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Research on Severe Accidents in Nuclear Power Plants

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1. Introduction

Severe accidents (SA) in nuclear power plants (NPPs) are unlikely events but with serious consequences, as recently shown by the accident that occurred in April 2011 in the Fukushima Japanese NPPs. SA research started originally in the seventies with initial risk assessment studies and later on with experimental programs, development of numerical simulation codes, and Level 2 Probabilistic Safety Assessments (PSA2). A huge amount of research and development (R&D) was performed in the last thirty years in the international frame. This was pushed forward by the two core meltdown accidents that occurred: first in the Unit N°2 of the Three Mile Island (TMI-2) Pressurized Water Reactor (PWR) near Harrisburg (Pennsylvania, USA) on March 28, 1979; then in the Chernobyl RBMK (Water-cooled channel-type reactors with graphite as moderator, designed by Soviet Union) reactor in Ukraine. Large progress has been reached in recent years on the understanding of SA but several issues still need research activities to reduce uncertainties and consolidate the accident management plans.

Along with the progress of understanding and the limited amount of the national budgets on SA R&D, the high complexity of the physical phenomena and the high cost of experiments made necessary to better rank the R&D needs. In 2004 the European Commission judged necessary to better coordinate the national efforts to optimise the use of the available expertise and the experimental facilities in order to resolve the remaining issues for enhancing the safety of existing and future NPPs. This led to launch SARNET (Severe Accident Research NETwork of Excellence) (Albiol et al., 2008; Micaelli et al., 2005), in the framework of the 6th Framework Programme (FP6) of the European Commission, gathering most worldwide actors on R&D SA. One of the main outcomes was the identification of the highest priority SA issues still to be solved. A second phase of the network (SARNET2 project) has started in April 2009, again supported by EC in the FP7 for four years, again coordinated by IRSN.

Section 2 describes shortly what a severe accident is (most of the material described in this section is issued from the reference IRSN-CEA, 2007). Section 3 presents the general approach on SA R&D. Section 4 explains in details the approach that was adopted in SARNET to rank the R&D priorities. Section 5 describes the current SARNET2 FP7 project and the common research programmes, and finally Section 6 focuses, for the sake of illustration, on the important issue of coolability of a degraded core during reflooding.

2. Severe accidents in nuclear power plants

2.1 Case of present nuclear power plants

The “severe accident” refers to an event with an extremely low probability of occurrence (such as 10^{-5} per reactor per year¹ for internal events), thanks to the preventive measures implemented by NPP operators, but causing significant damage to the reactor core, with more or less complete core meltdown and finally possible serious consequences in case of release of radioactive products into the environment.

SAs are generally caused by a cooling failure within the reactor cooling system (RCS), which prevents proper evacuation of residual power from the core, and by multiple dysfunctions, arising from equipment and/or human error, including the failure of safety procedures. A series of complex phenomena then occur, according to various scenarios and depending on the initial conditions of the accident and on the operator actions. For the purpose of this document, “early releases” are those liable to occur before all the measures aiming at protecting the general public can be implemented. Figure 1 schematically presents the major physical phenomena that may occur during a SA, as well as a few safety systems involved.

If the reactor core remains uncovered by water for an extended period of time (typically a few hours), nuclear fuel progressively overheats due to residual power. Steam initiates an exothermic oxidation of zircaloy fuel cladding, resulting in substantial production of hydrogen and thermal power. Additionally, chemical reactions between fuel and its cladding produce low-melting-point eutectics, resulting in relocation of molten materials (called “corium”) in the core. The fuel first releases the most volatile fission products, then the semi-volatile products.

Progressively, a corium pool forms in the core and progresses towards the lower head of the vessel. When it reaches water remaining there, water is vaporized, corium is fragmented and forms a debris bed. During core degradation, standby supplies of water can be delivered to the RCS or the secondary cooling system. Reflooding a degraded core is a complex phenomenon which may enable the accident progression to be slowed down or halted under certain conditions. In contrast, reflooding may also increase hydrogen production and cause further release of fission products.

Corium melts and debris accumulate in the vessel lower head and may cause its rupture, either by thermal erosion, creep or plastic failure, depending on pressure conditions in the RCS. After vessel rupture in case of high-pressure conditions within the vessel, part of the ejected corium is fragmented and may be dispersed into the containment. This may provoke a pressure spike, resulting in substantial heat exchange with the air, oxidation of the corium metallic components and, in some cases, simultaneous combustion of the hydrogen present in the containment building. This phenomenon is called “direct containment heating” (DCH).

Following vessel rupture, corium slumps and accumulates in the reactor pit. A corium-water interaction (called Fuel-Coolant Interaction or FCI) may occur if the pit contains some water, which may be followed by a more violent phenomenon called steam explosion. This explosion may create projectiles that could threaten the leak-tightness of the containment buildings. Without initial presence of water in the pit, corium will thermally erode the concrete basemat, which could cause the loss of containment: this is the Molten-Corium-Concrete-Interaction or MCCI. During this phase, a substantial quantity of incondensable gas (H_2 , CO, CO_2) is released, resulting in a progressive increase in pressure within the containment building. To avoid a potential break in this structure, a ventilation-filtration

¹ This figure could be updated in the future, following the recent Fukushima accident.

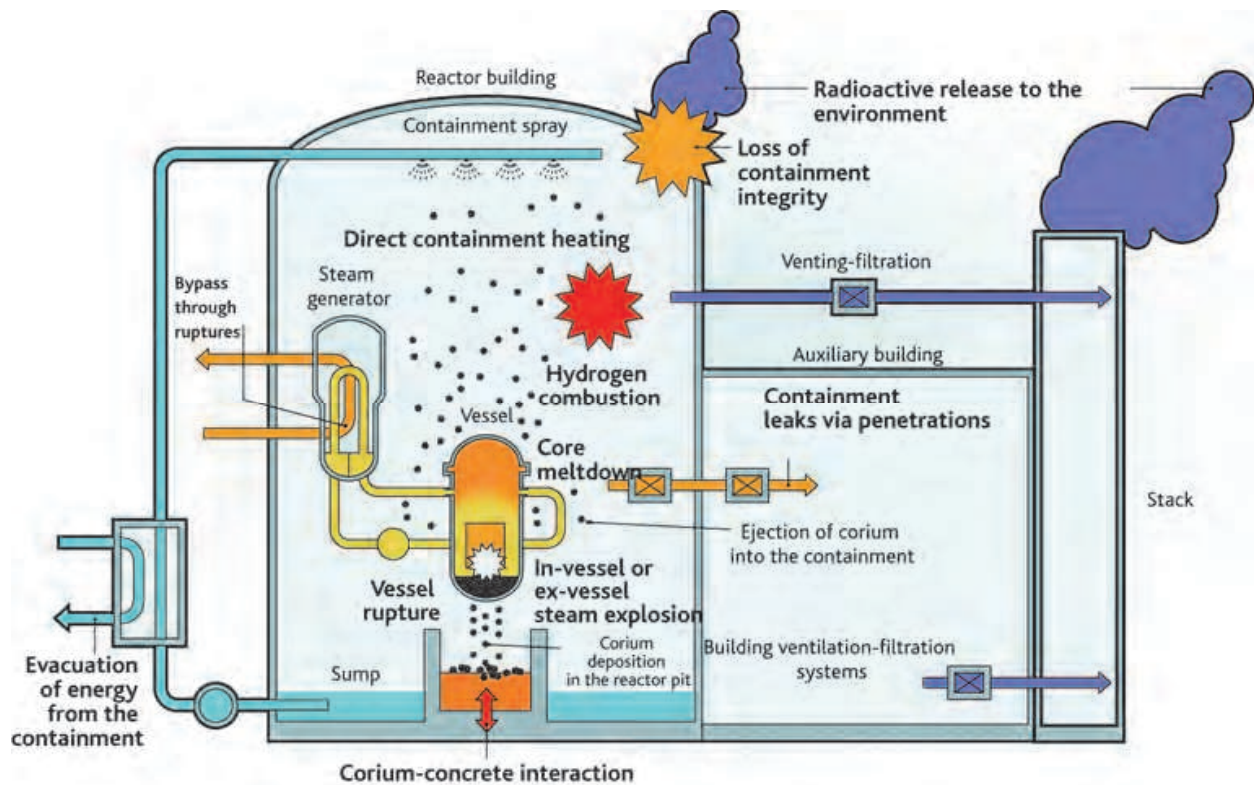


Fig. 1. Main physical phenomena during a severe accident

system has been installed in all Light Water Reactors (LWR): it can be activated in general 24 hours after an accident begins, if the containment heat removal system fails.

Hydrogen produced by core degradation is released into the containment, where it burns on contact with oxygen, provoking a pressure and temperature spike which may damage the containment building. This combustion can either be slow acting (slow deflagration) or more rapid (rapid deflagration) and, in some cases, explosive (detonation). Hydrogen combustion may lead to the loss of the containment barrier: a commitment to making this risk residual has been demonstrated by the progressive implementation of hydrogen Passive Autocatalytic Recombiners (PAR) in many NPPs.

For all modes of containment rupture, the release of fission products into the environment depends on the conditions affecting their transfer within the reactor. The transfer of fission products depends primarily on their physical and chemical properties, i.e. whether they are gases or aerosols and their chemical form. Iodine and ruthenium behaviour requires particular attention, given their complexity and their significant short-term radiological impact. Regarding longer-term accident consequences, particular attention must be paid to caesium releases.

In the event of a SA, operating personnel are called upon to follow the recommendations in the Severe Accident Management Guidelines (SAMG). Actions recommended in the SAMG serve primarily to maintain containment, aiming to:

- Avoid or minimise airborne radioactive releases outside the containment building,
- Provide sufficient time before potential containment loss to allow implementation of the public protection measures described in emergency plans.

2.2 Case of future nuclear power plants

For all new NPPs of any type under construction or planned today, named “Generation III” (noted Gen.III in the following), the provisions aim to significantly enhance accident prevention and the SAs are addressed from the design phase.

For the EPR (European Pressurized Reactor), the technical directives specified that:

- Core meltdown accidents, particularly under pressurised conditions, postulated to cause large early releases must be “practically eliminated”. While such accidents remain physically possible, design measures must be implemented to prevent them. For instance a dedicated pressurisation valve, coupled with an isolation valve, was integrated in the RCS, in addition to the standard safety mechanisms protecting this system from overpressure. PARs have also been installed in the containment.
- Low-pressure core meltdown sequences must proceed in such a way that the maximum conceivable releases only require measures very limited in duration and scope to protect the public. Thus a system was implemented to collect corium and stabilise it on the long-term: this “core-catcher” is built in the containment building and linked to the reactor pit. Besides, the containment has a double concrete wall, with filtration, to increase the containment tightness with respect to radioactive release.

In other Gen.III NPPs, different designs have been elaborated for stopping corium progression or limiting its consequences. Some NPPs aim at maintaining corium within the vessel (In-Vessel-Retention or IVR) by cooling the external surface of the vessel lower head through water injection into the pit. Others have designed core-catchers differently from the EPR one that is based on corium spreading. Advanced VVERs² adopt a core-catcher underneath the vessel (like the one at Tian Wan in China or being built in Belene in Bulgaria): this core-catcher makes use of sacrificial materials consisting mainly of steel, iron oxide ceramic and alumina.

3. Research and development on severe accidents

3.1 General approach

The general approach for SA R&D (Figure 2) is based on one hand on experiments and on the other hand on computer codes for simulation of physical phenomena.

The SA R&D presents some specific features that imply very high costs:

- Complexity of the physical phenomena,
- High number of phenomena, with the need of considering them together due to their mutual interactions (“coupling”),
- Extreme conditions: very high temperatures (above 3000°C), high pressure (up to 200 bars), irradiation effects,
- Need of tests with real materials (importance of the “material effect”),
- Difficulty to extrapolate to the reactor scale.

It has involved in the past very substantial human and financial resources as well as collaboration between nuclear stakeholders, industry groups, research centres and safety authorities, at both the national and international levels. The international programmes concerned mainly the Framework Programmes of Research and Development of the European Commission (see <http://cordis.europa.eu>) and the projects conducted under the

² VVER: water-cooled water-moderated power reactors (PWR type) that were developed in the ex-Soviet Union.

auspices of the CSNI (Committee on the Safety of Nuclear Installations) of NEA (Nuclear Energy Agency) in the OECD (Organisation for Economic Cooperation and Development) (CSNI, 2000; OECD, 2007).

The research in this area thus aims to further understand the physical phenomena and reduce the uncertainties on their quantification, with the ultimate goal of physical developing models that can be applied to reactors. These models, implemented in computer codes, allow predicting SA progression and consequences.

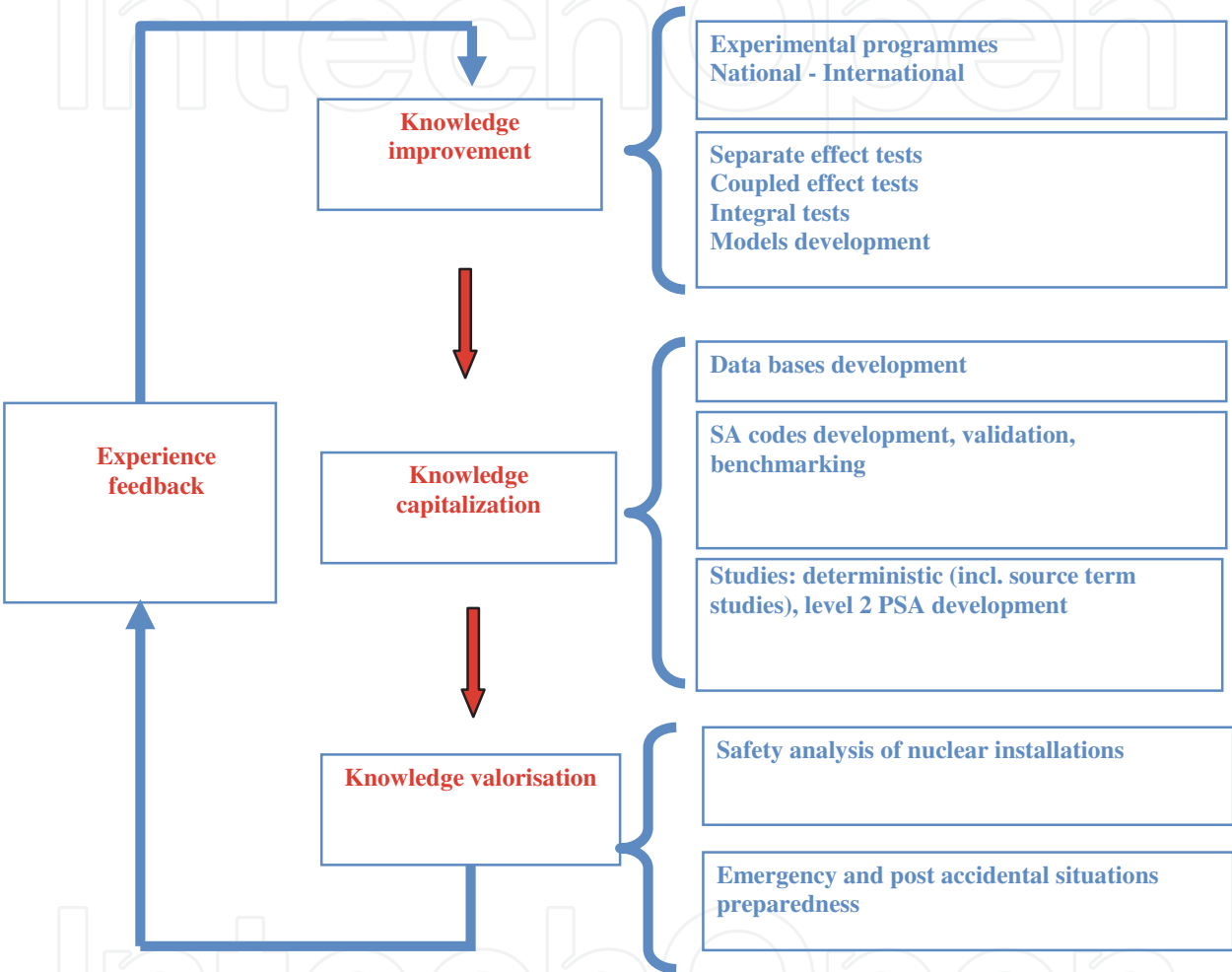


Fig. 2. General R&D approach

For Gen.III NPPs, this research allowed to design specific devices for SA prevention and for mitigation of consequences, as described in Section 2.2. However, for existing plants (called Generation II and noted Gen.II in the following), SAs were not a design consideration. Consequently, modifications of their design are limited and the research in this area is primarily aimed at limiting the potential impact of SAs. Specifically, there are two complementary research orientations: a) characterising releases and studying modes of containment failure, and b) developing methods to limit the consequences of the SA scenarios.

3.2 Experimental R&D programmes

Different categories of experiments are usually defined:

- *Separate effect tests (SET)* investigate a single phenomenon and yield data for development of a model which describes its effect and which is to be integrated as a sub-routine into a computer code. The corresponding facilities are typically single-purpose, small-scale channels, loops or vessels equipped with specialised, sophisticated, high-accuracy instrumentation.
- *Coupled effect tests (CET)* investigate the coupling of two or more phenomena previously explored in SETs, and provide data for the appropriate integration of the corresponding models into a code. The corresponding facilities are typically of small to intermediate-scale, using a test loop or a test vessel with comprehensive instrumentation adapted to the effects to be investigated.
- *Integral Experiments* represent all or part of a reactor accident sequence. They examine the interactions of several phenomena previously studied in SETs and/or CETs. The data obtained are needed for confirmatory validation of a code and its application, i.e. adequate problem set-up by the code user, correct and complete modelling of the relevant phenomena and their interactions within the code. The corresponding facilities are typically intermediate to large-scale models of full size bundle or containment, in the latter case with variable infrastructure for investigating many aspects of containment behaviour.

3.3 Development of computer codes for SA numerical simulation

3.3.1 Types of codes

Three classes of SA codes can be defined, depending on their scope of application: integral codes, detailed codes and dedicated codes.

- *Integral codes* (also called “system” codes or, in the past “engineering-level” codes): these codes simulate the overall NPP response, i.e. the response of the RCS, the containment, and the source term to the environment, using “integrated” models for a self-consistent thorough analysis of the accident. They include a well-balanced combination of phenomenological and user-defined parametric models for the simulation of the relevant phenomena. They must be (relatively) fast running to enable sufficient number of simulations of different scenarios to be performed, accompanied by parameter studies to address uncertainties: the computing time should be roughly around the accident real time. These codes are primarily not designed to perform Best-Estimate simulations, but rather to allow the user to bound important processes or phenomena by numerous user-defined parameters. Integral codes are usually used to support PSA2 analyses and for the development and validation of Severe Accident Management (SAM) programmes. In the last years, the rapid increase of the computer performance enabled more and more the replacement of parametric models by mechanistically based ones in the integral codes. The main internationally used codes are today ASTEC (see Section 5.6), jointly developed by IRSN and GRS (Van Dorsselaere et al., 2009), MAAP, developed by Fauske & Associates Inc. (USA), and MELCOR, developed by Sandia National Laboratories (USA).
- *Detailed codes* (also called mechanistic codes): they are characterised by best-estimate phenomenological models, consistent with the state of the art, to enable as far as possible an accurate simulation of the behaviour of a NPP in case of SA. In order to better illustrate the differences with the approach of integral codes, in most cases, a numerical solution is found for integral-differential equations while in integral codes

some correlations may be used. Basic requirements are that the modelling uncertainties are comparable with the uncertainties on the experimental data used to validate the code and that user-defined parameters are only necessary for phenomena that are not understood due to insufficient experimental data (including scaling problems). Since, as a basic principle, these codes should have as few as possible user options, existing uncertainties in the simulation of the different phenomena must be specified to enable the definition of the uncertainties of the key results. The main advantages of these codes are to give a more detailed insight into the progression of a SA and to design and optimise mitigation measures. They can also be used for benchmarking the integral codes. Due to the high computation time, they simulate only a part of the plant, e.g. RCS or containment. Their computation time depends on the scope of the application but it can span over days and weeks. The main internationally used codes are today: for the RCS behaviour and the core degradation ATHLET-CD (GRS), SCDAP/RELAP5 (INL in the USA), RELAP/SCDAPSIM (ISS in the USA) and ICARE/CATHARE (IRSN) and for containment CONTAIN (ANL in the USA) and COCOSYS (GRS).

- *Dedicated codes*: these codes that deal with a few phenomena have become important in context with the requirements of the regulatory authorities to take into account SAs in the design of new NPPs and to reduce uncertainties of risk-relevant phenomena. In general they have to be very complex with the drawback of large calculation time. Typical issues for which dedicated codes are required include: steam explosion and melt dispersal (e.g. MC3D at IRSN), structure mechanics (e.g. CAST3M at CEA in France, or ABAQUS in the USA). This family of codes includes the CFD (Computational Fluid Dynamics) codes that solve Navier-Stokes thermal-hydraulics equations in 3D geometry, such as GASFLOW in KIT (Germany), TONUS in IRSN, CFX as commercial tool, etc....

3.3.2 Process of code development and validation

The general process of code development is composed of the following steps, with possible iterations between them:

- Code requirements (scope of application, computing time, etc...),
- General specifications (structure, programming language, level of details of modelling, numerical schemes, etc...),
- Detailed specifications, possibly with prototyping to check a model or a numerical scheme,
- Physical model development,
- Implementation into a computer code,
- Code verification (tests on analytic solutions of equations, laws of conservation of mass, energy, and momentum, portability on diverse computers types, numerical coupling between models, etc..).

The code validation process aims at providing a sufficiently accurate representation of the reality of the SA phenomena. But this SA field presents some very peculiar features due to the continuous evolution of knowledge and to the extreme conditions that occur in a SA, notably the geometry scale that is difficult to achieve in laboratory experiments. The VASA project (Allelein et al., 2001) took place in the FP4 of the European Commission to analyse this SA validation process in details. Two stages can be defined:

- Comparing the code results with results of experimental programmes, which leads to define a “validation matrix”.
- Verifying the code capability to adequately simulate real SA scenarios at full-scale, which may be done through several types of work:
 - Benchmarking the code results of plant applications with results of other codes, either integral codes or detailed ones,
 - Applying the code to real plant SA scenarios, which is very scarce except for the TMI-2 and Chernobyl accidents (and in addition the Fukushima accident in the future when reliable data become available),
 - Performing uncertainty analyses in order to show the consistency and the reliability of code results, including the analysis of the influence of nodalisation and of numerical time-steps.

The CSNI International Standard Problems (ISP) provide a particularly valuable source of information for code validation: they are comparative exercises in which predictions of different computer codes for a given physical problem are compared with each other and with the results of a carefully controlled and well documented experiment. Over the last thirty years, forty-nine ISPs have been sponsored (CSNI, 2000).

The qualification of the code user is an important part of the code validation process. The user may have an impact on the quality of the SA analysis. It is considered essential that users have a good knowledge of the modelling inside the code and that the codes should not be run as “black boxes”.

4. SARNET approach on severe accident R&D ranking

Most of the material described in the Section 4 is issued from the SARNET reference (Schwinges et al., 2008).

4.1 Objectives and work scope of the SARP group

The EURSAFE thematic network (Magallon, 2005) yielded a list of 21 areas of needed research in the SA domain, which included recommendations for experimental programmes and code development. To further develop this list as a living document, the work package “Severe Accident Research Priorities” (SARP) was established in the SARNET FP6 project.

The activities in SARP focused on the identification of areas where the knowledge has been considerably improved and where further experimental research and/or model development seemed not to be of high priority. Furthermore it had to identify research areas which needed reorientation and, last but not least, the needed research not yet being covered. The outcome of the SARP work was an up-dated ranking, giving different priorities to the research issues“, and helping” (thus linking the last sentence to the previous one..) decision to perform the different research programmes.

The working scope was outlined as:

- Agree on methodology,
- Identify issues resulting from EURSAFE not appropriately covered and review them,
- Analyse R&D recent progress,
- Analyse results from PSA2 studies,
- Reassess the ranking of research issues and reorient the priorities,
- Review potential experimental and theoretical programmes to address these issues,
- Make recommendations for revision of R&D programmes.

4.2 EURSAFE methodology and results

The objectives of EURSAFE were to establish a large consensus on the SA issues, where large uncertainties still subsist, and to propose a structure to address these uncertainties by appropriate R&D programmes making the best use of the European resources. It incorporated issues related to existing plants (PWR, BWR³ and VVER), lifetime extension of these plants, evolutionary concepts (higher burn-up and mixed oxide –MOX– fuels), and safety and efficiency of future systems. Twenty partners representing R&D governmental institutions, regulatory bodies, nuclear industry, utilities and universities from nine European countries and Canada worked together in a network structure, which was supposed to be a starting point of SARNET.

To achieve the objectives, sufficient convergence on issues and phenomena and on their importance in terms of safety and knowledge was required among all actors. The final objective was a consensual approach to resolve the remaining uncertainties and open issues. Establishing Phenomena Identification and Ranking Tables (PIRT) has been proved in other areas (e.g. Loss of Coolant Accidents or LOCA) to be an efficient and unbiased way to reach such a consensus (NUREG, 2000). In EURSAFE the PIRT integrated all the SA issues from core degradation up to release of fission products from the containment, taking into account any possible counter-measures and the evolution of fuel management.

As a basis, a comprehensive list of 1016 SA phenomena was established. The phenomena were classified in five groups:

- In-vessel (162 phenomena),
- Ex-vessel (149 phenomena),
- Dynamic loading of the containment (461 phenomena),
- Long-term loading of the containment (116 phenomena),
- Fission products (128 phenomena).

Three safety-oriented groups of experts scrutinized these phenomena of the five lists and ranked them in accordance to their safety importance for primary circuit, containment and source term. Two evaluations were established: the safety importance ratio and the knowledge ratio. Starting with 1016 identified phenomena, the list was reduced to 239 items, important for safety, of which 106 were found with significant lack of knowledge.

After completion of the two ranking phases, this procedure clearly emphasized the phenomena being simultaneously highly important for safety and significantly lacking of knowledge. The remaining 106 phenomena were obviously candidates for further R&D work:

- 24 phenomena for In-vessel,
- 28 phenomena for Ex-vessel,
- 26 phenomena for Dynamic loading of the containment,
- 10 phenomena for Long-term loading of the containment,
- 18 phenomena for Fission products.

As a further step, the research needs and programmes to address each selected phenomenon of the PIRT list were identified and established in a list. According to the similarities in terms of research needs and physical processes, with the scope of being able to set up a limited number of coherent R&D programmes, several phenomena were merged into research issues without further elimination or selection. A rationale for these research needs

³ Boiling Water Reactors

was established based on safety relevance and lack of knowledge. The outcomes of this process are summarised in Table 1, which gives the 21 items of needed research (EURSAFE Research Issues or **ERI**) and corresponding rationales drawn from the 106 phenomena selected in the PIRT.

ERI	EURSAFE Research Issue	Rationale for selection
1,1	Hydrogen generation during reflooding or melt relocation into water	Rapid generation of hydrogen which may not be accommodated by recombiners and the risk of early containment failure. Improve knowledge about the magnitude of hydrogen generation.
1,2	Core coolability during re-flooding and thermal-hydraulics within particulate debris	Termination of the accident by re-flooding of the core while maintaining the integrity of the RCS. Increase predictability of core cooling during reflooding.
1,3	Corium coolability in lower head and external corium catcher device	Improve predictability of the thermal loading on Reactor Pressure Vessel (RPV) lower head or corium catcher devices to maintain their integrity.
1,4	Integrity of RPV due to external vessel cooling	Improve data base for critical heat flux and external cooling conditions to evaluate and design Accident Management (AM) strategies of external vessel cooling for in-vessel melt retention.
1,5	Integrity of RCS	Improve predictability of heat distribution in the RCS to quantify the risk of RCS failure and possible containment bypass.
1,6	Corium release following vessel failure	Improve predictability of mode and location of RPV failure to characterise the corium release into the containment.
2,1	MCCI: molten pool configuration and concrete ablation	Improve predictability of axial versus radial ablation up to late phase MCCI to determine basement material failure time and loss of containment integrity.
2,2	Ex-Vessel corium coolability, top flooding	Increase the knowledge of cooling mechanisms by top flooding the corium pool to demonstrate termination of accident progression and maintenance of containment integrity.
2,3	Ex-Vessel corium catcher: corium ceramics interaction and properties	Demonstrate the efficiency of specific corium catcher designs by improving the predictability of the corium interaction with corium catcher materials.
2,4	Ex-Vessel corium catcher: coolability and water bottom injection	Demonstrate the efficiency of water bottom injection to cool corium pool and its impact on containment pressurisation.

ERI	EURSAFE Research Issue	Rationale for selection
3,1	Melt relocation into water and particulate formation	Determine characteristics of jet fragmentation, debris bed formation and debris coolability towards maintenance of vessel and respectively containment integrity.
3,2	FCI incl. steam explosion: melt into water, in-vessel and ex-vessel	Increase the knowledge of parameters affecting steam explosion energetic during corium relocation into water and determine the risk of vessel or containment failure.
3,3	FCI incl. steam explosion in stratified situation	Investigate the risk of weakened vessel failure during reflooding of a molten pool in the lower head.
3,4	Containment atmosphere mixing and hydrogen combustion / detonation	Identify the risk of early containment failure due to hydrogen accumulation leading to deflagration or detonation and to identify counter-measures.
3,5	Dynamic and static behaviour of containment, crack formation and leakage at penetrations	Estimate the leakage of fission products to the environment.
4,1	Direct containment heating	Increase the knowledge of parameters affecting the pressure build-up due to DCH and determine the risk of containment failure.
5,1	Oxidising environment impact on source term	Quantify the source term, in particular for ruthenium, under oxidation conditions / air ingress for high burn-up and MOX fuels.
5,2	RCS high temperature chemistry impact on source term	Improve predictability of iodine species exiting RCS to provide the best estimate of the source into the containment.
5,3	Aerosol behaviour impact on source term	Quantify the source term for aerosol retention in the secondary side of steam generator (SG) and leakage through cracks in the containment wall as well as the source in containment due to revolatilization in RCS.
5,4	Containment chemistry impact on source term	Improve the predictability of iodine chemistry in containment to reduce the uncertainty in iodine source term.
5,5	Core reflooding impact on source term	Characterise and quantify the fission product release during core reflooding.

Table 1. EURSAFE Research Issue and rationale for selection

4.3 Decision procedure followed in SARP

In the SARNET FP6 project, the SARP work, lead by GRS, was performed in close collaboration with eight organizations, representing diverse types of organizations (IRSN, CEA, EDF, FZK now KIT, GRS, KTH, TUS, VTT). The collaboration between those who perform research and those who use its results was essential to correctly address the problem.

One requirement was that all steps of the decision process must be documented to allow a well-funded judgement for the SARNET community and for the end-users of SARNET products. For this reason a template (Trambauer, 2005) was constructed and updated as a living document during the whole process. The template was used for the three possible “Decisions”:

- a. “To be closed” meaning that the issue is closed by sufficiently improved knowledge
- b. “Reorientation” meaning that the work should be reoriented to fulfil the objectives
- c. “New Item” meaning that a new research item should be set up.

The structure of the template reflected the strategy of the intended decision process, with 10 successive “Actions” with a clear indication of responsibility and the time of execution. A comprehensive description of the “Rationale for the decision to be taken” was included, as well as a step of comments from the SARNET community and the end-users.

4.4 SARP results

The methodology applied in EURSAFE to establish an internationally agreed list of needed research items was proven as adequate. It started with a comprehensive list of phenomena and finished by using a bottom-up approach with 21 items of needed research areas. Investigations showed that a well balanced top-down approach gives very similar results. Therefore the EURSAFE methodology was applied in SARP too.

In a first step seven possible new or reoriented research items were identified. The assessment of the following items was performed (for most of them, only a reorientation or a small extension of experimental and/or analytical work planned in SARNET was necessary):

- Investigation of high burn-up cladding materials at high temperature and under reflooding conditions,
- Spreading of corium into cavity nearby compartments possibly filled with water,
- Effect of hydrogen mitigation systems (esp. PAR) on hydrogen distribution,
- Combustion of hydrogen jets in atmosphere with different hydrogen concentrations (in relation to DCH),
- Retention of aerosols in RCS, SG tubes or concrete cracks under consideration of mechanical resuspension,
- Decomposition of iodides by heat-up in PARs and its impact on source term,
- Ruthenium volatility and behaviour in containment.

Then all ERI issues from Table 1 were reassessed in a top-down approach with respect to gain of knowledge and further necessary activities in SARNET. This last step resulted in the merging of some issues and in the following ranking:

- Six issues were regarded to be investigated further with **high** priority:
 - Core coolability during reflooding and debris cooling of a not totally degraded core,
 - Ex-vessel melt pool configuration during MCCI, ex-vessel corium coolability, by top flooding,
 - Melt relocation into water, ex-vessel FCI,

- Hydrogen mixing and combustion in containment,
- Oxidising impact (ruthenium oxidising conditions/air ingress for high burn-up and MOX fuel elements) on source term,
- Iodine chemistry in RCS and in containment.
- Four issues were reassessed with **medium** priority; these items should be investigated further as planned in the different research programs. The risk significance was reduced due to significant progress of knowledge, but some questions were still open:
 - Hydrogen generation during reflooding and during melt relocation in vessel,
 - Corium coolability in vessel lower head,
 - Integrity of reactor pressure vessel due to external vessel cooling,
 - Direct containment heating.
- Five issues were assessed with **low** priority; the current knowledge was considered as sufficient according to the state and progress of knowledge and the risk and safety relevance and taking into account ongoing activities outside SARNET frame, and they could be closed after the end of the related activities:
 - Corium coolability in external core-catcher,
 - Corium release following vessel rupture,
 - Crack formation and leakages in concrete containment,
 - Aerosol behaviour impact on source term (SG tubes and containment cracks);
 - Core reflooding impact on source term.
- Three issues were marked as “issue **could be closed**”. This means that the current issue knowledge was considered as sufficient assessing both the state and progress of knowledge and the risk and safety relevance and taking into account ongoing activities:
 - Heat distribution in RCS and RCS integrity,
 - Ex-vessel core-catcher and corium-ceramics interaction, cooling with water bottom injection,
 - FCI including steam explosion in weakened vessel.

Concerning the end-users' opinion, the utilities and vendors are mainly interested in the practical value of research results which can be used for developing plant-specific solutions for SAM measures. Risk-informed ranking seems not to be a straightforward approach for assessment because the results of PSA2 are plant-specific. Anyhow the aspect of risk must be taken into account during the assessment discussions by reflecting the risk relevance of each phenomenon with respect to the source term (large releases or not) as threat to the environment. The discussions on each issue should definitely include the applicability of the expected research results to possible SAM measures (planned, envisaged or under development). The approach adopted in the SARP group is an attempt to come within a reasonable time with consideration of all pertinent factors to an agreed decision considering all aspects of innovation and economic efficiency. The outcomes of the final SARP report already served as a proposal to provide the orientations for joint activities for research of common interest and high priority in the SARNET2 FP7 project. An important aspect has also been the total traceability for possible future updates.

5. The SARNET2 project

5.1 General description of the network

Forty-two partners (Table 2) from Europe, Canada, Korea and the USA, participate in the SARNET2 FP7 project (Van Dorsselaere et al., 2011) that started in April 2009 for a period of

Partner	Short name	Country
Institut de Radioprotection et de Sûreté Nucléaire	IRSN	France
KFKI Atomic Energy Research Institute	AEKI	Hungary
AREVA NP GmbH	AREVA GmbH	Germany
AREVA NP SAS	AREVA NP SAS	France
Budapest University of Technology and Economics	BME	Hungary
Commissariat à l'Energie Atomique et aux Energies Alternatives	CEA	France
ENEA – Ricerca Sul Sistema Elettrico SpA	RSE	Italy
Chalmers tekniska högskola AB	CHALMERS	Sweden
Centro de Investigaciones Energeticas Medio Ambientales y Tecnologicas	CIEMAT	Spain
National Centre for Scientific Research "DEMOKRITOS"	DEMOKRITOS	Greece
Electricité de France SA	EDF	France
Energy Institute JSC Sofia	EI	Bulgaria
Ente per le Nuove Tecnologie, l'Energia e l'Ambiente	ENEA	Italy
Forschungszentrum Juelich GmbH	JÜLICH	Germany
Karlsruhe Institute of Technology	KIT	Germany
Gesellschaft für Anlagen- und Reaktorsicherheit mbH	GRS	Germany
National Autonomous Company for Nuclear Activities - Nuclear Research Subsidiary Pitesti	INR	Romania
Institute for Nuclear Research and Nuclear Energy	INRNE	Bulgaria
Inzinierska Vypoctova Spolocnost Trnava s.r.o.	IVS	Slovakia
Jozef Stefan Institute	JSI	Slovenia
Kungl Tekniska Högskolan	KTH	Sweden
Lithuanian Energy Institute	LEI	Lithuania
National Nuclear Laboratory	NNL	United Kingdom
Nuclear Research & Consultancy Group v.o.f.	NRG	Netherlands
Paul Scherrer Institut	PSI	Switzerland
Ruhr-Universität Bochum	RUB	Germany
Tractebel Engineering SA	TRACTEBEL	Belgium
Thermodata	THERMODATA	France
Technical University of Sofia	TUS	Bulgaria
Urad Jadroveho Dozoru Slovenskej Republiky	UJD SR	Slovakia
Ustav Jaderneho Vyzkumu Rez a.s.	UJV	Czech Rep.
University of Newcastle upon Tyne	UNEW	United Kingdom
Dipartimento di Ingegneria Meccanica, Nucleare e della Produzione - Università di Pisa	UNIPi	Italy
Universität Stuttgart	USTUTT	Germany
NUBIKI Nuclear Safety Research Institute	NUBIKI	Hungary
VTT Technical Research Centre of Finland	VTT	Finland

Partner	Short name	Country
VUJE Trnava, a.s. – Inzinierska, Projektova a Vyskumna Organizacia	VUJE	Slovakia
Commission of the European Communities – Joint Research Centres	JRCs	European Union
Atomic Energy Canada Limited	AECL	Canada
Korea Atomic Energy Research Institute	KAERI	Korea
United States Nuclear Regulatory Commission	USNRC	USA
Korea Institute of Nuclear Safety	KINS	Korea

Table 2. List of SARNET2 partners

four years. They represent a large majority of the European actors involved in SA research plus a few non-European important ones. Diverse types of organizations are represented: research organizations, universities, industry, utilities, safety authorities and Technical Safety Organisations (TSO). A new partner, BARC (India), is joining the network in October 2011.

The network is organised with a Steering Committee of ten members in charge of strategy and decisions, advised by an Advisory Committee of end-user organisations. A General Assembly, composed of one representative of each SARNET Consortium member, plus the EC representative, is called periodically for information and consultation on the progress of the network activities, the work orientations and the decisions taken by the Steering Committee. A Management Team, composed of the network coordinator and of seven Work-Packages (WP) leaders, is entrusted with the day-to-day management of the network. In the continuity of the SARNET FP6 project, the SARNET2 FP7 project has been defined in order to optimize the use of the available means and to constitute a sustainable consortium in which common research programmes and a common computer code on SA, ASTEC, are developed. ASTEC capitalizes the whole knowledge produced in the network through new or improved physical models. The Joint Programme of Activities can be divided into several elements:

- Ranking periodically the priorities of the research programmes, harmonizing and re-orienting existing ones and jointly defining new ones when necessary,
- Performing small and large-scale experiments on the highest priority issues as defined in the SARNET FP6 project and jointly analysing their results in order to elaborate a common understanding of the concerned physical phenomena,
- Developing physical models, integrating them into ASTEC, and validating this code versus experiments and through benchmarks on plant applications with other codes,
- Storing all the experimental results in a scientific database, based on the STRESA tool,
- Disseminating the knowledge to students or young researchers, as well to new nuclear emergent countries, through educational courses, textbooks, mobility of personnel between the network partners, and international conferences that become the major SA event in the world.

On the basis of the outcomes of the SARP work, the research programmes focus on the six high-priority issues that were presented in Section 4.4. They are analyzed in the WP N°5 to 8. The experimental efforts are mainly devoted to two of these issues for which real progress toward the closure of the issue is expected: corium/debris coolability and MCCI. For all these 6 issues, the same method is being adopted: review and selection of available relevant

experiments, contribution to the definition of test matrices, synthesis of the interpretation of experimental data, benchmark exercises between codes, review of models, synthesis and proposals of new or improved models for ASTEC. Indeed a key integration aspect is the set-up of the technical circles, each covering a specific detailed topic. They bring experimenters and modellers closer together, concerning test definition, interpretation, model development etc... In each of the domains, additional studies are being performed in order to bring research results into reactor applications. Calculations of SA scenarios in reactor conditions are being performed using various computer codes, including ASTEC, in order to evaluate the importance of the involved phenomena, in particular through uncertainty studies. Sections 5.2 to 5.6 summarize the work that is performed in the WP5 to 8 and on the ASTEC code assessment. Section 5.7 summarizes all activities related to dissemination of knowledge.

5.2 Activities on corium and debris coolability

The major motivation is to reduce or possibly solve the remaining uncertainties on the possibility of cooling structures and materials during SA, either in the core or the vessel lower head or in the reactor cavity, in order to limit the progression of the accident. This could be achieved by water injection, either by ensuring corium retention within the vessel or at least slowing down corium progression and limiting the flow rates of corium release into the cavity. These issues are covered within SAM for current reactors, and also within the scope of the design and safety evaluation of future reactors. The current PSA2 studies still show very large uncertainties in the results of the core reflooding phase. For the issue of in-vessel retention in principle two different aspects have to be considered, the probability for reflooding systems to begin operation in due time, and the status or degree of core damage. If core damage occurs at high pressure, low pressure reflooding systems cannot inject against that pressure. But they may be available with a high degree of reliability. In such conditions it is crucial to evaluate if and when depressurisation of the reactor coolant system occurs which would lead to immediate reflooding. In the bottom of BWRs vessel there is a continuous injection through the control rod and pump seal flushing water. Depending on the reliability and capacity of these systems and the pressure in the RCS, the core degradation may be inhibited.

The following three key situations and processes for the investigation of corium and debris coolability are considered.

Reflooding and coolability of a degraded core

The focus is on the accident phase after water boil-off in the core. Heating and melting may produce a severely damaged, partly molten core with relocated material and partly broken parts. Quenching of such a hot and partly degraded core is the main issue here. The specific case of reflooding of a debris bed is detailed in Section 6.

The experimental database on degraded core reflooding was analysed to derive the crucial information about success of reflooding. The QUENCH experiments in KIT constitute the main part of this database. The behaviour of fuel rod bundles can be outlined in a “reflooding map” with respect to the reflooding mass flow rate and the core damage state to deduce the limits up to which final bundle cooling can be expected to be successful and hydrogen production may be tolerated.

The analyses show that even at the onset of severe core degradation at temperatures up to app. 2200 K, the accident progression can be stopped with a sufficiently high flowrate for

core reflooding of ~ 1 g/s per rod. The reflooding map on core degradation and hydrogen release is still under development and is considered as a tool to summarize the existing knowledge and to identify blank areas for efficient future experimental work.

Remelting of debris, melt pool formation and coolability

If core cooling fails, a melt pool will form in the core and melt might flow down into remaining water in the lower head. The TMI-2 accident indicated that even though coolability of the core is not attained, a coolable configuration may result from break-up of the melt in the water of the lower head. If cooling in the core and in the lower head is not possible, the development of a melt pool in the lower head has to be analysed and it has to be established whether a melt pool can be kept in-vessel due to external vessel cooling; if it is not possible, the timing and modes of vessel failure have to be considered. This is the general objective of the LIVE programme (KIT). These phenomena resulting from core melting are studied experimentally in large-scale 3D geometry and in supporting separate-effects tests, with emphasis on the transient behaviour. One experimental result is e.g. that melt pouring near the vessel wall at the beginning of the test results in considerable asymmetric heat flux distribution even during the steady state. The time period of the solidification ranges from 50 minutes to several hours, depending on the cooling conditions and the position of the melt/crust interface.

The external cooling conditions, which are the second important aspect for achieving in-vessel coolability, are investigated by the CNU experimental programme (CEA) which is a unique experimental set-up, large scale, dedicated to the study of two-phase flow with steam production around a heated RPV geometry.

If all the attempts to cool down the vessel fail, the location and size of the vessel breach are of concern. Up to now the following main conclusions can be drawn for large PWRs: when the vessel fails, the liquid corium is mainly oxidic with potentially some metal. The mass of corium that can be ejected into the reactor pit at vessel failure is estimated between 2 and 20 tonnes. The breach is most probably located on the lateral surface of the vessel. Only local breaches are expected and not vessel unzipping.

Ex-vessel debris formation and coolability

A porous debris bed can be formed in a water pool of the reactor cavity due to the fragmentation of the molten corium jet ejected from the lower head of the vessel. The water pool is available through cavity flooding (e.g. SAMs in Swedish and Finnish BWRs) or water accumulation in the sump of a PWR due to Loss of Coolant conditions or containment spray. This is a similar process to the in-vessel situation, when melt relocates from the core to a water filled lower head. The large depth of water pools in BWRs yields additional effects.

The first issue concerns the debris bed formation by break-up of melt, with the DEFOR (KTH) and FARO (JRC/IE-Ispra) experiments. The second issue concerns the investigations on coolability of debris beds, with the STYX (VTT) and DEBRIS (IKE) experiments (for the latter, see more details in Section 6).

Bringing research results into reactor application

As an example of research results for reactor applications, the IVR via external reactor vessel cooling (ERVVC) has been recognised as a feasible and promising SAM strategy for VVER-440/V213 reactors. The most important design features of these reactors, favourable for

adoption of the IVR concept, are low thermal power, RPV without penetrations in lower head, massive stainless steel vessel internals, large volume of residual water in lower head and high driving head for natural circulation in ERVC loop.

Recent activities devoted to IVR concept via ERVC for standard VVER-440/V213 reactors are performed in the frame of SARNET as well as within national programmes performed in the countries operating this type of reactors. From the results obtained so far it follows that there should be sufficient gap width (~ 1 cm) between RPV wall and thermal/biological shield for the coolant flow in natural circulation regime alongside the outer surface of RPV wall. Further research should be focused on confirmation of the estimated heat flux values. Here the outcomes of the SARNET2 project and results of ASTEC analysis will be of high importance.

In order to evaluate the ability of current advanced codes to predict in-vessel core melt progression and coolability of the degraded core, a benchmark is being organized in close collaboration with the OECD/NEA/CSNI. It addresses an alternative scenario of the TMI-2 accident.

5.3 Activities on MCCI

In the postulated case of a SA with vessel melt-through, the containment is the ultimate barrier between the corium and the environment. The addressed situation is the reactor pit initially dry but with the possibility of water injection later during MCCI. The work programme has been designed to be complementary with the MCCI project of the OECD/NEA that finished in 2010.

Recent 2D experiments like VULCANO (CEA) in prototypical materials have provided new results that questioned the reliability of the available models and their extrapolation to reactor conditions. As an example, it becomes clear that new effects have to be taken into account to be able to describe the ablation anisotropy observed in case of silica-rich concrete and the different behaviour of limestone concrete. This anisotropy was also present in the ablation of Chernobyl silica-rich concrete. The intention is thus to gain sufficient experimental data in order to determine which phenomena are responsible for the observed isotropy/anisotropy of the concrete ablation.

Concerning the oxide/metal configuration, only few experimental programmes were conducted with stratified pools using simulant melts, for instance in KIT the large-scale 2D BETA test series with a large test matrix, and the series of COMET experiments, which were performed in alumina thermite within the LACOMERA EC project. They provided a valuable database on long-term MCCI for various initial and boundary conditions. The VULCANO experiments with oxide and metal pools have the unique characteristics of providing heat to the oxide layer, like in the reactor case. Several experiments were performed (VULCANO, MCCI-OECD, HECLA in VTT) but more data are required to improve knowledge in these configurations. The following question remains open: does a stratified configuration exist in the reality? The other need is to improve in the modelling the stratification criteria for onset and termination of stratification.

The other current experiments are MOCKA (KIT) at a large-scale in simulant materials, COMETA (UJV) for thermochemistry tests on real corium samples, the Laser melting facility (JRC/ITU), and CLARA and ABI (CEA) SETs in simulant materials.

Water-cooling is the main available way to terminate the concrete ablation. It was mainly studied within the OECD/NEA MCCI project. Recently, interest has been shown to pursue R&D on concepts that could be used to provide bottom-cooling in the cavity of current reactors.

5.4 Activities on containment issues

The considered issue is the threat to the containment integrity, due to two types of highly energetic phenomena: steam explosion and hydrogen combustion. Steam explosion may be caused by ex-vessel FCI due to a RPV failure and pouring of the reactor core melt in the flooded reactor cavity. Hydrogen combustion (deflagration and detonation) may be caused by ignition of a gas mixture with high local hydrogen concentrations, which may be due to the imperfect mixing of the containment atmosphere. Phenomena linked to these threats are considered as well. Essential insights and results from this research should be applicable to actual NPPs.

Ex-vessel FCI may lead to steam explosion. The corium ejected in the reactor cavity after vessel failure may lead to high-pressure loads on the containment or vital components in case of FCI. The work performed in the frame of SARNET and SERENA-1 (OECD/NEA project) allowed the identification of the major uncertainties that make difficult to quantify containment safety margins for an ex-vessel steam explosion. These uncertainties mainly concern the level of void in the pre-mixing phase and the role of material properties on explosion energetics. A new OECD/NEA project (SERENA-2) has been launched in October 2007 to resolve these uncertainties by performing a limited number of well-designed tests with advanced instrumentation reflecting a large spectrum of ex-vessel melt compositions and conditions in the KROTOS (CEA) and TROI (KAERI) facilities, and the required analytical work to bring the code predictive capabilities to a sufficient level for use in reactor analyses. The main objective in SARNET is to further review and debate the progress made in the SERENA-2 programme, and to propose and perform any activity that might be needed to complement (and possibly have positive feedback on) the work performed in SERENA-2, with the help of data produced in SARNET such as MISTEE, DEFOR and DROP experiments in KTH.

Phenomena that are linked to the hydrogen-in-containment issue, which is still today of highest priority, are addressed. This issue covers the containment thermal-hydraulics, including the hydrogen distribution, the different hydrogen combustion regimes, their impact on containment structures and measures to prevent (severe) combustion processes or at least to mitigate their consequences with specific devices like PARs or with accident management measures, like containment sprays. The involved experiments are: TOSQAN and ENACEFF (IRSN and CNRS/Orléans), MISTRA (CEA), HyKA and DISCO (KIT), CONAN (Univ. of Pisa), THAI (Becker Technologies, Germany). Benchmarks between codes are performed on most of these experiments.

5.5 Activities on source term

The overall objective is to reduce the uncertainties associated with calculating the potential release of radiotoxic fission products to the environment that may occur during a severe accident in water-cooled nuclear reactors. It concentrates on iodine and ruthenium, given their high radio-toxicity, noting that the release of ruthenium is enhanced in oxidising atmospheres, such as those that may follow air ingress into the RCS. The research treats the transport of these elements through the primary circuit, including consideration of the SG, and their behaviour in the containment. The prediction of volatile iodine and ruthenium species in the containment atmosphere of particular importance, because they are hard to remove by containment sprays or by filtration while venting the containment. For ruthenium, the enhanced release from the fuel in oxidising conditions is also studied.

Full advantage is taken of cooperation with international programmes such as Phébus FP (Clément et al., 2005), the International Source Term Programme (ISTP) (Clément et al., 2005), and the projects of the OECD/NEA/CSNI, to avoid duplication of experiments, to help consistency of the programmes and to identify remaining needs.

As concerns the oxidising influence on source term, the technical work concentrates in particular on ruthenium source term from the fuel up to its behaviour in-containment:

- Release of fission products from fuel (FIPRED experiments in INR, RUSSET in AEKI, VERCORS past RUSSET and VERDON future ones in CEA, Phébus FP past experiments in IRSN): release from high burn-up and MOX fuels; role of fuel cladding, i.e. the competition between cladding oxidation, UO_2 oxidation and fission products release; fission products release under mixed steam-air conditions, which are more realistic than 100% air conditions in accident situations;
- Ruthenium transport in RCS (experiments in Chalmers University): thermodynamic behaviour of ruthenium oxides; reactivity with surfaces and other chemical compounds such as caesium;
- Ruthenium behaviour in containment (EPICUR experiments in IRSN, VTT ones, THAI ones in Becker Technologies): behaviour of ruthenium oxides as aerosols, and their potential conversion to volatile forms; thermodynamic behaviour of ruthenium species in liquid phase and potential volatilization.

As concerns the iodine chemistry in the RCS and containment, two main situations are addressed:

- Iodine transport in circuits (CHIP experiments in IRSN, EXSI ones in VTT): kinetics of gaseous phase reactions; speciation of revaporised iodine and of other fission products; development of a databank from plant iodine spiking data and associated development of a correlation-type model covering some steam generator tube rupture (SGTR) events, volatile iodine mass transfers and adsorption/deposition in SG secondary side in case of a SGTR event;
- Iodine behaviour in containment (EPICUR experiments, RTF ones in AECL): mechanisms of iodine association with painted surfaces (adsorption of iodine from particulate iodides deposited on “wetted” surfaces); subsequent volatile iodine formation from iodine-loaded paint; radiolytic destruction of gaseous iodine species to form nucleate particles and subsequent behaviour of these particulate iodine oxides; iodine binding on sump materials and in sump screen blockages; effect of PARs on iodine source term.

5.6 ASTEC code assessment and improvements

IRSN and GRS jointly develop the ASTEC code to describe the complete evolution of a SA in a nuclear water-cooled reactor, including the behaviour of engineered safety systems and procedures used in SAM (Van Dorselaere et al., 2009). The new series of versions V2 (Figure 3) can simulate the EPR, especially its external core-catcher, and it includes the advanced core degradation models of the ICARE2 IRSN mechanistic code.

IRSN and GRS deliver the successive code versions and the corresponding documentation to the code users: ASTEC V2.0 in July 2009; V2.0rev1 in mid-2010 and V2.1 foreseen in 2013. They also assure the code maintenance and the support to the code users, notably through Users Club meetings that are organized about every eighteen months (the next one in spring 2012).

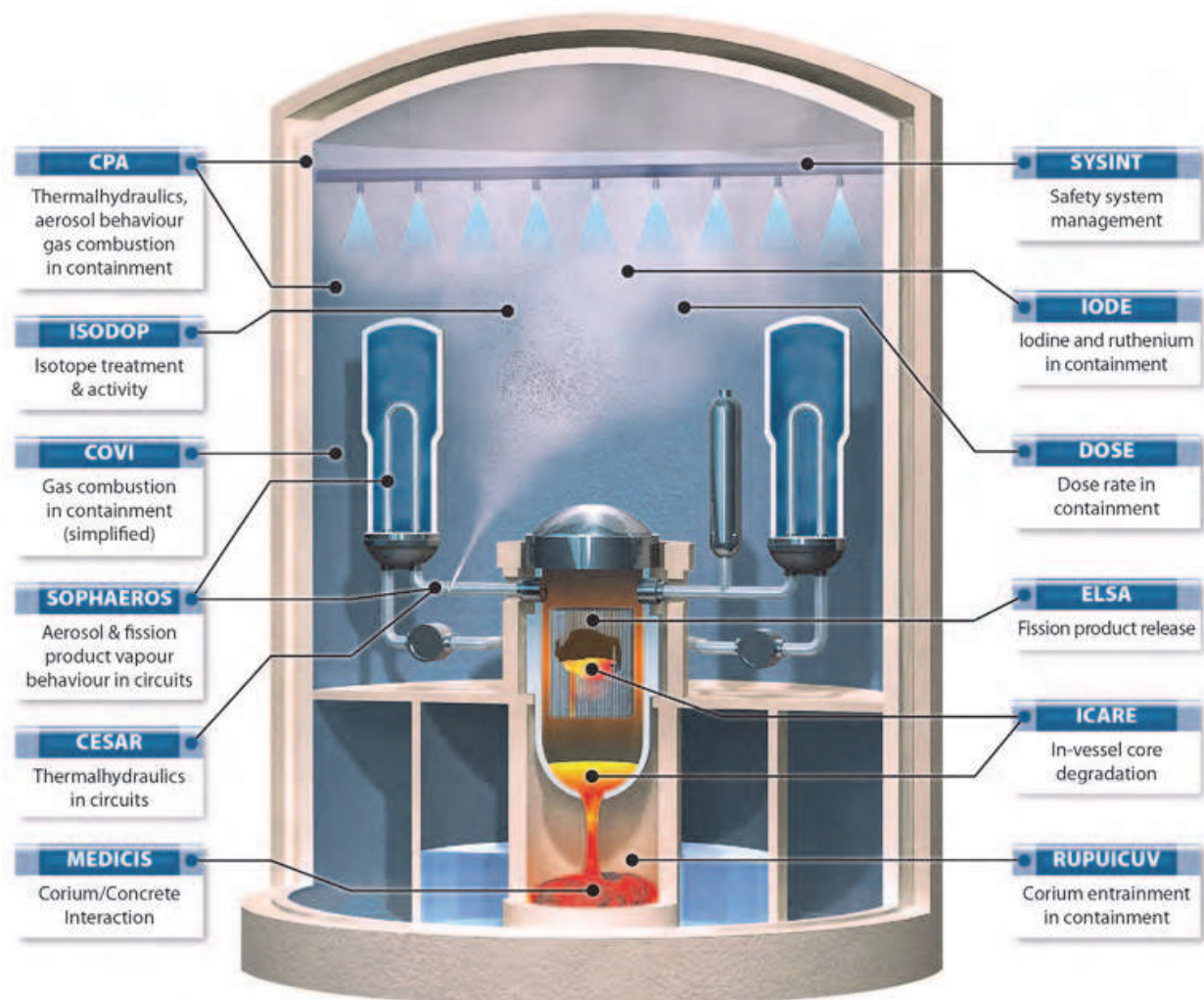


Fig. 3. Structure of the ASTEC integral code for SA simulation

Twenty-nine organizations collaborate on the development and assessment of the successive ASTEC versions. The developments will account for the model improvements proposed by the joint research activities of the JPA. Besides, four partners work on the model adaptations to simulate SA sequences in BWR and PHWR⁴ reactors: IKE and KTH for BWR, INR and AECL for PHWR. They write model specifications, and validate the code against adequate experiments and benchmarking with other codes. Most ASTEC models are already applicable to these two types of NPPs except for core degradation. The BARC Indian partner is developing new PHWR core degradation models and validating them against Indian experiments.

The assessment activity mainly consists:

- On one hand in validating the code against experiments of diverse types (SETs, CETs, and integral tests). The comparison of code results with integral experiments such as Phébus FP and with real plant accidents such as TMI-2 is an essential task;

⁴ PHWR : pressurized heavy water reactors (including the CANDU type that is designed by Canada)

- On the other hand in covering a broad matrix of ASTEC reactor applications, aiming at the most important SA scenarios for the diverse types of reactors (PWR including VVER, BWR and PHWR). Sensitivity and uncertainty calculations are being performed in order to demonstrate the reliability and consistency of the ASTEC calculations. Although not the prime objective, partners may benchmark ASTEC with other reference codes that they master, such as the integral codes MELCOR and MAAP and the detailed codes such as ICARE/CATHARE, ATHLET-CD, SCDAP/RELAP5, COCOSYS, CONTAIN, TONUS...

5.7 Spreading of excellence

The objective of the DATANET database, developed in the frame of SARNET, is to collect the available SA experimental data in a common format in order to ensure their preservation, exchange and processing, including all related documentation. The data are both previous experimental data that SARNET partners are willing to share within the network and all new data produced within SARNET. DATANET is based on the STRESA tool (Zeyen, 2009) developed by the Joint Research Centre (JRC) in Ispra (Italy) and now managed by JRC-IE in Petten (The Netherlands). It consists of a network with several local databases. All access rights are managed in accordance with the rules adopted in the SARNET consortium. The protection of confidential data is an important feature that is taken into account as the information security of the database. Six STRESA nodes are open and the results of about 250 experiments from 35 facilities have been implemented. JRC-IE can create new local STRESA nodes for partners and support the users through training sessions when necessary.

The public web site (www.sar-net.eu) aims at providing general information on the SA research field to the general public. For the communication between all network members, the e-collaborative Internet Advanced Communication Tool is used. About 300 papers related to SARNET work in the last 5 years have been presented in conferences or published in scientific journals. The dissemination of information is also done through periodic newsletters or participation to public events.

Four ERMSAR conferences (European Review Meetings on Severe Accident Research) have been organized in the last five years successively in France, Germany, Bulgaria and Italy as a forum to the SA community. They are becoming the major event in the world on this topic. The 4th one, hosted by ENEA (Italy) on May 11-12, 2010 in Bologna (Italy), gathered 100 participants.

The Education and Training programme is focusing on raising the competence level of the university students (Master and PhD) and researchers engaged in SA research. Towards this purpose, education courses are elaborated on the phenomenology of the SA various areas. The teaching is not a survey but an in-depth treatment in order to allow the students and researchers to understand the methodology in the topics further and use analysis computer codes, mainly ASTEC, more effectively for any type of NPP. The description of the scenarios with event trees and fault trees is performed, with indication of the probabilities of the various events occurring. Best-estimate analyses are provided with uncertainty analyses. Close links exist with the European ENEN association (European Nuclear Education Network). Four one-week educational courses were organised during the last five years, gathering from 40 to 100 persons: the latter was organised in the University of Pisa in January 2011, with a special focus on Gen.III NPPs. Another training course will be

proposed in the future for staff of plant operators or regulatory authorities, with emphasis on identifying what the SAM procedures are based on, and why they are effective.

The textbook on SA phenomenology was drafted during the SARNET FP6 project. It covers historical aspects of water-cooled reactors safety principles and phenomena concerning in-vessel accident progression, early and late containment failure, fission product release and transport. It contains also a description of analysis tools or codes, of management and termination of SA, as well as environmental management. It gives elements also on Gen.III reactors. The final review was performed in 2010, and the publication is planned in the second part of 2011.

Finally, a programme enables university students and researchers to go into different laboratories for education and training in the SA area. Some stages for master thesis may be organised in the ENEN framework to obtain the 20 credits necessary for the achievement of the European EMSNE (European Master of Science in Nuclear Engineering) certification. The staff deputation programme has involved for the last five years about 40 secondments with an average duration of 3 months: a researcher from one laboratory can spend several months in another European Laboratory where he/she would participate in an area of the SA research ongoing there.

6. Illustration on a specific R&D issue: Reflooding of debris beds

One of the high-priority issues concerns the core and debris coolability and thermal-hydraulics within particulate debris during core reflooding. PSA results do not give a unanimous answer for the ranking of the issue. While in German PSA studies the possibility of reflooding is classified with low probability, French PSA on 900 MWe reactors give a higher probability. Finally because reflooding of a degraded core can potentially terminate the core degradation and stop the accident, corresponding SAM measures are intended and consequently the investigation of conditions for successful reflooding is important.

New QUENCH-Debris experiments (KIT) and CODEX experiments (AEKI) are foreseen in bundle configurations, analysing the relocation of cladding and fuel and the formation and cooling of in-core debris beds to gain information on the characteristics of the created particles. The main objective of these tests is the investigation of these processes under prototypical boundary conditions for a whole bundle. The QUENCH-Debris facility consists in modifications of the QUENCH existing facility to study debris formation and coolability within a rod bundle. Two tests are planned during the SARNET2 timeframe.

The DEBRIS facility (IKE) concerns model-oriented experiments for improvement of constitutive laws for friction and heat transfer as well as study of specific two dimensional effects under top and bottom flooding conditions at different system pressures. New POMECO test facilities (KTH) are designed and constructed to perform isothermal and boiling two-phase flow tests with better instrumentation and flexibility to accommodate various prototypical conditions: they aim at analyses under boil-off conditions with emphasis on basic laws and specific 2D effects (downcomers) more oriented at lower head or ex-vessel situations but also addressing basically the situation in the degraded core. Both DEBRIS and POMECO programmes deal with irregular particles aiming at realistic debris.

IRSN is preparing larger quenching experiments with 2D porous media allowing multi-dimensional progression of the quench front. This PEARL programme (Figure 4) (Stenne et

al., 2009) will simulate the reflooding of a debris bed, characteristic of an in-core debris bed, surrounded by a more permeable medium (such as intact structures and rods). PEARL goes beyond DEBRIS quenching analyses by the larger size (60 cm diameter vs. 15 cm in DEBRIS) and thus the possibility to perform extended analyses on multidimensional effects. It will also provide a general basis for the assessment of the overall behaviour described in the codes (both in- and ex-vessel phenomena).

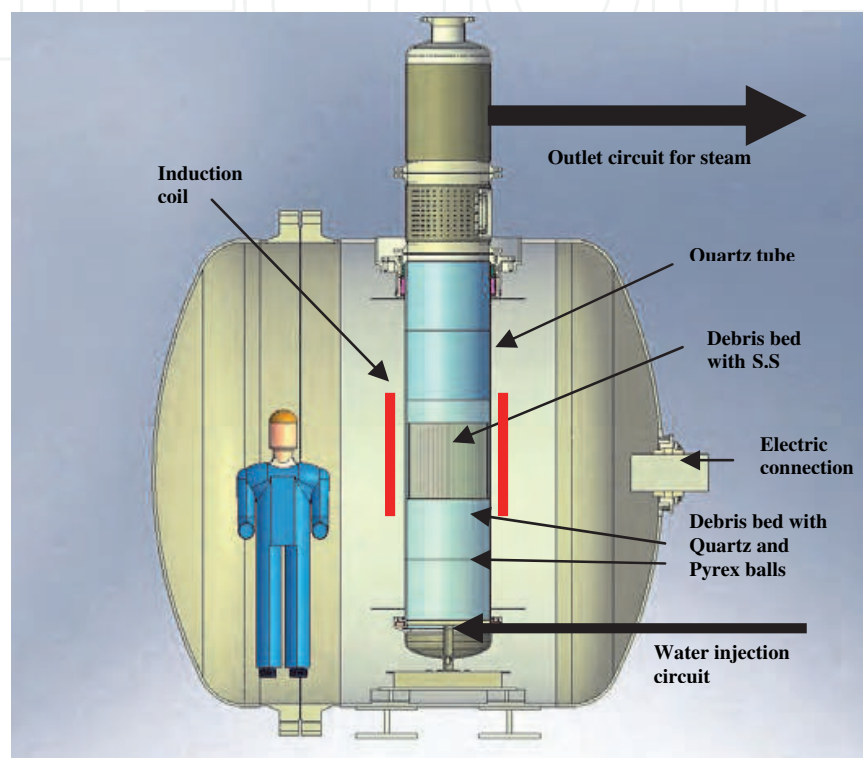


Fig. 4. PEARL facility (©2010 IRSN)

The PRELUDE preliminary program is ongoing in IRSN to test the performance of the induction heating system on stainless steel particles, in order to optimize the instrumentation in a two-phase flow. The debris bed is one-dimensional, with a smaller size than PEARL, at atmospheric pressure and up to temperatures of 1000°C. The investigated parameters are:

- Stainless amagnetic steel particles, 2 and 4 mm in diameter,
- Inlet water velocity between 1 and 8 mm/s (4 to 30 m³/h/m²), in the range foreseen in PEARL test matrix,
- Power at 300 W/kg (maintained or not during the reflooding phase),
- Initial temperature before reflooding at 420 K, 500 K, 600 K and 1000 K.

Additional PRELUDE experiments were performed to evaluate the power distribution inside a larger debris bed diameter (from 110 to 280 mm). This campaign ended with two experiments with a heating sequence of a debris bed (test section diameter 180 mm, particles 4 mm) up to 1000 K at about 140 and 200 W/kg before the water injection. Those

experiments were well instrumented with many thermocouples inside the debris bed (different radial and axial positions) to follow the water front propagation along time (Figure 5, shows the thermocouples measurements at different axial levels).

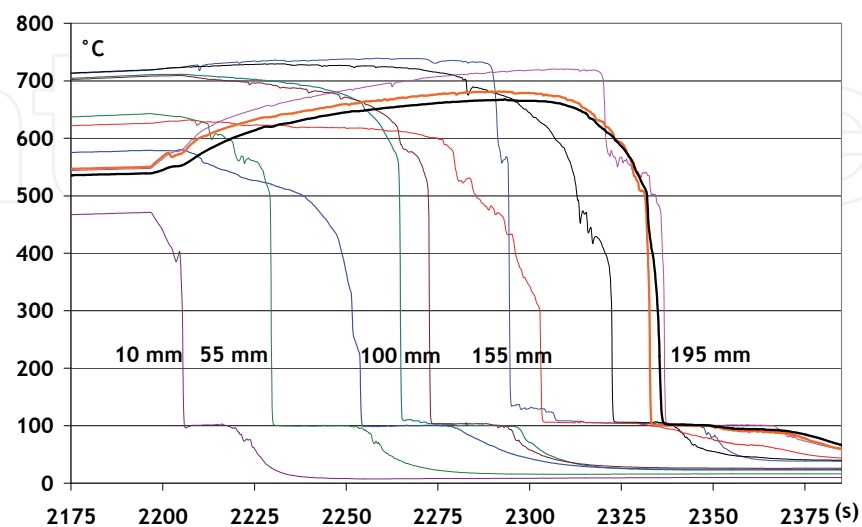


Fig. 5. PRELUDE measurements of the water front evolution along time

The objective is to assure the consistency between the PEARL, DEBRIS and QUENCH-DEBRIS experimental programmes.

Concerning the coolability of porous media, theoretical analyses have indicated the importance of multi-dimensional effects. Quenching analyses performed in SARNET showed agreement concerning a strongly favoured coolability by inflow of water from lateral water-filled regions of the core with higher porosities. Since lateral water inflow, especially via lower regions, strongly improves coolability, in general the coolability is much better than concluded from 1D analyses with top flooding. 2D/3D computer codes including adequate descriptions of constitutive laws are thus required to analyse the real coolability situation. Also, it is necessary to improve the modelling of the formation of porous media in the core.

In parallel, plant applications must be performed, along with uncertainty studies, on different in-vessel geometrical configurations, taking into account water supply, in order to reveal major trends (cooling vs. melt pool formation). The modelling work is being done in the detailed codes ICARE/CATHARE (IRSN) and ATHLET-CD (work by IKE on the MEWA module of the German GRS code). These codes aim at a detailed understanding and simulation of the physical phenomena. Their validation on above experiments will allow improving their models. The final objective is to derive simplified models from these codes and to implement them into the ASTEC code.

7. Conclusion

For severe accidents in Generation II-III nuclear power plants, R&D is progressing on the remaining open issues that have been ranked in the SARNET network of excellence. The

work concerns new experiments and new physical modelling, in particular in the ASTEC integral code that is considered as the European reference SA code. For Gen.II NPPs, the objective is to reduce the remaining uncertainties on SA and consolidate the accident management plans in order to lead their safety level closer to the level of the new Gen.III plants under construction.

The accident in the Fukushima plant occurred when this chapter was written. It is very likely that the 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 highest-priority issues that were selected in the SARNET work programme have clearly conditioned the evolution of the accident: for instance reflooding of a degraded core, hydrogen explosion in the containment, and degradation of fuel bundles and release of fission products in an air atmosphere (in the case of spent fuel pools). The SARNET work programme is being reviewed in order to get as soon as possible a feedback of this accident on the needs of further R&D. In particular, it seems likely that the aspects of mitigation of the SA consequences will have to be emphasized in the near future.

A first step towards a sustainable integration of the European SA research capacities has been reached. A strong link must be kept between the “SARNET community” and the “PSA2 community”, in continuation of the ASAMPSA2 FP7 project on PSA2 best-practice guidelines (ASAMPSA2, 2011). The European SNETP (Sustainable Nuclear Energy Technology Platform), that gathers all nuclear fission actors and aims at providing the stakeholders and the public with a 2020–2050 vision on R&D, has delegated to SARNET the coordination of R&D on SA for Gen.II-III NPPs. Such a living and unique pool of experts should periodically assess the remaining issues on SA, notably using the results of all international programmes (OECD/NEA, EC FP7-8, ISTP,...), update the research priorities and propose relevant R&D programmes to address them. Their scope of investigations could extend in the future to the new Generation IV plant designs.

Efforts will continue in parallel on the transfer of knowledge to younger generations through the ERMSAR periodic international conferences, educational courses and delegations in laboratories.

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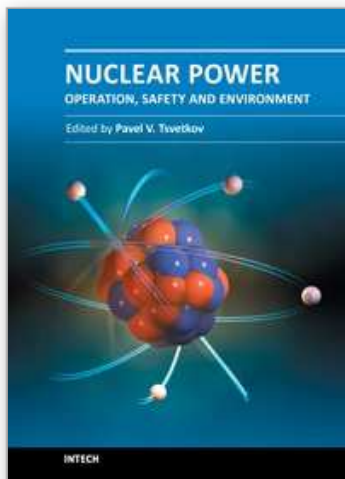
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Today's nuclear reactors are safe and highly efficient energy systems that offer electricity and a multitude of co-generation energy products ranging from potable water to heat for industrial applications. At the same time, catastrophic earthquake and tsunami events in Japan resulted in the nuclear accident that forced us to rethink our approach to nuclear safety, design requirements and facilitated growing interests in advanced nuclear energy systems, next generation nuclear reactors, which are inherently capable to withstand natural disasters and avoid catastrophic consequences without any environmental impact. This book is one in a series of books on nuclear power published by InTech. Under the single-volume cover, we put together such topics as operation, safety, environment and radiation effects. The book is not offering a comprehensive coverage of the material in each area. Instead, selected themes are highlighted by authors of individual chapters representing contemporary interests worldwide. With all diversity of topics in 16 chapters, the integrated system analysis approach of nuclear power operation, safety and environment is the common thread. The goal of the book is to bring nuclear power to our readers as one of the promising energy sources that has a unique potential to meet energy demands with minimized environmental impact, near-zero carbon footprint, and competitive economics via robust potential applications. The book targets everyone as its potential readership groups - students, researchers and practitioners - who are interested to learn about nuclear power.

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