with probabilistic safety analysis;
- improving the quality the PSA 1, 2, 3 developed in European countries.

For example, the project ASAMPSA2 covers partially the above subjects. Within ASAMPSA2, some needs for further activities have been already identified (treatment of external events, review of existing L2 PSA models on the basis of the guideline, development of probabilistic tools...) but other activities may be defined and promoted by ETSON partners.

The post-Fukushima Dai-ichi NPP accident investigations do not threaten the previously defined R&D programs but they claim for a rearrangement of the priorities and more accurate definition of some tasks with a particular focus on the treatment of extreme external events and operational management of a multi unit nuclear site. PSAs should be able to address properly these issues.

- for the deterministic methodology:
  - up-dating and expanding the lists of initiating events, systematically including:
    - the shut-down, zero-power and intermediate-power states of the plant;
    - the transients originating outside the reactor – e.g. in the temporary fuel storage pools;
  - □ accounting for any combination of system failures;
  - □ up-dating the assumptions on system reliability and performance;
  - investigating the human and organisation factor in depth, including the human aptitude to manage in crisis conditions:
  - up-dating the safety criteria to adapt safety objectives to new reactor design.

It is straightforward that all the abovementioned topics are likely showing-up in an increased need for very demanding research in different fields of endeavour. Several projects can gather partners to address these issues within the topic-area. The following examples of collaboration can be mentioned:

- the PHÉBUS experimental programs were carried-out to check and validate physics and phenomenology completeness in case of Large Break Loss Of Coolant Accident (LB LOCA), for the core cooling, the hydrogen production and the control of oxidation rate;
- the collaborative investigation on the reliability of the passive safety systems.

#### 2.1.2

#### THE SAFETY MARGINS

A growing interest is developing worldwide for the *safety margins* methodology, because it comprehensively combines the advantages of the deterministic and probabilistic approaches.

The safety margins have been traditionally adopted with the ultimate goal of protecting the public and the environment from the radiological consequences of the releases from the nuclear power plants (NPPs). They provide a metric to evaluate the sensitiveness of the risk to the uncertainties, either aleatory or epistemic, which affect the safety variables.

These variables are represented as a set of volumes constrained by successive physical barriers, the integrity of which is threatened by internal and external events and aggressions, including malevolence. Protective design options and safety systems (both active and passive ones) are intended to preserve the integrity of these barriers or at least to mitigate the effects of their failures under internal and/or external loads.

An extension of the approach has been recently proposed, based on the likelihood of incurring some damage in a event sequence, *i.e.* on the conditional probabilities of barrier failure (or bypass) leading to the damage or to the release of contaminants to the environment.

In this approach, the set of Design Basis (DB)

scenarios is extended to the complete set of all credible scenarios, including out of design situations, which all together generates the so called *risk space*.

Unlike for the PSA, all the safety criteria currently addressed in the deterministic DB accident analysis are to be addressed to define such a *risk space*.

Moreover, uncertainties of several types, including all those currently assessed in traditional deterministic accident analysis and PSAs, can be explicitly taken into account in the calculation of the conditional damage probabilities. The use of realistic dynamic models is encouraged in the safety margin approach because the end state of a sequence is no more relying on success criteria for safety functions which depend on a broad set of thermalhydraulic and mechanical calculations.

Any change in the design and/or the operating mode of a NPP may have an impact on the *risk space*, because it could challenge the *safety margins*, in spite of fulfilling all regulatory requirements. Power up-rate, plant life extension and increased fuel burn-up as well as the cumulative effects of simultaneous and/or subsequent modifications are pertinent examples for application.

The significance of the approach increases even more as new designs push the plants closer (or even beyond) the edge of the original design space.

The generalization of the safety margin approach is likely to ask for new and very demanding research, mainly in the field of modelling, advanced computation and reliability evaluation.

Examples of projects in the topic area are:

- the SM2A (an OECD/CSNI project) on the evaluation of the safety margins variation in case of a power up-rate;
- the pre-normative research in safety, i.e. the review of technical criteria insuring the safety objectives.

#### 2.2

### Relevant and reliable material properties for extended service (inspection, maintenance and repair)

The safety demonstration of the GEN 3 and 3+ NPPs relies on the exclusion of any early release. It is reinforced for GEN 4 with the severe accident practical elimination, which states that the severe accidents have to be practically excluded by design "so that only limited protective measures in area and time are needed for the public" during the whole NPP operation lifetime. This principle is likely to be generalized and extended progressively to the current NPPs, too, to achieve a fully harmonized safe operation of the different NPP fleets.

Obviously, the satisfaction of those objectives relies on reliable design options, robust numerical computations and improved component technology, but also, and mainly, on the capacity of an early detection of any threat which can challenge the integrity of the barriers. That requires also extended and enhanced inspection, maintenance and repair capacity.

More generally, the likelihood for degradations during the operation of the plant is accounted for at the design stage when defining:

- the limit values of the safety variables to be matched during normal and degraded operation to ensure the respect of the safety objectives (protection of the workers, the public and the environment);
- the operating conditions;
- the external/internal hazards;
- the features of the components and structures, including their design, qualification, fabrication, examination and protection;

■ the Design Basis recovery.

Margins are accordingly enforced to the component and system design, accounting for their relevance to safety and the feedback from the plant operation.

These conservatisms are generally translated into integrity, stability, functionality or resistance-to-the-break requirements for passive components and systems and operability for active ones. They can also account for the PSA boundary conditions, such as the initiating events for passive and active components, the Instrumentation and Control (I&C) features....

The improvement of methods and techniques for inspection, maintenance and repair capacity, taking into account notably ageing as well as the enhanced reliability of components and systems expected by the utilities, is likely to require new and specific R&D efforts in different fields to ensure that the basic rules of the safety demonstration are still satisfied all along the NPP life cycle. Industries will lead the programmes but TSOs are interested in understanding the different methods.

## 2.3 Ageing

Ageing is currently a major challenge for extended operation of NPPs.

Ageing assessments have been generally performed – and they currently are – and are a challenge for TSOs in ensuring the conservatism of data, physical assumptions, computation models and methodologies, as well are their proper justifications, considering:

- the large uncertainties originating from:
  - □ the insufficient knowledge of the ageing mechanisms:
  - □ the lack of plant operating experience on ageing;
  - □ the scarce available experimental evidence, due to an insufficient number

- of suitable and targeted experimental programs.
- the crucial need for simplification dictated by:
  - □ the limited computation capacity, including running-time, capability to refine meshes in the finite element analysis;
  - □ the missing or anyway insufficient coupling of codes, which prevents for any ab initio accurate investigation;
- the industrial constraints on component properties data;
- the evolution of some industrial practices, for example, use of low neutron fluence core loading or evolution of the primary and secondary systems chemistry;
- the lack of agreed international context for norms and practices.

Accordingly, the concern in mastering ageing problems spans a very large and diverse domain and claims for, at least:

- a common understanding of the ageing mechanisms of materials and components, a crucial issue which can end-out in both theoretical investigations and experimental programs, conducted through an enlarged network of international collaboration;
- the development of both dedicated computation tools and advanced multiscale computation code systems capitalizing the acquired knowledge;
- the implementation of suitable component with properties suitable for a long-term operation;
- the harmonization of practices (at least in European countries) with the abovementioned R&D activities;
- the development of in-service inspection methods and tools, as well as the capacity to decide replacement in due time.

Summarizing the above-mentioned, the following priorities in R&D activities could be defined from the TSO perspective. Significant improvements are necessary in the physical understanding of all relevant ageing mechanisms and their driving parameters. The main goal would be their pro-active anticipation and acknowledgement, including those either of low importance in the current plant operation perspective (such as the creep and thermal ageing) or not yet clearly identified. It would be worth identifying not only the thresholds for initiation of defects and the kinetics for their propagation, but also the precursor state which could engender defect nucleation. That would enable an early detection of potential degradations and allow an efficient mitigation of the ageing consequences.

Presently, the following priorities for investigation are identified:

- corrosion (e.g. stress corrosion cracking);
- concrete ageing;
- fatigue;
- irradiation effects;
- material embrittlement;
- stainless steel cracking in stress conditions;
- creep;
- thermal ageing.

Knowledge on ageing should be based on both:

- analytical experiments and operating experience feedback (which demands for extended examinations on actual components, before and after replacement);
- numerical simulation, which can be used to obtain the relevant parameters, and prioritizing them through sensitivity analysis.

As far as the computation capacity is concerned, there is an acknowledged necessity for improving the modelling of these mechanisms to ensure reliable long-term predictions of the ageing and its effects. That claims for modelling fundamental phenomena in physics and chemistry, at different scales, from atomic to macroscopic level.

Moreover, the numerical simulation capacity should be extended to include the evaluation of local values of variables from the averaged ones generally provided by measurements.

Some research projects could be launched under the NULIFE network in the FP7 of the European Commission, or under complementary initiatives, or could be driven by TSOs independently from industry.

#### 2.4

# Safety related to coupling multi-physics

Advanced computer hardware and significant progress in the software design of simulation platforms make it nowadays possible to tackle the modelling of transients and accidents in a "multi-physics" fashion, i.e. by describing the actual physical phenomena through a tight coupling of thermalhydraulics, thermalmechanics, reactor physics, and overall system response. Due to the vast number of physical variables, the massively increased computational effort as compared to lumpedparameter system codes, and the complex user interface, such coupled code systems require extensive verification and validation efforts. As a consequence, system codes have remained the major "work horses" for the description of phenomena occurring in the primary and secondary circuits during the transients, in order to demonstrate compliance with the regulatory safety limits. However, to demonstrate that the safety limits are still satisfied considering enhancing performance, components ageing and better physical modelling, the methods and codes used for the safety demonstration by the utilities are more and more complex.

In order to limit the run-time requirements, system codes widely adopt simplified representations of flow characteristics which impact the mixing, stratification, natural circulation, and heat and mass transfer in two-phase flows. In an attempt to account for the 3D-nature of phenomena, most system codes massively rely on closure laws (correlations) derived from representative, large scale, thermal hydraulic tests carried-out in devoted facilities. As a result, in many cases, they are neither scalable nor adaptable in geometry.

Consequently, to be sure that the phenomena and uncertainties are correctly addressed and that safety criteria are fully satisfied, the computer codes still need further development, mainly as far as closure laws are concerned: reducing dependence on empirical correlations represents a rather general challenge in the domain of thermalhydraulics.

In the near future, investigations of very local and/or highly complex phenomena, such as:

- the flow stratification in pipes and tees;
- the cold plumes touching hot walls;
- the impinging jets with temperature differences;
- the Pressurized Thermal Shock (PTS), a major safety issue, intimately connected and relevant to ageing;
- the thermalhydraulic phenomena in the secondary phase of the steam generator (considering high void fraction and dissymmetric situations such as clogging, fouling...);
- the local double-phase phenomena in case of LOCA;
- mixing phenomena in downcomer and lower plenum, e.g. for the assessment of boron dilution events;
- transport of insulation material during a loss-of-coolant event ("sump clogging");

 inhomogeneous two-phase flow distributions for the assessment of cooling conditions of heated bundles (especially relevant for BWR);

are expected to rely more and more on Computational Fluid Dynamics (CFD) techniques to complement the conventional tools and obtain more precise physical descriptions.

In principle, CFD has the potential to address this challenge appropriately and to handle arbitrary geometries and scales. For this reason, CFD codes are expected to be applied more frequently in reactor safety analysis, both for single-phase and two-phase applications in Light Water Reactors (LWR).

Single-phase CFD applications are already reasonably mature to address all these issues, although some models need improvement, but two-phase and multi-phase CFD modelling still require R&D efforts. For the application to LWRs, the single-phase CFD tools must be further validated against a range of dedicated integral experiments. The development of efficient sensitivity and uncertainty propagation methods for the application to CFD-problems becomes important in light of the vast amount of computational data.

Accordingly, the implementation of suitable CFD techniques will be a major R&D challenge in the short/mid term, which can widely profit from the progress of the advanced numerical simulation. It will provide users with an extended capability to calculate local parameters and allow a better understanding of the physics, which should end in more reliable designs, reduced costs and/or precisely quantified safety margins.

Advanced numerical simulation tools are necessary for the simulation of the relevant scenarios from physics first principles to meet the requirement of scalability of the calculation tools. For the evaluation of the operation and safety of current and future (GEN 3) reactor designs the new tools are expected to offer better accuracy, higher computing efficiency and increased user-friendliness.

The following broad areas of development have been identified:

- availability of verified and validated tools (correctly solving the appropriate equations and compared against relevant experimental data);
- availability of multi-physics tools connecting several disciplines necessary for reactor calculations;
- availability of multi-scale tools (connecting several scales from the microscopic to the macroscopic level);
- availability of methods to propagate uncertainties of (coupled) multi-physics and multi-scale tools using both probabilistic and deterministic methods;
- availability of generic tools to manage the supervision of the calculations as well as pre- and post-processing of very large data sets of computational results (workflow description, tools for mesh-generation incl. importing CAD-data, visualization, assimilation of experimental data with simulations, archiving and data mining).

Research is duly expected to sustain the production of *best-estimate* simulation tools widely validated for the modelling of thermalhydraulics and core physics phenomena, and suitable methodology for predictive estimation of errors. To conveniently address these needs, it would be worth:

- accommodating the progress in computing techniques, such as the parallelisation through supercomputing and the improvement and extension of existing recovery methods and residualbased (or adjoint) methods to unsteady (parabolic) and hyperbolic problems;
- improving and systematizing the validation methodology and approaches, which today are quite diverse, because they combine, at different extents, comparisons of code predictions with measurements from representative mock-up experiments to derive biases and uncertainties for design and safety studies - and more

formal and structured processes, where the individual sources of discrepancy are identified, quantified and propagated through statistical techniques.

A first step in this direction has been undertaken by the NURESIM Integrated Project in FP6 which has started the development of a European reference simulation platform for nuclear reactor applications, with a focus on PWR, VVER and BWR. The project has involved 13 European countries and 18 partners. According to the NURESIM roadmap, the project is to continue with the NURISP and NURENEXT Coordinated Projects now underway and/or planned within the FP7 programme.

#### 2.5 Fuel

2.5.1

#### FUEL BEHAVIOUR DURING OPERATION

The fuel behaviour, in normal and degraded conditions, currently is and will continue to be a major issue for the safe, secure and economical operation of NPPs.

Utilities are continuously looking for improvement and optimization of the nuclear fuel. In recent times, new core designs, advanced loading strategies – including hybrid and low leakage ones - and operating modes – including enhanced flexibility in operation - have been implemented by utilities. Therefore, nuclear fuel is operated at ever more aggressive conditions.

All those changes claim for more research on fuel, which has to accommodate new design and safety requirements, higher Uranium enrichment and Plutonium recycling, higher burn-up, up-rated power, as well as increased life-time and cycle length. In the meantime, LWR fuel technology undergoes improvements on cladding materials (new alloys), and fuel pellets (different types of MOX, Gadolinium poisoning, doped and the large grain-size fuels, short fuel pellets....).

Moreover, the fuel related safety criteria established in the sixties and seventies are currently being reviewed.

Modelling of fuel performance strongly relies on proprietary data, and results in a proprietary-vendor-code. Thus, sharing knowledge and development turns-out quite difficult. Nevertheless, the collaborative work in the area of LWR fuel behaviour is very important and for normal operation conditions has benefited from international benchmarks. An extension of these benchmarks is required in view of new fuel designs. Moreover, R&D programs addressing RIA (Reactivity Insertion Accidents) and LOCA are in progress in Europe, and additional ones are under discussion. Test reactors are involved in R&D program on fuel behaviour (integral testing, mainly) under operational and accidental conditions, particularly the integral testing in the NSRR, LSTF, Halden and CABRI facilities. In the future the Jules Horowitz reactor (JHR) in south of France could also be involved in such research. In parallel, analytical tests are on-going in several European countries with irradiated or simulation materials.

Among the main fuel-related pending R&D issues, the focus is to be put on the clad ballooning and fuel relocation, which can affect:

- in the case of LOCA or RIA, the short and long term coolability threatened by the clad ballooning;
- the cladding mechanical behaviour and failure conditions both in LOCA and RIA

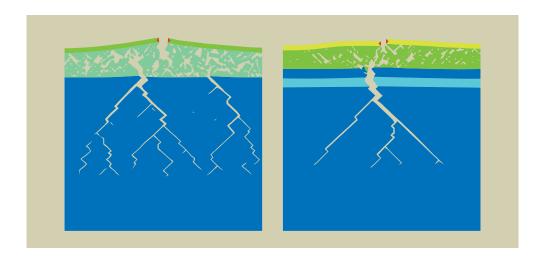
(figure 3) (effect of hydrogen uptake, axial load during quenching, effect of rod contacts on cladding failure conditions);

- the axial relocation of fuel in the ballooning parts of the fuel rods;
- the effect of fuel ejection, particularly from high burn up fuel, on the core in LOCA and RIA;
- the fission gas behaviour in the fuel matrix;
- the source terms aspects in case of LOCA and RIA;
- the heat transfer between clad and coolant in case of RIA;
- the coupling of fuel thermo-mechanical modelling with thermalhydraulics and neutronics (impact of the fuel behaviour on neutron parameters during normal and degraded operation).

Suitable research programs on thermal hydraulic aspects and coupled neutronics/ thermalhydraulic aspects and on fuel behaviour, connected mainly to LOCA or RIA condition, should be continued, relying on dedicated experimental facilities and international networking (figure 4).

At present, theoretical research is addressing the multi-time-scale simulation. The fuel rod behaviour over time scales from milliseconds to years is predicted by 2D fuel performance codes, while more detailed models, including 3D approach at very local scale, are available

Figure 3: Advanced modelling of clad rupture during a RIA transient.



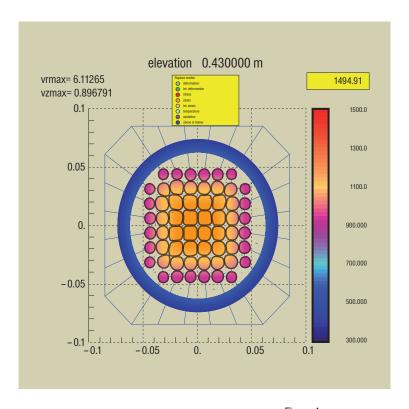
to predict some specific aspects like fission gas behaviour, fuel fragments relocation or cracks propagation in the cladding. Support experiments include scanning and transmission electron microscopy, ceramography analysis, secondary ion mass spectrometry, thermo-physical properties measurements (thermal-diffusivity, heat capacity), as well as Knudsen-cell measurements for irradiated fuels.

To achieve a complete multi-scale simulation, the current methodology needs to be complemented with atomistic techniques such as the molecular dynamics (MD) or the kinetic Monte Carlo (KMC) in combination with electronic state calculations (ab initio). Since the availability and accuracy of the inter-atomic potentials, including those for multi-component alloys, is a major challenge to the adoption of atomistic techniques, comparison and experimental validation are necessary. This process has already been launched in some organisations (IRSN ...) during the last years (figure 5).

Collaborative works have been performed in the past to assess the LWR fuel behaviour in accidental conditions; they are planned to be continued. Current development efforts mainly address the single fuel rod modelling during LOCA and RIA, accounting for fuel fragmentation and relocation, clad stress and ballooning, clad rupture, ejection of fuel fragments and possible under-cooling. A full 3D representation of the thermo mechanics of the whole fuel rod and even a representation of the whole fuel assembly are believed necessary for a reliable simulation.

## 2.5.2 FUEL MANAGEMENT

Safety is a major concern through the whole fuel cycle, which involves fuel fabrication, fuel storage and transportation, fuel reprocessing, and waste management. Globally, the safety problems encountered in fuel handling are especially related to the risk of criticality and of fuel (or fuel containment) degradation implying release of radioactive material to the environment.



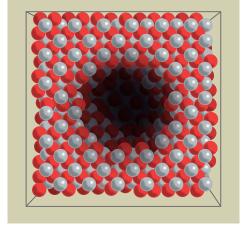


Figure 4: Example of DRACCAR IRSN code simulation of cladding deformation and rod contact of fuel bundle after a LOCA transient.

Figure 5: Atomistic simulation of a fission gas bubbles in UO<sub>2</sub> matrix.

1) Safety authorities must be able to determine whether measures implemented by operators are acceptable regarding criticality and release of radioactive products, and be prepared with the management of situations involving the occurrence of such events. Studies that provide perspectives of these risks should be performed for consideration by decision-makers imposing administrative and design controls on spent-fuel storage and transportation casks. Also, knowledge of the steps that can lead to criticality accidents is necessary to evaluate the safety of the nuclear fuel cycle, optimize detection and be prepared in case of an accident.

Solving a criticality problem is performed using nuclear data, and a series of models describina neutron phenomena are integrated in computer codes. Such models describe for example neutron migration across matter, neutron leakage from the system, slowing down of neutrons, etc. Since several years, calculations are performed by considering more realistic configurations, and by modelling in detail the surroundings of the fissile material. These tendencies, encouraged by increasing computing performances, are correlated with economic optimization needs. Indeed, these surroundings have an impact on the neutron interactions, and may therefore be determinant to justify sub-criticality of nuclear installations or transport conditions. These codes need thus to be able to solve very different problems because of the diversity of fissile environments encountered and of equipment geometries that are designed. However, these optimized calculations generally imply a decrease of conservatism in the safety assessment. Qualification of these numerical codes is essential, and relies, besides others, on comparison between numerical simulations of specific benchmark cases and corresponding experimental results. In the TSO's point of view, more complete experimental campaigns (as the one performed during the MIRTE program) as well as theoretical work in a detailed modelling of neutron behaviour should be sponsored, in order to improve numerical codes (as the MORET code of the IRSN for example).

2) Storage of the spent fuel is also a source of specific concern. Before reprocessing or dry cask storage, residual heat released from spent fuel must be removed for a long time. Spent fuel is stored in interim water pools, and forced water circulation around the fuel elements is imposed for its cooling. Such devices rely thus on the good working of heat removal systems, and in particular of some active elements (pumps). The Fukushima Dai-ichi NPP event has unfortunately shown that this cannot always be guaranteed, and passive heat removal systems are progressively being designed. In these designs, natural convection of water should be sufficient

to ensure heat removal of the spent fuel. Some configurations are also considered to ensure, in case of water leaks of the pool, heat removal with natural convection of air. Validation of such devices is more difficult, as it relies on the use of more complex 3D thermo-hydraulic codes to account for instabilities that could, for example, prevent natural convection and as margins with acceptable limits are getting smaller.

Further, several nuclear power plants will be out of room in their spent fuel pools in the next years, requiring pool enlargement (if possible) or rearrangement, or use of temporary storage of other kind. Fuel assemblies themselves have become more reactive. Also, the neutron absorbers in the pools installed to maintain criticality requirements have exhibited degradation. Some have lost a significant portion of their neutron absorbing capability, leading to criticality concerns. To accommodate these factors, both the criticality analyses and the storage design requirements have become more complex, leading in general to a reduced conservatism claiming for more accurate models.

In the TSO's point of view, safety reasons justify improvements in the design and modelling of the spent fuel pools, with special attention to the above mentioned concerns. The Fukushima Dai-ichi NPP events motivate also better prediction capabilities of the consequences in such systems during accidental conditions. This requires in particular adequate theoretical and numerical two-phase flow models.

3) Finally, one of the key issues in nuclear energy, also identified as one of the 8 key R&D target areas at the SNETP governing board of 31 March 2011, is the elaboration of acceptable and safe solutions for the radioactive waste produced during the various stages of the nuclear fuel cycle. These waste products are classified into long-, medium- and short-lived waste. Longlived waste products especially constitute a very important social problematic, as they imply possible environmental repercussions for several future generations.

Management of nuclear waste needs further significant research efforts. Firstly, possibilities for waste volume reduction are examined, considering several conditioning methods or even transmutation to shorter lifetime products. The Joint Research Centre of the European Commission (JRC), for example, is particularly active in such research, through the Institute for TransUranium elements (JRC/ITU), based in Germany, and the Institute for Reference Materials and Measurements (JRC/IRMM), based in Belgium.

Secondly, deep geological disposal concepts are studied for disposal of long-lived waste products. Such disposal concepts are based on a multi-barrier system that contains and isolates the radioactive waste from the biosphere. The entire disposal concept must provide isolation for considerable long periods. In order to gain confidence in the long term capabilities of such devices, safety research is still required in several domains, as for example:

- repository sealing tightness;
- long term degradation of engineered barriers (e.g. corrosion of the container);
- radionuclide geochemistry and migration;
- risks associated with reversible storage, like fire....

Several underground laboratories exist in which such studies are or could be performed: TOURNEMIRE in France or HADES in Belgium for example. These studies should help TSOs in the global safety assessment of deep geological disposals.

# 2.6 Human and organisational factors

Experience shows that human behaviour and organisation play a major role in safety during the life cycle of nuclear installations, not only in operation, but also in design,

construction, modification, and dismantling. Human and organisational factors (HOF) may be defined as all elements of the working situation and environment that may have an influence on human performance. The integration of HOF in safety and radiation protection expertise concern a wide range of topics related to individual, technical and organisational characteristics and it needs most often multi-disciplinary approaches to deal with.

Human and organisational factors are now more and more considered as important for safety. Indeed, nuclear community has to cope with a lot of challenges in the operation of existing plants and the design of new ones. These challenges, such as new technology, modernization programs, organisational changes, design of advanced reactors, changes in staff and competences, evolution of requirements and regulation, etc. all have an influence on how human and organisations may work safely in daily operation of plants and manage all kind of situations, including unanticipated ones.

Knowledge and methods from academic field as well as practices from different industrial fields are available for supporting nuclear safety expertise. But because HOF is quite a new discipline which covers a broad scope of areas and because nuclear safety faces changing requirements and challenges, a lot of researches have still to be carried out for improving their integration into safety expertise. They should provide more knowledge useful for supporting expertise but also a comprehensive selection of validated approaches and methods.

In particular, following broad topic areas could warrant further research.

#### 2.6.1

## HUMAN FACTORS ENGINEERING (HFE) – MODIFICATIONS AND NEW BUILDS

It is now a commonly recognized principle to integrate HOF analyses within design processes, for the design and construction of new plants as well as modification and modernisation of existing plants, from the starting phases of projects to the operation phases. A main objective of such integration is to create the conditions to minimize the risk of human errors during operation.

A lot of experience, knowledge and methods are already available in that area, coming from researches and applications in different industrial fields including nuclear field. But because technology is in constant evolution and expectations are at a high level in nuclear safety, further researches should improve the design and use of man-machine systems facilitating a high level of human performance for safety.

Several examples of possible research topics in this area are:

- better understanding the influence of automation on operations (multi-agent systems¹, enhancing of automation transparency, ...);
- studying impact of automation or computerised operating system failures and degraded conditions on operating and safety. Experience shows that operators have often difficulties to manage such situations where the behaviour of the computerized operating system is not clearly understandable;
- evaluating the applicability of innovative human-system interaction concepts and technologies in the nuclear field. For instance, to study the influence of new computerised tools for easily-available documentation (in place of the current NPP manufacturing documentation) on safe operation;
- determining the impact of operational complexity on human performance in order to reduce complexity in new design;
- plant operation improvement can be achieved by enhancing the human performance and reliability simplifying operating procedures and providing appropriate support. Large progress can be done thanks to numerical simulation and high performance computing,

enabling visualization of phenomena as they occur in the circuits. Fast-running simulation could also provide operation with decision-aid support;

developing human-centred design<sup>2</sup> approaches for new plants and plant modifications, in the perspective of integrated design of socio-technical system for a better integration of technical, human and organisational aspects in the design process.

# 2.6.2 HUMAN AND ORGANISATIONAL FACTORS IN OPERATION

A lot of components of the working situation and environment have an influence on human performance during operation of nuclear installations, at different levels such as individual, teamwork, organisation, management, as well as other elements coming from outside the licensee such as regulation, population, economical competition, etc.

This is a broad scope and complex area that should benefit from development of research topics involving disciplines such as ergonomics, psychology, sociology, anthropology, risk management as well as management sciences, economy, etc.

Some examples of possible research topics in this area are:

- searching for a better understanding of human and organizational practices in normal, degraded and incidental operation of current reactors in the international framework could lead to the definition of practices to share and take advantage from this knowledge;
- improving the ability of operators to manage unanticipated events and situations is of paramount importance regarding experience and accidents, in particular accounting for the recent one in Fukushima Dai-ichi NPP. A research issue is how to aid personnel to recognize that they are in an unanticipated situation;

1>

A multi-agent system is a system where human and machine agents work cooperatively to accomplish plant safety and production goals.

2 >

The human-centred design approach may be defined as an approach which involves as much as possible potential users of the system during the entire process design and development.

- improving preparation and realisation of maintenance activities could benefit from research and developments in areas such as technical, human performance or organisational ones. An example of research topic in technical area is virtual reality models, which could be very worth to inspect containment building where access is not easy before the outage, and could also be used to train maintenance personnel or prepare maintenance tasks;
- identifying elements for designing reliable organisation for outage, that is a sensitive period during operation of a nuclear power plant.
- studying communication needs and practises among staff on different types of situations in order to provide new tools or to improve existing ones. For instance, Radio Frequency Identification (RFID) could help checking on-line the pertinence of actions underway; communication could be improved, reducing the risk of mistakes or misunderstanding;
- identifying and analysing the impacts of high-level industrial strategies of licensees and/or institutional actors on low-level human activities on the field in nuclear plants. An example of such strategies is the outsourcing activities; another one is productivity pressure;
- carrying out extended research for better understanding and analysing the impact of subcontracting in human-issue sensitive industry, including non-nuclear one;
- justifying a better understanding and defining characteristics of robust and reliable organisations (High Reliability Organisations HRO researches). Resilience engineering is a concept developed quite recently and which could be useful for that research topic;
- studying the role and influence of individual motivation and commitment in management of safety. How to promote and facilitate consideration of safety in an appropriate way in decisions and actions performed at all levels of the organisation.

- Outcomes should lead to the identification of elements favouring or hindering safety sense-making for nuclear plant personnel;
- analysing the impact of prescriptions, procedures, instructions, etc. on the human performance in sensitive situations. An issue is to better determine an appropriate level of guidance to operators.

## 2.6.3 SAFETY CULTURE

Safety culture is a widely used notion, although it suffers from a lack of theoretical background. The notion would greatly benefit from being reconceptualised on the basis of an anthropological view of culture. In this later view, culture is not a variable "manipulated" by managers, but what emerges from groups of people and teams. Culture may be studied with the help of an ethnographic approach, which demand extensive field work, in order to grasp those aspects of safety that often remain hidden with other perspectives.

### 2.6.4 HOF IN EXPERIENCE FEEDBACK

Operating experience feedback (OEF) is a very important process for taking lessons from experience to improve safety. To be a learning organisation is an essential characteristic of high reliability organisations.

There are different existing processes for dealing with OEF, but HOF causes are most often insufficiently considered and analysed. The level of analysis not always allows to identify root causes, in particular organisational ones, and then to define appropriate corrective actions. Another difficulty is that OEF processing is resource demanding and that organisation's motivation may decrease in long term when no major event occurs.

Examples of possible research topics in this area are:

- better identifying and characterising the favourable conditions and the barriers for improving the integration of HOF in root cause analysis. In particular, outcomes should lead to the development of approaches and tools for in-depth analysis of organisational factors in OEF;
- getting lessons of accidents in general, not only from nuclear field, often show a lack of communication and transfer of knowledge related to OEF within organisation and also among organisations. Research should study reasons of such a lack;
- identifying characteristics of a perennial and robust organisation for dealing with OEF in long term.

#### 2.6.5

#### **MANAGEMENT OF COMPETENCES**

Management of competences concerns the development, maintenance and renewal of skills and competences in nuclear community.

Examples of possible research topics in this area are:

- competences at teamwork level (in particular characterisation of collective and distributed skills);
- transmission of knowledge and expertise among several generations of workers;
- reinforcing interaction between university and industry, this could be sustained by suitable educational programs at the European level (education of a new generation, training and attracting of nuclear technicians, engineers, and scientists).

#### 2.6.6

#### OTHER "EMERGENT" RESEARCH **TOPICS**

- Development of methods for organisational diagnostic, in the perspective of preventive approach.
- Decision making processes and risks in

design. HOF are very often focused on the licensee's activities: for instance, the organisation of the Safety Management System, the elaboration of procedure, the management of competencies, etc. However, risks are also managed through the relation between the licensee, the regulation authority, and eventually the technical support organisation. These aspects are not sufficiently explored, although a number of accidents must be explained by considering the whole set of institutional actors. This is for example the case in the analysis of Vaughan on the "Challenger Launch decision". More recently the Fukushima Dai-ichi NPP accident seems to give evidence that the type of relation between these institutional actors in safety are of utmost importance. There is a new challenge to be taken concerning the sociological and historical study of past decisions in nuclear installation design and modification, taking into consideration all the actors involved.

#### 2.7

### Instrumentation and Control (I&C) systems

Electrical Instrumentation and Control systems (I&C) (both active and passive ones) provide the NPP operation with the necessary control, protection and safeguard capability in any normal and degraded operation as well as accidental circumstance. That way, they are relevant to the plant safety through the reliability of the defence-in-depth approach.

The design, qualification, implementation, operation and maintenance of I&C are of prime safety-importance. Accordingly, they have to be accurately controlled and periodically checked against any likely danger of inoperability, degraded-operability and/or failure, in a whole. The same stringent criteria must be applied to their individual components.

Ageing threatens the electrical and electronic components of the control, safety and safeguard systems, so increasing their failure probability, which affect their reliability vs. time, in a whole.

Moreover, the I&C systems show-up very sensitive to the component obsolescence. affects some no longer manufactured system spare parts. In case of failure, should no more lasting stocks be available, the failing component cannot be straightforwardly replaced, so that either the whole system is to be removed and replaced with another one having closeoperation and performance, or the whole design is to be modified.

R&D can help utilities to adapt the safety justification (e.g. providing the software with qualification). Furthermore, a common approach should be developed either to create obsolescence-resistant technologies, possibly with support from other industries, or to adapt nuclear procedures to even faster evolving domains, vendors using more and more off-the-shelf technologies and components.

Presently, it is mandatory notably for TSO searching for a clear understanding of the environmental parameters which influence I&C ageing, establishing branch technical positions - including both architectural and technological aspects - and anticipating the new challenges and the follow-up of technology changes and improvements.

The wider adoption of digital I&C in NPPs will require safety research, too. Very demanding reliability requirements have to be applied to the safety critical software embedded in such digital systems. This is to drive important research efforts both on the side of developers and the organisations supporting the Regulator.

Among the R&D priorities, it is worldwide acknowledged that the following two ones are prime urgency:

- the definition of requirements, methodologies and finally the assessment of the risks associated to the implementation of new technological solutions;
- the development of test methods and devices to predict ageing phenomena and counter obsoleteness (e.g. the adoption of programmable digital automation).

Moreover, in the mid-term, it is also necessary to develop suitable test methods to predict the ageing effects on I&C components, and to search for an extended harmonization of the I&C component ageing knowledge.

The ageing effects need to be analyzed under the must unfavorable harsh environment accident conditions. Electrical Instrumentation and Control systems have crucial importance for measurement of safety related parameters and identification of NPP status (incl. degradation level) during progression of an accident. The correct measurement of safety related parameters helps to initiate adequate accident management and emergency response measures. Therefore, the I&C qualification for accident and post-accident harsh environmental conditions (e.g. vibration, temperature, pressure, jet impingement, radiation, humidity) is an issue of high importance. A complex systematic analysis and multi-disciplinary efforts are required in order to implement the consistent equipment qualification programme, which includes such steps as:

- identification of initial data for equipment qualification:
  - □ identification of accident scenarios requiring qualification;
  - □ identification of accident and postaccident time frame for safety function performance:
  - □ identification of equipment requiring qualification;
  - □ Identification of harsh environment conditions.
- establishment of requirements methods for equipment qualification taking into account harsh environment conditions and equipment ageing effects.

The Fukushima Dai-ichi NPP accident underlined the lack of reliable measurements characterising NPP state. The harmonised international efforts in above described activities would contribute to solution of this point.

#### 2.8

#### Internal and external loads and malicious acts

In recent years, new threats are forcing to focus not only on internal hazards, but also on the destructive action of external agents, including major earth-crust events, such as the earthquakes and the tsunamis, severe weather conditions, explosions, blasts and fire, external dynamic loads and malevolent acts.

A challenging prerequisite for any effective protection against external hazard is to accurately assess them by adopting deterministic, probabilistic or combined methodologies, relying on suitable collections of recorded observations. Accordingly, the reliability of the assessment highly depends on the quality and exhaustiveness of the available data. The environment around a NPP is continuously changing for climatic and morphological reasons, so that, even if perspective studies provide valuable information, nevertheless they are affected by ever large increasing uncertainties, when focusing on the extreme events - and their possible combinations - which have to be accounted for safety purposes.

Moreover, either a deterministic combination of these phenomena-whenever a common mode origin can be postulated - or a probabilistic combination of them - in any general case - are to be accounted for and accurately addressed to assess the plant safety, as well as the system and component reliability.

Accordingly, when assessing the safe behaviour of a system, a component or equipment, its robustness and resistance to all kinds of internal and external aggressions is to be demonstrated, adopting the penalizing combination of loads mentioned above here.

The knowledge gained through the studies and the feedback from NPP operation allows continuous updating of the uncertainty appraisal methods, which, in turn, drive evolution of the safety requirements issued

by the regulatory bodies, the design and construction rules adopted by the vendors, as well as the safety assessment methodologies.

All these items are the potential engines for aggression-oriented extended investigation and short-term and middle-term R&D addressing, in first priority:

#### 2.8.1

#### THE INTEGRITY OF EQUIPMENT AND **STRUCTURES**

The integrity of equipment and structures includes the civil engineering works and the mechanical equipment through:

- the improvement of the knowledge and understanding of phenomena governing the degradation modes of equipment and structures to master and anticipate their behaviour and to verify the suitability of maintenance programs and in-service inspections, and put them in actuation;
- the development of suitable methodologies to assess the structure and components status (detection devices for non destructive examinations of metallic components, concrete cracking detection, in service inspection);
- the improvement of existing methodologies and the development of new techniques for repair, and replacement of components;
- the accurate appraisal of the consequences of the industrial obsolescence which affect the operability of systems, "as originally designed".

Moreover, some electricity-infrastructure related problems - which are not specific to the nuclear industry - but can engender risks to the current and future NPP safe and secure operation through the loss of external energy supply source and the cold ultimate heat sink, in any circumstance, should be addressed. An alternative solution could be provided by developing new and advanced cooling technologies and by expanding high-voltage transmission infrastructure, as well.

The external aggressions include all the

loads that the environment can charge on the plant, as the consequence of earthquake, tsunami, flooding, blast, explosion and their combination, which can engender huge consequences on the plant operability and lead to severe accidents.

Their cumulated effects include:

- the destruction of buildings and access ways;
- the debris build-up;
- the loss of electrical power supply;
- the loss of cooling capacity of the reactor core and the fuel pools.

The R&D should mainly address the capacity of recovering the system in convenient time, improving the operator reactivity preparedness through specific training.

The Fukushima Dai-ichi NPP accident, the outcome of which is investigated within a specific Task groups in the SNETP are likely to provide input for definition, ranking and conduction of R&D actions.

#### 2.8.2

#### MALICIOUS ACTS

There are many possibilities to achieve malicious actions. Some rely on internal aggressions, some other to external aggressions.

Concerning the so called "internal threat", should be addressed the consequences of malicious actions of an internal adversary working regularly in a facility and thus having a good knowledge of its operation. Different scenarios are possible depending on the motivations of such an adversary. R&D work could focus on the methodological aspects to assess the various scenarios and its consequences and propose means to limit the consequences. Another field of investigation relates to computer security. This is to assess computer security vulnerabilities that could be used to commit a malicious act.

Concerning the so called "external threat", the protection of the NPP's main and fuel building against acts of malevolence mainly addresses the external barrier resistance to external aggressions. Nevertheless, the likely consequences on the other barriers - actually the second intermediate one - must not to be neglected, to avoid any consequence of debris impact and spreading-off.

The activity should address not only materials, but also structures, such as the containment and the supports, such as the tendons.

Moreover, the fire propagation issues being quite generic and they can be addressed separately. The explosion, blast, plane and missile crashes are the main items for R&D.

As far as the crash is concerned, activity should be focused on the simulation of the impact, both theoretically, through appropriate simulation, and experimentally, through ad hoc experiments. The nature and features of projectiles generating the crash, the sensitivity of the dome to the incidence angle of projectile and the motion of the fuel and the propagation of fire have to be considered as well.

#### 2.8.3

#### **FIRE AND SMOKE**

As far as the fire and smoke are concerned, four major investigation fields are identified:

- fire growth and propagation: the evaluation and the quantification of the sensitivity of a material-in-fire pyrolysis to the oxygen concentration in the buildings is a major safety concern for NPPs. The R&D in the field should be aimed at developing predictive methodologies enabling and supporting the simulation of the fire propagation in a confined and vented environment, including extreme conditions with quite low oxygen concentration. Moreover, the relationship among the pyrolysis of the fuel, the production of soot and the radiation heat transfer should be investigated in the aim at obtaining useful indication for the fire detection, too:
- make and heat propagation: due to the particular arrangement and layout of the

different buildings within nuclear installations, the early detection of fire relies also on the mastering of the smoke propagation modes. The R&D should contribute to a better understanding of turbulent flow in a thermal stratified media and provide elements for CFD computation tool validation. It should address the mechanisms of the stratification and propagation of smoke relying on both theoretical (simulation) and small-scale experiments. The activity goal would be the development of large-scale integral simulation models in order to better appreciate safety feature;

- fire sectorisation elements: the passive elements of sectorisation (walls, fire doors, dampers, openings) are the most important safety barriers regarding fire propagation assessment. Concerning the behaviour of these elements, past R&D programs³ gave partial information about thermal stress or overpressure. A lack of knowledge has been identified about the consequence of cumulative effects of a fire (mainly thermal stress, overpressure, humidity) and R&D could be carried out:
- equipment vulnerability: the vulnerability of the equipment to fire is a major challenge for a safe and secure operation of NPPs. The assessment of equipment resistance to fire should be carried-out both through definition of codes and standards and specific product oriented R&D;
- moreover specific effort should be devoted to a generalized improvement of the computational capacity and to modelling, in particular the generation of soot and unburned materials in a confined environment, the mechanisms of heat transfer through radiation, and turbulence;
- fire detection: an early detection of the fire is a crucial issue in the defence of equipments, materials and buildings. R&D should contribute to the development of a suitable methodology to detect fire through a non-intrusive investigation of the flame diffusion under the effect of gravity and the pyrolysis of materials.

#### 2.8.4

#### CONTAINMENT

To mitigate radiological consequences, any potential release due to over pressure of the containment has to be addressed considering the species (mainly gaseous).

After the accident of Fukushima Dai-ichi NPP, further development of passive systems (HE filters, sand filters, etc.) and design means could be done to improve their efficiency. For that purpose, a lack of knowledge concerning the efficiency of iodine purification using activated carbon with humidity (high loss of efficiency when relative humidity increases) has been identified.

#### 2.8.5

## CAPABILITIES OF TSO FOR MEASUREMENTS AND TEST FACILITIES

Experimental data are needed in developping and verifying numerical methods and models or simply to test behaviour of structures (crash test or blast load).

TSOs have developed capabilities for measurements and test facilities (for instance: impact test facility of IRSN-France and other facilities for fire, containment tests).

The complete characterisation of the TSOs means (experimental setups and simulation tools) is under implementation.

#### 2.9

### Severe accident, phenomenology, prevention and management

Despite the highly efficient accident prevention measures adopted for the current NPPs and the still more demanding ones for the advanced and innovative concepts, some accident scenarios, in low probability circumstances, may result in severe

3 >
Programme on
• fire propagation
(PRISME & PRISME 2)

fire sectorisation behaviour
 (PICSEL)

accidents, which can eventually end-out in core melting, plant damage and dispersal of radioactive materials out of the plant containment, thus threatening the public health and the environment.

The main issues in severe accident R&D are:

- the release of radioactive materials from a degrading core, addressing mainly gaseous fission products and aerosol chemistry, and their migration modes within the core:
- the chemistry of the fission products in the primary circuit and in the containment;
- the hydrogen production, accumulation and deflagration and the measures to counteract;
- the damaged core behaviour and stabilization, including the formation of the corium (molten core);
- the corium in-vessel retention and exvessel coolability;
- the fuel damage in the spent fuel pool (SFP) and the associated accident progression and its consequences, in particular on the source term;
- the long-term behaviour of fission products in the containment and the measures for mitigation of the source term.

R&D programs could provide the opportunity to resolve several pending issues which have

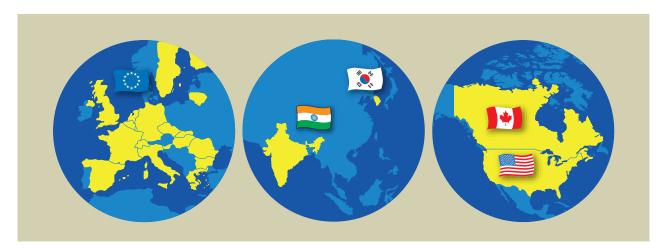
been under discussion for very long time without actually reaching a consensus. New experiments could supply evidence to help coming as near as possible to close these issues.

Launched under the FP6, the SARNET (Severe Accident Research Network of Excellence) network (www.sar-net.eu) has been continuing to work under the FP7 for April 2009 (figure 6). Today, it gathers 43 partners from 22 countries from Europe, Canada, Korea, United States, and recently India, representing diverse types of organisations (safety authorities, TSOs, research organisations, utilities, vendors, universities). Its objective is to enhance the coordination of the national efforts to optimize the use of the available expertise and resources, in particular the experimental facilities, to address and solve the severe accident issues for enhancing the safety of GEN 2 and GEN 3 NPPs.

In particular, R&D efforts must go on in order to make the consequences of severe accidents, in particular the source term, for GEN 2 existing NPPs closer to the ones evaluated for new GEN 3 ones. New prevention or mitigation devices should be investigated.

A group in SARNET has succeeded in 2008 ranking priorities on severe accident R&D for further investigation (other items being discarded or postponed). The R&D activities proposed in the severe accident R&D widely rely on this prioritization work and should be carried-out in close contact and agreement

Figure 6: SARNET network participating countries.



with SARNET. These issues were: corium/debris coolability, Molten-Core-Concrete Interaction (MCCI) and ex-vessel corium coolability by top flooding, melt relocation into water and ex-vessel Fuel-Coolant-Interaction (FCI), hydrogen mixing and combustion in containment (flame acceleration), oxidising impact on source term (Ruthenium oxidising conditions or air ingress for high burn-up and MOX fuel elements), and iodine chemistry in circuits and in containment.

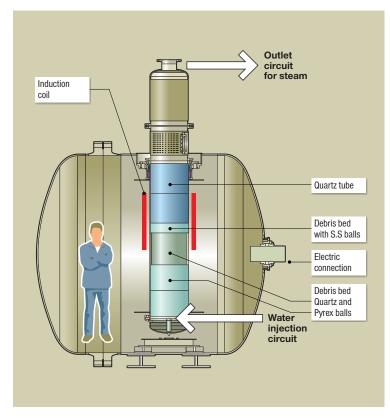
Among the main R&D issues in the current program, the following main two ones are to be considered of prime importance.

In-vessel and ex-vessel damaged core coolability

The coolability of the core and the spent fuel pool after a severe accident includes several fields of endeavour, among which it is worth mentioning:

- □ the damage core coolability during the reflooding phase (figure 7);
- □ the in-vessel coolability of debris and corium within the vessel lower head;

Figure 7: Scheme of the PEARL facility (IRSN) for debris bed reflooding.



- the ex-vessel Fuel-Coolant Interaction (FCI), also in relation with the international OECD/CSNI SERENA2 project;
- □ the Molten Corium Concrete Interaction (MCCI) and the coolability of debris and corium on the cavity, with water on the top of molten corium, in relation with the OECD/CSNI current and future projects;
- □ the fuel coolability in the reactor spent fuel pool.
- lodine and ruthenium chemistry in the Reactor Coolant System and the containment

The objective is to reduce the remaining uncertainties on the evolution of fission product reactivity in both gas and heterogeneous phases in the Reactor Coolant System and in the containment at mid and long term periods of the accident (delayed releases). The most important elements to be considered are iodine and ruthenium.

Plant ageing should also be accounted for source term evaluation models. This whole increasing knowledge will also be used to investigate complementary mitigation measures, natural or engineered (such as filtration, interaction between fission products and containment wall paints), so as to minimize releases to the environment. This R&D should be done in close relation with the OECD/CSNI new STEM project, coordinated by IRSN.

The research in this area of severe accidents will consist in experimental programmes and elaboration of physical models. In order to allow predicting the severe accident progression and consequences in NPPs, these models must be implemented in computer codes. In priority, the efforts should continue to capitalize this knowledge in the ASTEC integral computer code, jointly developed by IRSN and GRS, and considered now as the reference European severe accident code, in particular for TSOs, and to continue its assessment and validation (figure 8).

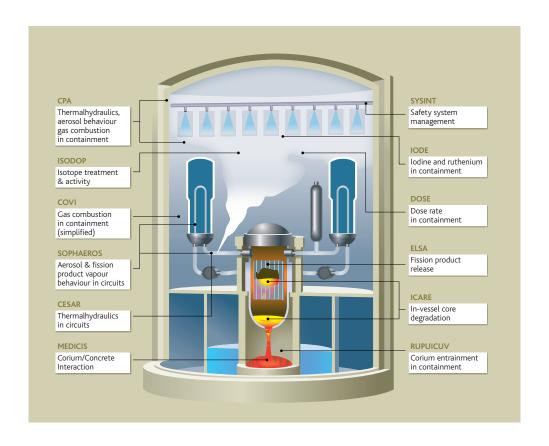


Figure 8: ASTEC integral code for simulation of severe accidents.

It is very likely that the Fukushima Dai-ichi NPP accident will imply some reorientations of the planned R&D but it is too early to get a clear view on that aspect. Nevertheless it can be already observed that the phenomena (on core cooling and degradation) that conditioned the evolution of the accident belong to the highest-priority issues that were selected in the SARNET work programme. But it seems likely that the aspects of mitigation of the severe accident consequences will have to be emphasized in the near future, as well as the behaviour of spent fuel pools.

#### 2.10

# Emergency preparedness, crisis management, short terms consequences

The emergency situations can originate from both internal and external factors and loads – including malevolent and malicious

acts - during the whole life-time of nuclear infrastructures - nuclear reactors, but also fuel storage pools, fabrication and reprocessing plants, transportation and storage devices - in operation and at shut-down.

Emergency is to be managed in any circumstance of nuclear infrastructures to minimize its impact on the environment and the population, and, should any major contamination occur, prevent the population from being irradiated and conduct the prevention, confinement, rescue and evacuation plans. Once the immediate urgency ended, the emergency plans must also provide guidance to organize the systematic site inspection and recovery.

The main characteristics participating in the emergency management are:

- the plant, the status of which is to be accurately monitored whenever possible, on-line to provide the operators with accurate information on its actual operating conditions and the likelihood of initiating event occurrence at the crisis notification;
- the internal and/or external loads, their

common mode origin and consequences and the likelihood and severity of the transients they can engender. All these items have to be accurately addressed in the prior-to-crisis preparation phase, the main outcomes of which have to be codified into guidelines to support the operator's conduct;

- the investigation and prediction tools, which must be accurate enough but, maybe more important, user-friendly enough to minimize the risk of misuse, confusion and misunderstanding, mainly in the very-early phase of the crisis. They can either include or rely on plant and full scale simulators, which allow operators to train on the management of actual situations:
- the operators, who have to be accurately selected and carefully trained, also - and mainly - addressing their preparedness and robustness to manage unforeseeable and fully unlikely situations which can certainly overload the human capacity of the crisis management team.

During the last 20 years, accounting also for the learning driven from TMI-2 and Chernobyl events, many significant progresses have been made in the crisis preparedness, which mainly resulted in delivery of operational procedures and tools for the early phase of emergency. As identified in the seminar on crisis management held by TSOs on the 7th of July 2011, reported at 2011 Eurosafe Forum, these procedures have to be more widely applied. The work already performed has now to be complemented and extended to the modelling of the longer term problems, also addressing the strategies for the long term rehabilitation of the contaminated areas. That could promote suitable R&D programs addressing, in the short term, the improvement of the modelling of the emergency management actions in agricultural and inhabited areas and the development of a framework for the sustainable rehabilitation of contaminated areas, as well.

In the mid term, the R&D should address the search for suitable improvements of the systems and tools to support the operator's decision and conduct in case of new, unforeseen - and unforeseeable - threats.

The recent event in Japan have demonstrated that a country alone cannot face a widescale disaster: in this case it would be very worth relying on the possible support from other countries. Moreover, in the longer term, the high standard of emergency preparedness and management has to be kept even if resources might decrease with time. Therefore, it is essential to develop a European platform for decision support and emergency management including a scenario generator for the planning, performance and evaluation of exercises. This will strengthen the preparedness and planning as well as keep competence in this area. This objective should gather all the actors of the nuclear field, including the stakeholders and the civilians.

# 

For the sake of citizens and workers safety, and in compliance with their national missions, the safety organisations are developing their competences (education, research, and operation experience feedback and knowledge management) in order to provide the national authorities with a state of the art expertise. Their mission also involves the duty to worn about emerging risks and to provide the corresponding proposals for developing the necessary competences.

As EU NPP designers and operators are also committed to develop energy systems with the highest safety standards, a large part of existing and future research may be shared by all stakeholders and participants in the SNETP framework. The list of common research priorities, as developed in this position paper, is particularly significant.

Considering the quality of contribution and the resulting debate promoted through

SNETP, the safety organisations are happy to bring this contribution. Safety organisations clearly express the hope to develop synergies in order to enrich and realize the necessary research program in the safety field, to favour the technical debate and enlarge the corresponding scientific consensus domain, hence further concentrate on the remaining issues of the safety debate.

An important benefit is expected from the SNETP instances to exhibit well balanced priorities, even if the detailed interest is not the same for each single stakeholder.

Were specific safety organisations research programs not supported by some stakeholders, it would be useful to consider the potential cost of a badly managed severe accident induced by design weakness, operation error or severe external aggression: responsibilities of all stakeholders would be involved, including safety organisations.

# 4 REFERENCES

- The final report of SAFIR2010 is available from the link http://virtual.vtt.fi/virtual/safir2010/pdf/T2571.pdf
- Safety Management and Organisational Learning (MANOR) pp.27-35
- Operator Practices and Human-system Interfaces in Computer-based Control Stations (O'PRACTICE) pp. 87 - 108
- "Some outcomes of the SARNET network on severe accidents at mid-term of the FP7 project", J.P. Van Dorsselaere et al., ICAPP'11 Conference, 3-7 May 2011, Nice (France)
- "The ASTEC integral code for severe accident simulation", J.P. Van Dorsselaere et al., Nuclear Technology, 165, pp.293-307 (2009)
- SARNET2 Public web page: http://sar-net.eu
- SNETP Public web page: www.snetp.eu



Nowadays, the main objective to further limit the spreading of radioactive products in the environment strongly depends on the capacity to establish long term duration core cooling in all the Dai-ichi units, i.e. to reach a reliable close circuit cooling. Whereas the Fukushima Dai-ichi NPP accident cannot yet be considered as completely under control, it is clear that years will be necessary to clean the reactors in order to be able to measure the degradation status suitable to validate the severe accident computer codes.

However, several aspects of the Fukushima Dai-ichi NPP accident raised questions on the capacity to avoid and to manage nuclear accidents. In order to identify general orientations to improve nuclear safety without delay, the following aspects can already be commented:

- 1 General characteristics of the accident
- 2 Overall comparison with other accidents

- 3 Lessons drawn on the crisis management
- 4 Consequences on the safety expertise methodologies
- 5 Potential output from the stress
- 6 Consequences on the research programmes

This short paper proposes a review of the main orientations that can be enhanced as a first statement of the consequences of the Fukushima Dai-ichi NPP accident.

#### General characteristics of the accident

The Fukushima Dai-ichi NPP accident starts from a largely beyond design basis external aggression. Despite the highest experience and expertise of Japanese organisations to cope with earthquakes, the 11 March 2011 aggression's level is such that it appeals several lines of defence and destroys some of them simultaneously: main and back up electrical power supply, main and mean term coolant source.

This characteristic of the accident calls for:

- extending the defence-in-depth safety approach to any type of aggression that may arise, in particular the external aggressions;
- considering the defence against beyond design basis accidents in any case of aggression;
- and developing more independence between the different lines of defence with respect to beyond aggressions.

## **Overall comparison**

Considering the three NPP main accidents, it appears that the main initiating events were different:

with other accidents

- in the TMI2 accident, the poor reliability of the primary circuit relief valves cumulated with bad OEF system (similar initiating event on another unit in USA). overwhelming man-machine interface and indirect in-core water level measurement in the primary circuit leading to wrong analysis of the core cooling status;
- in the Chernobyl accident, the incoherent human organisation resulted in nonrespect of the normal operation domain, which gave way in a strong reactivity insertion accident. On top of that, it appears that the design of these reactors favours the reactivity build-up, the I&C system does not preclude operating out

of the normal operation domain, and the containment is particularly weak to prevent the radioactive products to spread around;

in the Fukushima Dai-ichi NPP case, beyond the defence-in-depth against external aggressions and independence among the lines of defence, it can be considered that the plant ageing intervened to weaken the capacity to resist in the sense that, despite a good structural resistance to earthquakes, the reactor construction did not evolve sufficiently to raise the plant safety level up to the best GEN 3 designs.

This diversity of initiating events calls for a further review of what could be the other very high intensity aggressions that could smash the reactor safety defence lines.

The cumulative effect of design weaknesses on the capacity to preserve safety further enhance the need to separate the defence lines and to consider the response of the systems to any beyond design basis accident.

## Lessons drawn on the crisis management

The main difficulties in managing the Fukushima Dai-ichi NPP accident resulted from:

- concomitant effects of the event, both due to the management of the population affected by the tsunami and due to the management of the reactors;
- simultaneous accident for four units and their pools, the status of each one unit making more difficult to care for the neighbouring units;
- no rapid backup supply available for the electrical energy and for the coolant.

The involvement of four NPP units certainly overloaded the human capacity of the crisis management team and crisis management means; an experience feedback has to be performed on that specific point. Another line of defence in such a case is to take opportunity of possible support from other countries. In order to be effective, this foreign support must not disturb the national organisation but should be driven carefully into the national organisation. This capacity in supporting implies the crisis management to be sufficiently harmonized among the different countries in terms of organisation, technical communication, management methodology and scientific calculation tools.

Many of the competent nuclear safety experts were and are still keen to help the Japanese organisations to cope with these several simultaneous accidents. A possible way to do so includes these experts to undergo real training on existing plants and to share the crisis management methodology. Beyond preferable involvement in national crisis organisation, some help in the plant status diagnostic, prognostic and management strategy could be brought through long distance communication means only if the crisis organisation is prepared to do so. A necessary preparation would be to develop multinational crisis management exercises.

The conditions of the accident are such that crisis management means would preferably be independent for each unit and independent from the local conditions. This characteristic also underlines the usefulness of technical backup means available permanently not too far from the existing NPPs. Modifications in the reactor design could be necessary in order to enable feeding the primary circuit of NPPs through additional nozzles in the main tank.

Use of more realistic tools is possible to manage the accident as soon as it lasts long enough to study several recovery strategies.

ETSON has decided to launch a process of harmonisation of crisis management

for on how to share technical information, to manage dose check-up of emergency staff and population, environment monitoring and cartography, decontamination methodologies and radioactive waste management.

#### Consequences on the safety expertise methodologies

The different orientation through which the safety expertise may be improved includes the following provisions:

- agree at an international level about the high level safety objectives to be respected; for example generalize the WENRA objectives application;
- examine the different possible design at a multinational level up to delivery of a common design certification; i.e. decide an extension of the MDEP process; in practical terms, it necessitates processes to perform the licensing with groups of regulatory bodies and technical safety organisations;
- adopt a common policy to propose on the international market only such certified designs, while protecting the intellectual property of the reactor designs;
- generalize the stress test definition to be applied to all NPPs in the world;
- harmonize the methodologies through which the NPP's safety level is evaluated;
- elaborate common standards for competence development and staffing amplitude in the safety domain for NPPs;
- on the NSS (Nuclear Safety Standards) side, it appears necessary to extend the defencein-depth application to any aggression source, adding the beyond design basis behaviour in order to mitigate the effects.

#### **Potential output** from the stress tests

In order to check the safety level of existing plants, stress tests have been defined with the participation of all the stakeholders in the nuclear industry. In this field, we consider the WENRA proposal to be the reference definition for such tests.

Those tests are suitable to exhibit weaknesses in different technical and human capacities domains when NPPs are submitted to conditions like earthquake resistance, behaviour in case of extreme flooding, loss of electrical power and loss of heat sink.

Several lessons could be brought to develop new policies in the safety policy and in the safety research.

### Consequences on the research programs

On many of the safety issues, the design of NPPs aims at demonstrating the existence of safety margins with respect to reference scenarios. One of the consequences of the Fukushima Dai-ichi NPP accident is the requirement to be able to know the best estimate behaviour of the plant systems up to beyond design basis accidents. Also, it is worthwhile to know the behaviour of the system is continuous with respect to the intensity of stress applied, and when the behaviour is becoming non-linear or catastrophic. This may call for a large extension of the physical modelling and computer tools development in different areas.



#### MAILING ADDRESS

IRSN BP 17 92262 Fontenay-aux-Roses Cedex France

#### **HEAD OFFICE**

31, av. de la Division Leclerc 92260 Fontenay-aux-Roses Phone +33 (0)1 58 35 88 88 Fax: +33 (0)1 58 35 84 51

Association n° W921001929