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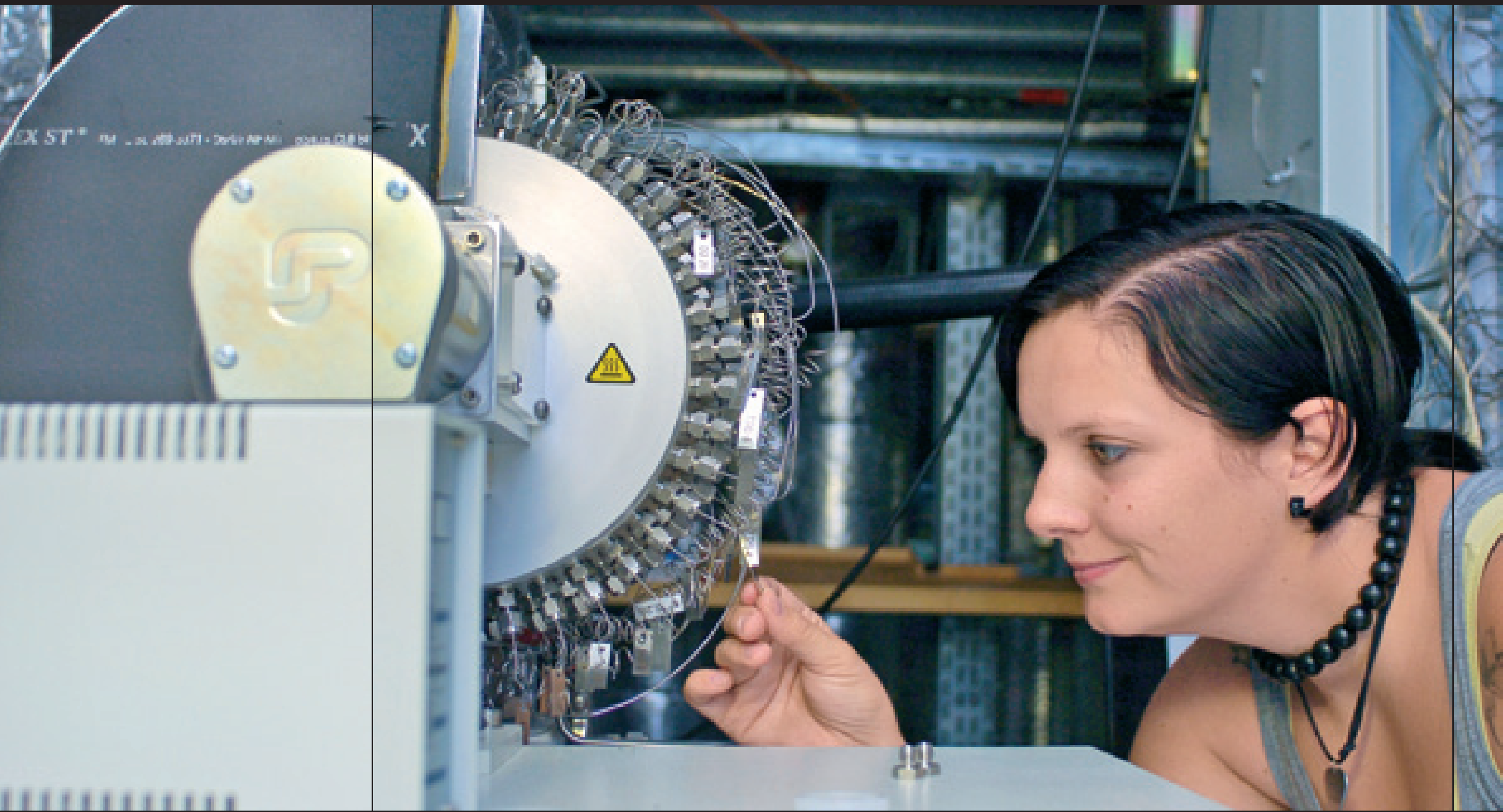

Scientific Highlights 2010

Nuclear Energy and Safety

Cover photo:
Implementation of capillaries in the multiport valve of the mass spectrometer system used for gas mixture composition measurements in the PANDA facility.

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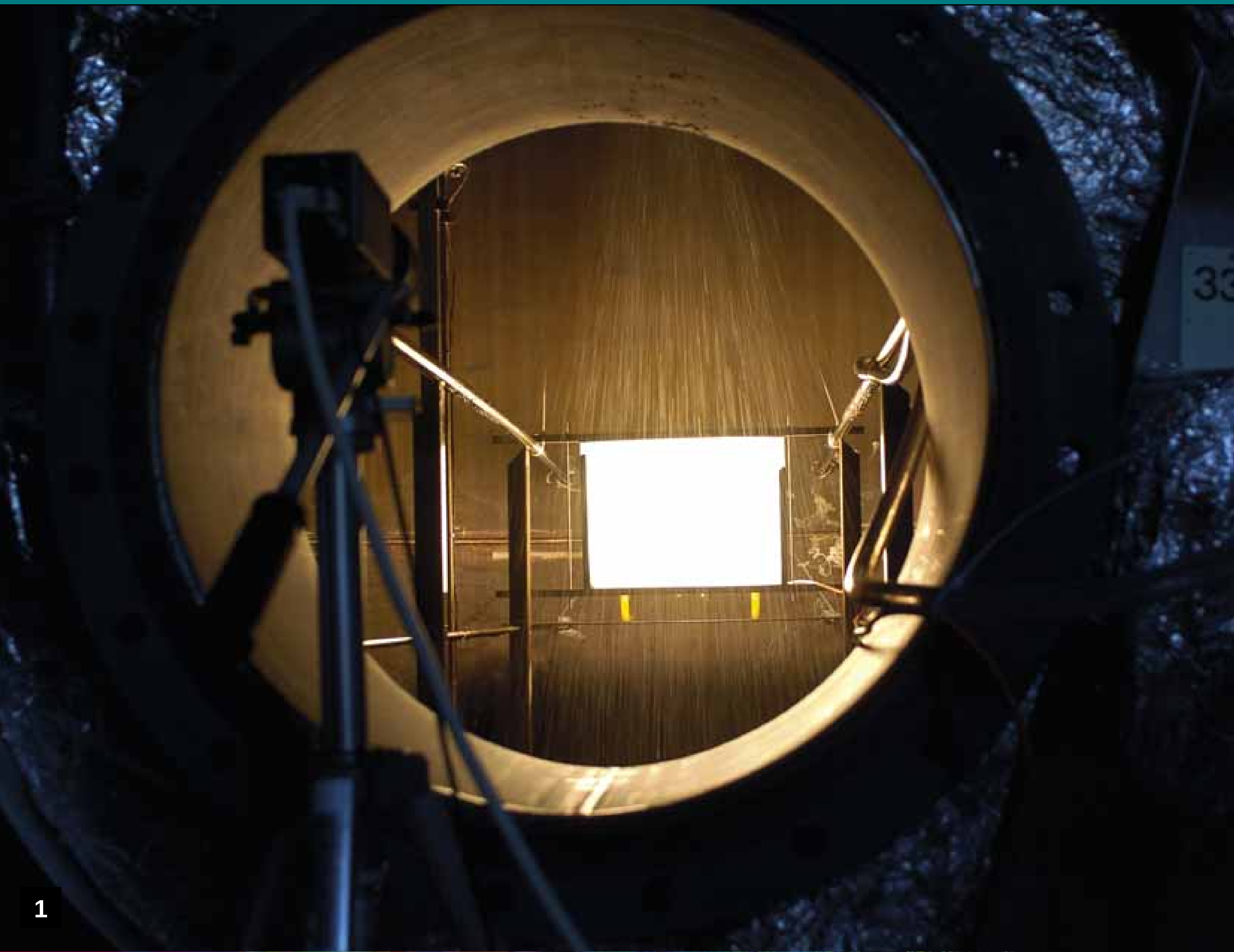
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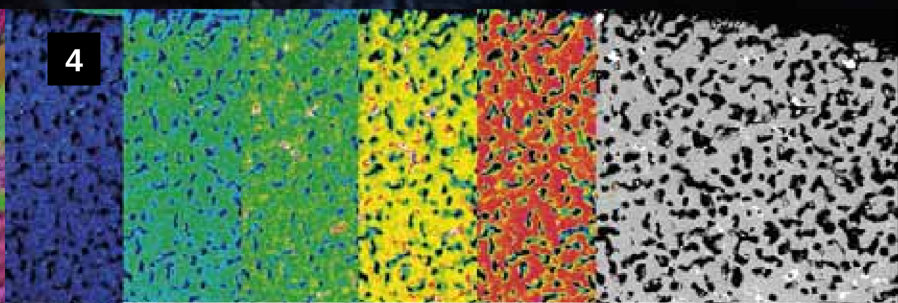
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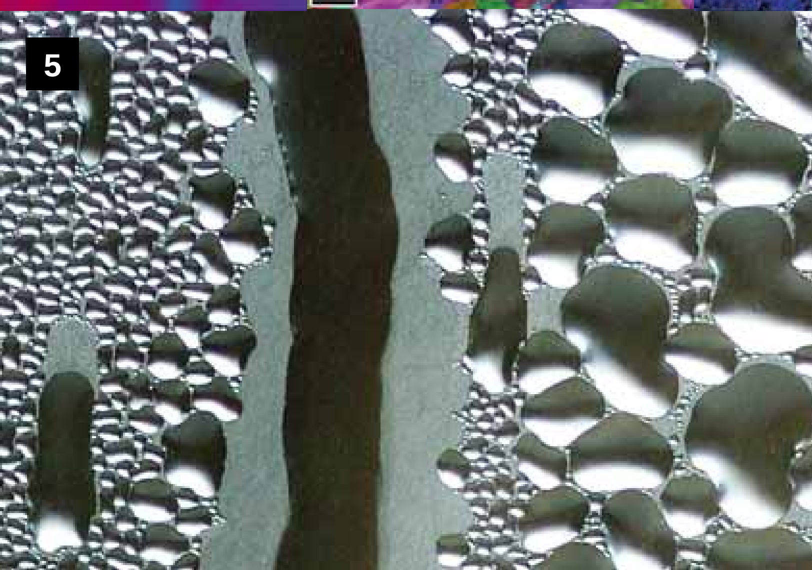
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Nuclear Energy and Safety Research Department (NES) 5

The strategic areas of activity within NES comprise the following:

- contributions to the safe and economic operation of the existing Nuclear Power Plants (NPPs) in Switzerland, and proof of the safe geological storage of radioactive waste, by reinforcing the scientific bases of the technologies in the appropriate areas;
- support to the reactor operators and safety authority in Switzerland, as well as the securing of stand-by functionality in key areas, particularly those requiring the services of a Hot Lab;
- preparation of inputs to 'stakeholders' for decision-making purposes;
- promotion of nuclear energy by means of R&D in terms of increased sustainability, safety and economy;
- training of young nuclear specialists, including those with experience in other energy systems, over a broad spectrum of disciplines;
- support and participation in worldwide activities involving nuclear energy, and of its relevance to current needs within Switzerland.

NES is structured into five research laboratories according to its specific scientific and technical areas of competence. It operates the only Hot Lab in the country, and the Reactor School offers education and training programmes for present and future reactor operators. In strong partnership with the two Federal Institutes of Technology, ETHZ and EPFL, NES contributes to the newly launched education programme: the Swiss Master in Nuclear Engineering.

◀ **Snap-shots of selected NES activities:**

1. **Spray droplet characterization experiment;**
2. **Gamma-spectroscopic image of a fuel rod;**
3. **Crack propagation with grain boundaries;**
4. **Electronic-probe-micro-analysis results;**
5. **Droplet formation on a cold surface;**
6. **Aerosol deposition on steam generator tubes.**

Scientific findings on nuclear safety issues to guide future stakeholder decisions

Jean-Marc Cavedon
Nuclear Energy and Safety Research Department, PSI

Although from a purely objective viewpoint the economical and ecological ratings of nuclear electricity production are good, and often better than those of renewable energies, its social acceptance is ultimately dependent on stakeholder opinions regarding plant safety and the safe disposal of radioactive waste. This year's selection of scientific highlights seeks to illustrate how we strive to provide answers to these central issues. From the behaviour of the plant as a whole, the actions of its operators to prevent the release of radionuclides to the environment, or by studying the effects of small or large losses of coolant, we establish the scientific facts necessary to facilitate decision-making at the stakeholder level.

There is a growing consciousness of the factual synergy between electricity produced by nuclear power plants and that from renewable sources, seen together as a portfolio of technologies to ease, and even eliminate, our dependency on fossil fuels, with their accompanying detriments to our society and environment. This synergy, in improving the ecological and economical dimensions of a sustainable energy supply, does however not extend to the third dimension of sustainability: the social one.

Renewable sources, at least at their present stages of development, are socially rather well-accepted. In contrast, the history of the perception of nuclear electricity production in our developed societies is often one of concern over the safety of operation of present nuclear plants, and the proper handling of nuclear waste.

Scientific studies in recent years have been used mainly to support technical decisions. This has contributed to the fact that Switzerland now has three technically-acceptable reactor sites for the replacement of its existing plants, or for the building of new ones. The country also has six technically-acceptable sites for deep underground repositories. The decision process has now moved to the stakeholder level.

Through this year's selection of scientific achievements, we seek to illustrate how we use scientific and technical state-of-the-art tools to answer precise questions relating to the two central societal issues: *Are nuclear power plants safe? Has the waste disposal problem been solved?*

At the time of printing, the severe nuclear accident at the Fukushima Daiichi site has not yet delivered all the information that one could hope to draw from it. However, the incident already strikingly underlines the relevance of safety studies launched well before any such accident occurs, especially for extremely low-probability events affecting nuclear power plants.

Simulating and mitigating unlikely events

One rare event in the operation of a nuclear reactor is the rupture of a steam generator tube, which would connect the primary cooling circuit of a Pressurized Water Reactor (PWR) to the normally contamination-free secondary circuit. Should such an event result in a sizeable opening (of area a few square centimetres), and should it occur together with some other unlikely lack of cooling event, it would open a pathway for radionuclides in aerosol form from the reactor core to the environment, by-passing the containment vessel. Although the event chain is highly unlikely, it was deemed necessary by a group of international nuclear regulators, nuclear constructors, operators and research centres to study the high-velocity release of aerosols from a tube breach, and to estimate the efficiency of possible mitigation measures. The experiment was coordinated and executed at PSI. The title of the article by T. Lind et al., included in this document, gives both the problem and part of the solution: secondary-side flooding of the ruptured pipe reduces the aerosol transport by a factor of 100 or more.

When this and other very low-probability event chains are studied, other unlikely events have also to be brought into the safety analyses, such as delays in action (or even errors made) by highly trained operators to emergency situations. This theme is taken up by V. Dang et al. in their article describing analysis of dynamic operator/plant interaction in a postulated emergency situation. A computer simulation of plant behaviour is combined with the simulated response of the operators to determine what actions they should take, and when they should take them. In the context of an international exercise, new insights into the assumptions made concerning the constraints imposed on successful operator response have been gained, and a tool to study some of the key uncertainties that remain in this challenging topic has been constructed.

The previous examples deal with topics for which improvements can be brought to existing nuclear plants. But we have also explored the near future: i.e. the upcoming (third) generation of nuclear plants worldwide, which incorporate, by design, many safety features and improvements derived from operational experience gained from existing plants, plus some important novel safety features. Y. Yun et al., in collaboration with the Finnish regulator STUK, are conducting independent safety assessments of the EPR nuclear plant under construction at Olkiluoto, Finland. Results indicate that the hottest fuel cladding temperature for this plant would not reach levels that would initiate cladding rupture, even in the event of a loss of coolant through a large pipe break.

Safety in normal plant operation and waste management

The life of a nuclear fuel pellet, even under normal operating conditions, is quite challenging. The pellet is subjected to high temperatures and steep temperature gradients in a high irradiation flux environment, and the appearance of new chemical elements (the fission products) within the pellet, some in gaseous form, generate significant mechanical stresses. C. Degueldre et al. have succeeded in determining how irradiation affects a fuel pellet at the atomic level. The irradiation times considered were such that the pellet reached burn-up values beyond the average level encountered in Swiss nuclear plants. The nature of the chemical links between plutonium and its atomic environment was examined at various locations within the pellet using one of the finest analytical tools available: the µXAS beamline at PSI's SLS facility. First results already indicate that under irradiation plutonium had not undergone major changes in its chemical links (speciation). This not only confirms that the pellet had performed well in the reactor, but, equally important, indicates that the chemical valences that would make the plutonium more soluble in water, and thus more mobile under repository conditions, do not appear. This means that should the pellet be disposed of in an unchanged state at a repository site, and should water ever come to leach the pellet, the plutonium would still hardly dissolve in, or be entrained by, the water.

But what would happen to radionuclides if they were to be dissolved in pore water and transported by diffusion from such an underground repository? M. Bradbury et al. find that radionuclides with a variety of valences (II to IV) are sorbed strongly by the two clay barriers being considered in the repository concept of the Swiss National Cooperative for the Disposal of Radioactive Waste (Nagra): namely, the natural clay (Opalinus Clay) into which the galleries will be bored, and the engineered one (bentonite MX-80) used for filling the galleries once the waste containers have been positioned. The predictive capacity of the sorption model developed over the past few years, supported by an extensive thermodynamic database, reinforces our confidence that we can quantitatively predict radionuclide uptake in these complex geochemical systems.

Conclusions

Whether it be the failure of a steam generator tube, the response of plant operators to an emergency situation, or keeping fuel cladding at safe temperatures in a new reactor concept, our research teams have found that operational safety can be assured for both present and future reactors provided proper safety margins are built in, and that appropriate mitigation measures are implemented. Additionally, evaluation of the solubility of plutonium in water, and of the sorption capacity of a number of radionuclides by clay surrounds, has increased our confidence in the safety of repository systems.

In short, we are able to report here on scientific findings that support the statement that Light Water Reactors and clay repositories can be made safe. Of course, carefully chosen to be representative of the most important safety issues, our studies are not an exhaustive demonstration of the safety of all systems, under all conditions. Safety studies with short-term objectives, ever more detailed, and calling for ever more basic understanding, will remain a core activity of NES for many years to come.

Facts and figures for 2010

Peter Hardegger
Nuclear Energy and Safety Research Department, PSI

With optimisations in the cost structure, and increased efforts in the acquisition of third-party funding, we were able to compensate for increased costs and the 'hidden' inflation in the federal funds for nuclear research. As in previous years, the increase in third-party funding continued during 2010. Our biggest partner, Swissnuclear, presented its new research strategy during the year, and negotiations on the future structure of its research funding were initiated. On the personnel front, the turnover rate has come back to a stable level in 2010 from its highs in 2007 and 2008. The 18 new staff members arriving during 2010 exactly compensated those leaving, so that the total workforce of NES remained constant over the year.

The steep decline in federal funds for nuclear research in the 1990s has progressively been compensated as much as possible within NES by third-party financing. A stable funding situation was reached after the year 2000, with a small, and sustainable, growth being maintained since 2005. This trend has continued through 2010, the growth being a direct result of efforts made in the acquisition of additional third-party funding.

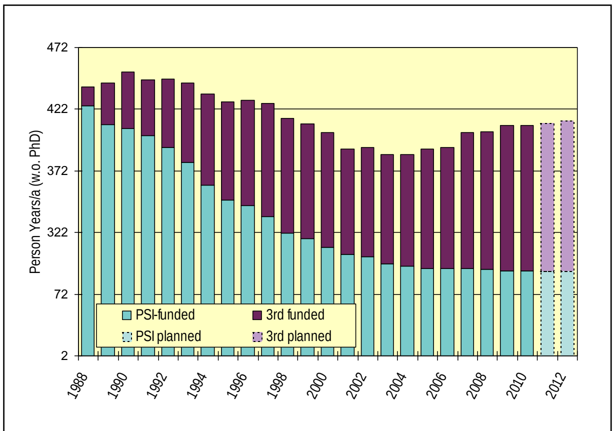

Figure 1: **NES funding resources 1988–2012.**

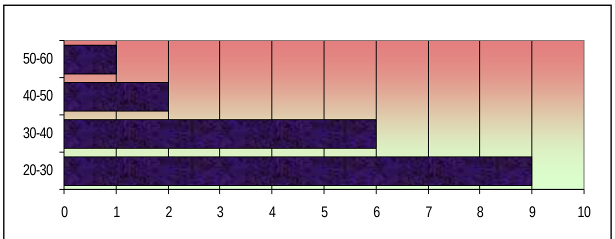

Figure 2: **Age breakdown of new recruits during 2010.**

The positive overall climate for nuclear research continued during 2010, and Swiss industry in general has fully recovered from the effects of the recent economic crisis. The employment situation within NES stabilised in 2010, fully consolidating the staffing levels.

Personnel

After three years in which more than thirty people were recruited each year, the recruitment rate came back to a normal level of less than 10% of the total staff complement per year in 2010. New staff members joining NES numbered eighteen, so those who left could be fully replaced, leaving only a small number of vacancies remaining open at the end of the year. In addition, the recruitment of young people has further improved the overall age structure within the department.
In total, the personnel situation in 2010 was very stable, providing those members of staff arriving in the last three 'hectic' years some breathing space for sustained integration.

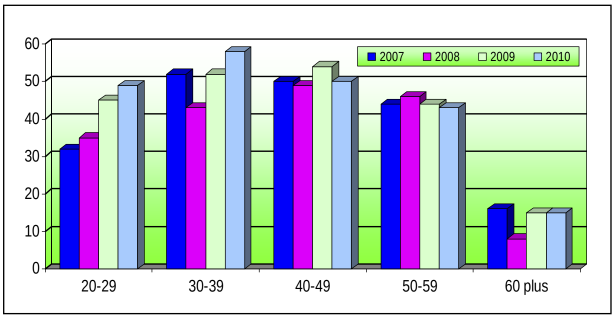

Figure 3: **NES age structure 2007–2010.**

Also, our young supervisory staff profited from this situation, further consolidating the positions of their groups and their lead role in the research activities.
Of the eighteen NES staff who left during 2010, nine were doctoral students or postdoctoral fellows who had successfully completed their terms of study. The 48 doctoral students and postdoctoral fellows in NES now represent 22% of our staff.

Finances

The funding of NES by PSI, which represents direct federal funding made available through the domain of the Federal Institutes of Technology (i.e. the ETH-domain), remained under pressure, and did not increase during 2010. As a consequence, inflation, and the rises in unit costs, could not be covered with respect to personnel outlay and the funds available for consumables and expenses.
The increased costs, including those needed to sustain the growth of the department, were compensated by the acquisition of new contracts, with existing and new partners, and with the renewal of existing contracts adjusted to the increased cost levels.
In regard to consumables, investments and maintenance, a total of 7.4 MCHF were made available: 1.7 MCHF from PSI and 5.7 MCHF from third-party funding.

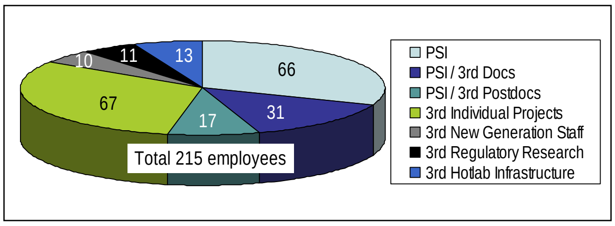

Figure 4: **NES staff salaries, showing the breakdown into PSI, third-party and joint funding.**

Partners and contracts

Safety Issues relating to the existing Nuclear Power Plants (NPPs) within Switzerland account for 40% of the NES budget, the prime emphasis being on research. It should be noted that research in support of the continuing safe operation of the Swiss NPPs remains a national and political responsibility for NES. The main partners in these activities are Swissnuclear (the association of operators of the Swiss NPPs) and ENSI (the Swiss Nuclear Safety Inspectorate). As a consequence, they remain two of the three largest third-party funding partners of NES.

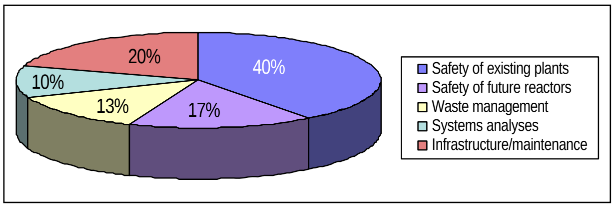

Figure 5: **Distribution of resources within NES during 2010.**

Safety-related issues for future reactor concepts are also an important research field fostering international collaboration and attracting young researchers to NES. Funding in this area presently amounts to 17% of the NES budget, and comes mainly from large international collaborations, such as EU and OECD projects, but also from small, specific contributions, such as the Swiss National Research Fund (SNF).
The growing public awareness of the nuclear waste disposal issue confirms the importance of this research area within NES, currently making up 13% of the total NES budget. The main funding partner is Nagra (the National Cooperative for the Disposal of Radioactive Waste), which is the third of the three principal funding partners of the department.
An important and growing field of research is Energy Systems Analysis, which is a joint activity between NES and the General Energy Department (ENE). The research in this area accounts for 10% of NES funding, somewhat less than in other areas, but one with a very high public visibility. Financing comes from diverse sources, including EU projects, federal offices (such as BFE and Bafu), ENSI, PSI-funded activities in other departments, and from numerous industrial partners.
In addition to the four principal research areas, NES also operates two large nuclear facilities: the Hot Lab, for the handling and examination of highly radioactive materials, and the PROTEUS reactor, in which experiments in reactor physics are conducted. The continued operation of these two facilities amounts to 20% of the total NES budget.

Many new contracts were finalised during the year. The most important of these were with ENSI, HRA and NORA. In addition, Nagra continued its funding in waste management research, and three new EU contracts were signed: ERCOSAM, LEADER and CatClay. Also, the infrastructure project with AREVA continues with a PhD programme, and a significant additional contribution to the CCEM project CARMA was signed with Swisselectric. The ecoinvent partners (ETHZ, EPFL, EMPA and ART) signed a cooperation contract, and, after long and intense negotiations with EMPA, a contract to operate the ecoinvent centre was also concluded. Finally, two new SNF projects were initiated, and, within the framework of our existing contracts, eleven work statements/orders were signed: two for HRA, two for STARS, and seven for the Hot Lab.

The strengthened role of NES in nuclear education

Rakesh Chawla[1], Horst-Michael Prasser[2]
[1] *EPFL and Nuclear Energy and Safety Department, PSI;*
[2] *ETHZ and Laboratory for Thermal Hydraulics, PSI*

Since its inception in 2008, the joint EPFL-ETHZ Master of Science Degree in Nuclear Engineering has received crucial contributions from NES towards the effort needed to cover the material. Starting from Sept. 2010, the curriculum for this highly inter-institutional, academic programme has been upgraded from 3 to 4 semesters. The additional semester is to be spent largely at PSI, thus further strengthening the role of NES in nuclear education within Switzerland.

The Swiss Master of Science in Nuclear Engineering [1] is the first, and (currently) the only, common degree offered jointly by the Swiss Federal Institutes of Technology in Lausanne (EPFL) and Zurich (ETHZ). From the beginning, NES has played a crucial role in the running of the programme by providing – through its on-going R&D activities – the framework for various Masters-level research projects. In addition, NES staff have served as lecturers for several of the core courses.
From Sept. 2010, i.e. for the 3rd batch of students, the curriculum has been upgraded from 3 to 4 semesters, the additional semester to be spent largely at PSI.

Experience with the first two batches

The initial 3-semester (90 ECTS*) curriculum was followed by the first two student batches: i.e. those who began in Sept. 2008 and Sept. 2009. Twelve students made up the first batch, and thirteen the second, representing a diversity of origin, just eleven in total originating from Swiss universities. Among the various students, the most popular bachelor-level degrees were in physics (11) and mechanical engineering (6).
Of the twelve students in the first (2008) batch, ten have either found employment in industry, or have embarked upon doctoral-level research programmes. One student dropped out after two semesters, and one has been delayed due to prolonged military service.
Of the thirteen students who started in 2009, one left the programme after his first semester. Most of the remaining twelve have now completed their course work and industrial internships, and are currently pursuing research projects within NES. For these students, as for the first batch, the research topics chosen correspond to a wide range of technical interests, as illustrated in Table 1.

	Research Topic
1.	Neutron assay of highly burnt fuel
2.	SCWR[a] design validation using PROTEUS data
3.	Monte Carlo calculations of RPV[b] fluence
4.	Analysis of the PHENIX pre-shutdown tests
5.	Neutron tomography of two-phase flows
6.	Experimental validation of CFD[c] modelling results
7.	Fission gas analysis via isotopic dilution
8.	Temperature evolution in a geological depository
9.	LCA[d] of waste disposal and CO_2 sequestration
10.	HRA[e] application to simulated emergencies

[a] Super-Critical Water Reactor; [b] Reactor Pressure Vessel;
[c] Computational Fluid Dynamics; [d] Life-Cycle Analysis;
[e] Human Reliability Analysis

Table 1: **A selection of the NE Master thesis topics.**

Upgrading of the curriculum

The first two years running the NE programme have shown that: (a) the course work, concentrated into two semesters, was too intense; and (b) the allotted 25-week period for the industrial internship, plus the Masters' research project, was too short. As a result, it was decided to introduce an additional semester, upgrading the curriculum to a 120 ECTS

* European Credit Transfer System [2]

programme. The additional semester is to be spent largely at PSI, and is intended to be used for:
- a semester project (shifted from the 2nd Semester, and with its weighting increased from 6 to 8 ECTS's);
- 4 core courses (3 of them new, and totalling 14 ECTS's; see Table 2), offered as block courses, i.e. in modular form, rather than spread over the semester;
- an industrial internship, rated at 8 ECTS's

The other curriculum changes are relatively minor: an "exchange" of semesters between the *Nuclear Fuels and Materials* and the *Radioisotope and Radiation Applications* courses, and an increase in the free electives from 4 to 10 ECTS's. The Masters' thesis at PSI – with its unchanged 30-ECTS weighting and 17-week duration – is moved to the 4th Semester.

Course	Responsible University	Course Type	Held at	ECTS
Radiobiology and Radiation Protection	ETHZ	Compulsory (new)	PSI	4
Adv. Topics in Nucl. Reactor Materials	EPFL	Core Elective	PSI	4
Nuclear Computations Lab	EPFL	Core Elective (new)	PSI	3
Beyond-Design-Basis Safety	ETHZ	Core Elective (new)	PSI	3
Total:				14

Table 2: **NE core (block) courses in the 3rd Semester of the new 120 ECTS curriculum.**

The shifting of the semester project from the 2nd to the 3rd Semester implies that the presentation of R&D activities by NES is to be made to the students at the end of the 2nd Semester, rather than in the 1st Semester.

Of the 3rd-Semester core courses (Table 2), *Advanced Topics in Nuclear Reactor Materials* is already one of the core electives in the 3-semester curriculum. The other three courses are really new. *Radiobiology and Radiation Protection* as a separate (8th) compulsory course fills a need felt for some time, the number of hours previously devoted to this subject within the *Nuclear Safety* course having been deemed insufficient. *Nuclear Computations Lab,* as a 3-ECTS core elective, will complement the compulsory courses *Special Topics in Reactor Physics* and *Nuclear Safety,* and will focus on the practical use of large computer codes for NPP neutronics and multi-physics analyses. Finally, *Beyond-Design-Basis Safety* will provide a more detailed description of the complex phenomena occurring during severe accidents, and the measures needed to limit their consequences. This important subject could not previously be adequately treated in sufficient detail under *Nuclear Safety* in the 3-semester curriculum.

Except in the case of *Radiobiology and Radiation Protection,* the main responsibility for the course work to be covered during the additional semester at PSI will lie with senior scientists within NES, over and above the department's existing involvement in the programme.

Effectively, the upgrading of the NE Master curriculum implies that the students will spend a complete academic year in NES, and not just a single semester.

Expected impact

The extension of the NE Master curriculum is clearly of great significance for the programme, since, as a (3+2)-year system, conformity has now been established with the Bologna Process [3], which in turn implies full compatibility with other major nuclear engineering programmes within Europe. With EPFL and ETHZ both being members of the European Nuclear Education Network [4], the potential for international collaboration has now been enhanced considerably, not only in terms of partner universities, but also in the context of industrial organisations, such as EDF and AREVA. This is expected to help stabilise the number of students joining the programme each year; a batch strength of about 20 being targeted for the medium term.

Acknowledgements

The authors acknowledge the invaluable contributions of the many persons within EPFL, ETHZ, PSI and the Swiss nuclear industry who have contributed to the success of the NE Masters' programme.

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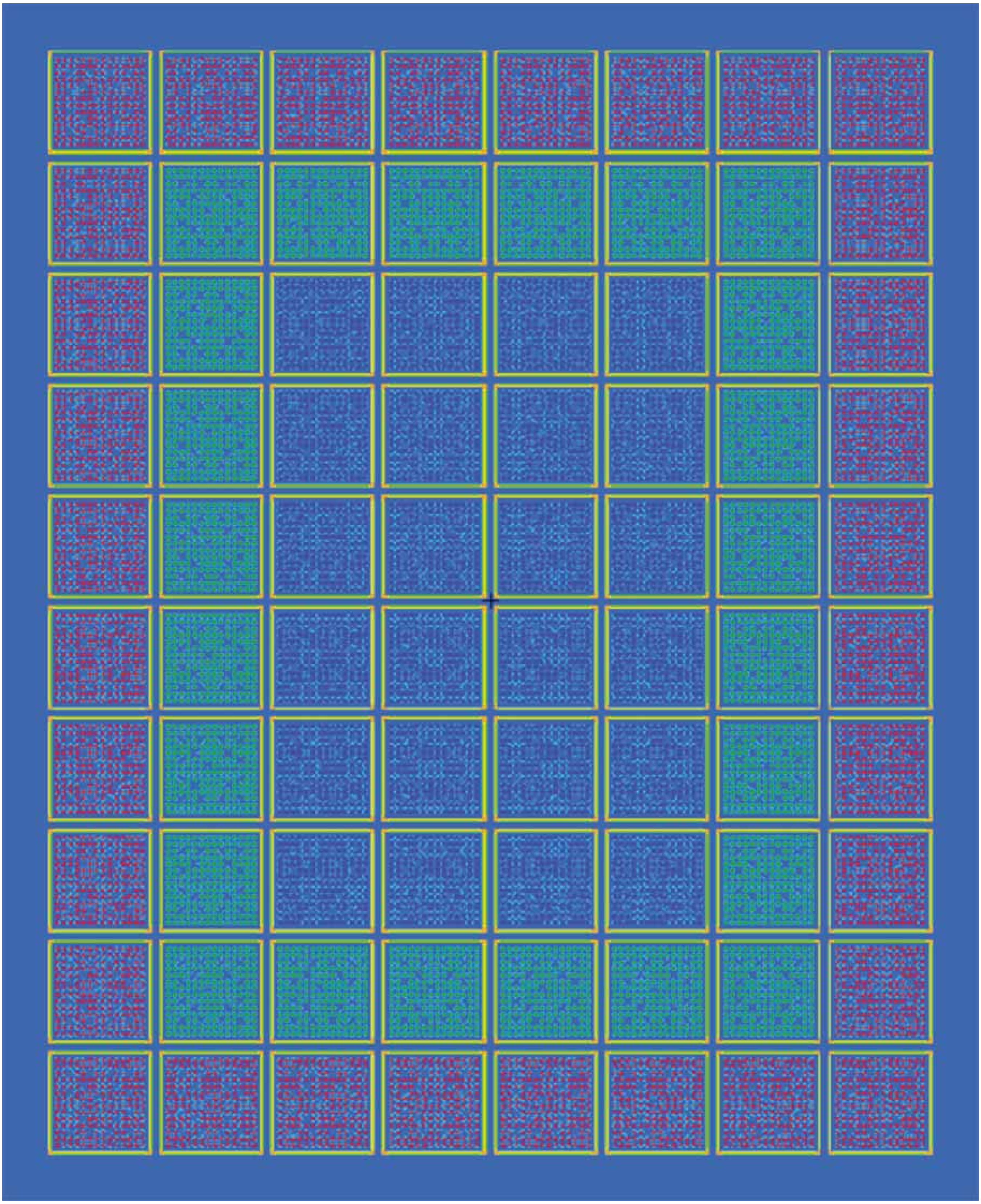

Laboratory for Reactor Physics and Systems Behaviour (LRS)

Current activities within LRS are centered on four principal project areas:

LIFE@PROTEUS

This project aims at improving measurements of fission-rate distributions across the interface between fresh and highly burnt fuel. The data will serve to validate modern reactor analysis codes, especially in regard to detailed pin-power calculations.

PROTEUS-UPGRADE

This project is concerned with the upgrading of the PROTEUS zero-power reactor to ensure containment of significant levels of radio-activity as a pre-requisite for the measurement of large quantities of highly burnt fuel. At the same time, requirements by the Swiss Regulatory Authority (ENSI) in respect to earthquake resistance and instrumentation and control are being addressed.

STARS

The STARS project aims at the development, maintenance and application of a comprehensive code system, together with the necessary input database, with the general goal of providing a safety assessment strategy for the Swiss nuclear reactors from “the pellet to the turbine”. Areas of particular focus include neutronics analyses of reactor cores (including fast-fluence evaluations and criticality safety assessments), coupled neutronics/thermal-hydraulics transient analyses of plant systems, uncertainty analyses for thermal hydraulics and neutronics problems, and fuel behaviour modelling. The STARS project cultivates the multi-physics aspects of plant safety assessment for LWRs.

FAST

This project is aimed at the development and implementation of a code system to represent state-of-the-art safety analyses of nuclear systems with fast neutron spectra (Gen IV reactors). Specialties here are the comparative analyses of gas-cooled, lead-cooled and sodium-cooled fast reactor systems, and the evaluation of iso-breeding equilibrium cores.

EDUCATION

The research projects within LRS provide motivation and subject material for numerous doctoral and masters-level theses in support of training in nuclear technology at EPFL and ETHZ. As such, the projects are strongly coupled with the joint EPFL-ETHZ Nuclear Masters Programme, the curriculum involving teaching assignments by senior LRS staff members.

◄ **Layout of a spent-fuel storage pool, used as the basis for a Monte-Carlo simulation of neutron transport characteristics.**

Power distribution in spent fuel – measurement technique and measurement station development

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In nuclear fission reactors, fuel enrichment and discharge burn-up have steadily increased over the years to improve the cycle length and reduce costs in an increasingly competitive electricity market. As a result, fuel assemblies with higher initial enrichment are loaded into the reactor core adjacent to highly burnt assemblies, an arrangement which induces very strong neutron flux gradients at the interfaces between them. PSI is engaged with swissnuclear in a project to better characterise these gradients through fission-rate measurements in a representative mock-up of fresh and spent-fuel Pressurised Water Reactor (PWR) assemblies in the PROTEUS zero-power reactor

LIFE@PROTEUS is a PSI-swissnuclear programme to study the reactor physics of high-burnup fuel in the PROTEUS reactor. Within the programme, large amounts of spent fuel are to be loaded into PROTEUS to study the behaviour at the interfaces between fresh and spent fuel PWR assemblies [1]. This ambitious project necessitates an upgrade of the PROTEUS reactor.

The interface will largely be characterised in terms of the radial pin-by-pin power rate distributions in the fresh and spent fuel following irradiation in PROTEUS. However, measuring the power in spent fuel, re-irradiated in a zero-power reactor, is a new concept. Consequently, during the period 2009–2010, two new general-purpose measuring techniques have been developed in PROTEUS. The techniques are based on measuring: (i) the delayed neutrons, and (ii) the high-energy gamma rays (> 2200 keV) emitted by newly generated, short-lived fission products.

Development of the measuring techniques

The details of the new techniques, and their first results, have already been documented [2] in the context of the WOLF-B experiments, in which fresh and spent fuel segments (36, 46 and 64 MWd/kg) were arranged in a SCWR*-type lattice in PROTEUS and irradiated. The fission-rate ratio between the fresh and spent fuel segments was determined using the two techniques defined above.

In 2010, the techniques were further refined, results compared against predictions made using the monte-carlo code MCNPX, and the impact of the statistical and nuclear data uncertainties on the total uncertainties of the fission-rate ratio between fresh and spent fuel evaluated. Good agreement between experiments and calculations was obtained, with typical 1σ (one standard deviation) uncertainties for fresh-to-spent fission-rate ratios being in the region of 2–3% and 3–5% for the delayed-gamma and delayed-neutron techniques, respectively [3,4]. For the delayed neutron technique, all calculations agreed with experiments within 1–2%, which is much less than one standard deviation. This suggests that uncertainties in the nuclear data for the delayed-neutron technique (delayed neutron yields and relative abundances) are over-conservative.

Measurement station design

For the next phases of the LIFE@PROTEUS programme, with the introduction of fresh and spent PWR fuel pins into PROTEUS, a new measurement station is planned, to be positioned on top of the test zone. The fuel pins will be accessed from above, and moved upwards through a stainless steel guide tube into the measurement position. The neutron or gamma detector will then be embedded in a larger tube adjacent to the pin (Fig. 1).

The radial and angular positions of the measurement station will be adjusted to access the different pins in the test zone. For each measurement, the detection geometry will be kept constant. The minimum time required to move a pin into the

* Super-Critical Water Reactor

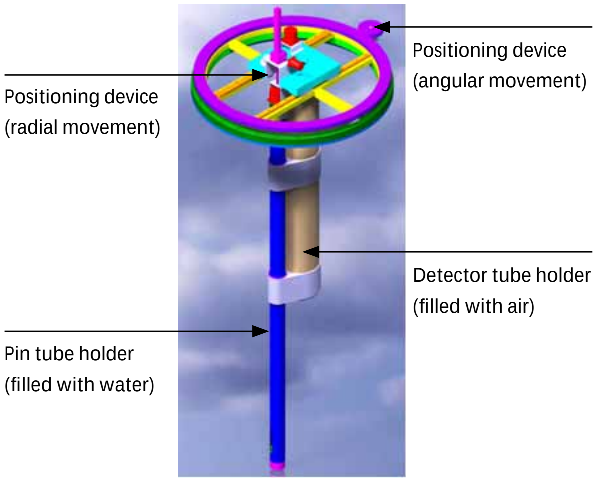

Figure 1: **3D view of the planned measurement station in PROTEUS.**

measurement position is estimated to be about one minute: sufficient for measuring the delayed-neutron signals, but long enough not to impose undue mechanical stresses on the pin.

A preliminary assessment of the geometric and shielding arrangement for the detector has been made, optimised to improve the signal-to-noise ratios and uncertainties in the measurements, and to maximise the life-span of the gamma-ray detector. The performance of the measurement station in detecting gamma-rays from fission products has been quantified using data from the WOLF-B experiments and MCNPX simulations (Fig. 2).

Conclusions

First results indicate that the germanium detector is adequately shielded against neutrons emitted from the core, and that the dead-time is limited to a reasonable value of 30%. Finally, it has been shown that the spent-to-fresh, fuel-fission-rate ratios for each pin could be measured with a 1σ uncertainty of around 1.5% for three irradiations of three hours duration at 800 W.

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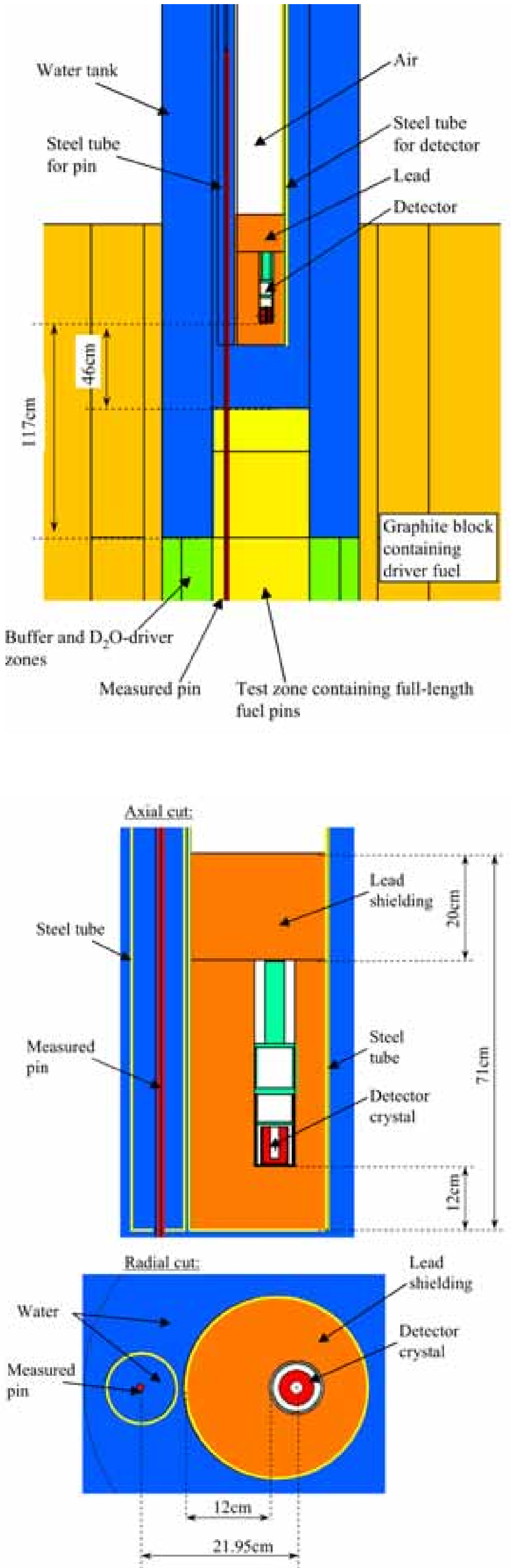

Figure 2: **Axial (top) and radial (bottom) views of the MCNPX model of the planned measurement station in PROTEUS.**

Safety analysis of the EPR^{TM} reactor within the STARS project

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The EPR^{TM} is a Generation-III Light Water Reactor which is among the candidates being considered for possible new reactors in Switzerland. In 2007/2008, the Finnish regulatory body (STUK) began a collaboration with PSI to conduct an independent safety assessment of the EPR^{TM} currently under construction in Finland. At the present stage of the collaboration, detailed plant/system and fuel models have been developed and, on this basis, preliminary Large-Break Loss-Of-Coolant Accident (LB-LOCA) analyses performed using the TRACE and FALCON codes. Results indicate that the response of the EPR^{TM} to this type of event is satisfactory.

The EPR^{TM} (Evolutionary Pressurized Water Reactor) belongs to the third generation of Light Water Reactor (LWR) designs. These designs are foreseen to constitute the new fleet of commercial reactors in countries where the nuclear energy option will remain, or become, one of the national energy options for the 21st Century. The principal feature common to all third-generation LWR designs is that primary focus has been given to enhanced safety performance: i.e. a significant reduction in the probability of occurrence of severe accidents (core melt), either through increased redundancy and diversification of the active safety systems (GIII 'evolutionary' designs), or through the introduction of passive safety systems relying mainly on inherent physical processes such as buoyancy and condensation (GIII+ "revolutionary" designs). Within the European landscape, GIII EPR^{TM}-type reactors are currently under construction in both Finland and France, and the EPR^{TM} is also being considered as a candidate by the national utilities responsible for new-build projects in Switzerland.

EPR^{TM} modelling at PSI

The mission of the STARS (Swiss Transient Analysis of Reactors in Switzerland) project [1] within LRS is to develop advanced multi-physics, multi-scale, "from Turbine-to-Pellet" (i.e. plant/core/fuel), computational methodologies for the deterministic safety analysis of Swiss reactors. Although the major focus is on the existing Swiss reactors, a collaboration was launched in 2007/2008 with the Finnish nuclear safety inspectorate (STUK) to produce independent safety evaluations of the EPR^{TM} currently under construction at the Olkiluoto site in Finland. For the STARS project, the collaboration was considered to be particularly valuable in regard to developing in-house GIII/III+ expertise. The first phase of the collaboration has recently been completed, with simulation models developed using, as basis, detailed EPR^{TM} drawings/data (see Fig. 1). Among these, a comprehensive plant/system model (i.e. involving vessel, steam generators, steam lines, pumps, control and safety systems) has been developed using the TRACE best-estimate code, aimed at studying the evolution of flow, pressure, temperature and energy distributions in the plant during a postulated accident sequence [2-4].

Complementary to the TRACE description, a comprehensive solid CAD model has been developed and used to generate a detailed mesh for the STAR-CD(CCM+) Computational Fluid

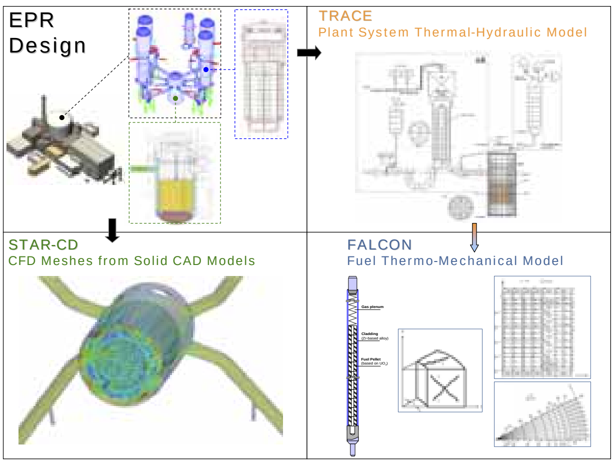

Figure 1: **EPR^{TM} Modelling in STARS.**

Dynamic (CFD) code [5]. In addition, very recently, a prototypical EPR™ fuel-rod model has also been developed using the FALCON Finite Element Method (FEM) code in order to study the thermo-mechanical behaviour of the fuel pellet, gap and cladding arrangement of the EPR™ fuel.

LB-LOCA simulations

The LB-LOCA scenario remains one of the classical Design Basis Accident (DBA) for LWR reactors. In regard to the EPR™, preliminary TRACE simulations have been carried out assuming a large break in one cold-leg of the primary circuit. The objective here was to verify if core coolability could be maintained during and after intervention of the Emergency Core Cooling (ECC) systems during reflooding of the reactor, assuming, in this context, availability of all accumulators and 2 (out of the 4) safety injection systems.

Core coolability is assessed by calculating the Peak Cladding Temperature (PCT) for all the fuel rods in the core, and verifying that it remains below the stipulated safety criterion of 1477 K. This limit, in combination with a criterion based on maximum allowable cladding oxidation, was originally established with the objective of ensuring there would be no embrittlement of the cladding (the first barrier for radioactivity release to the environment), and to guarantee thereby that the fuel rod would not lose its structural integrity in the case of thermal shock during and/or after the reflooding phase. As can be seen in Fig. 2, which shows the time evolution of the cladding temperature at different axial core elevations, a PCT of just above 1000 K is predicted about 59.2 seconds after the break.

Fuel behaviour analyses have subsequently been conducted on this basis [6]. To ensure appropriate fuel rod conditions (gap conductance, amount of fission gas, internal rod pressure) at the start of the transient, steady-state (base) irradiation calculations were first performed using the FALCON code coupled with PSI models for fission-gas release and gaseous swelling. Subsequently, transient FALCON analyses were performed using, as additional boundary conditions, the cladding surface temperature and the coolant pressure predicted by TRACE. Results indicate that the PCT would remain well below the cladding rupture temperature ($T_{Rupture}$), as calculated by the code as a function of the cladding differential pressures (as monitored by the hoop stress) using an empirical correlation validated from experiments. This is illustrated in Fig. 3 for the axial elevation, for which the minimum difference, $T_{Rupture}$-PCT, was obtained. As can be seen, the FALCON results indicate that a comfortable margin of about 80K remains before clad rupture. This margin is mainly due to the low PCT value predicted by TRACE during the transient, believed to be a direct consequence of the design enhancements made during the development of the EPR™ reactor.

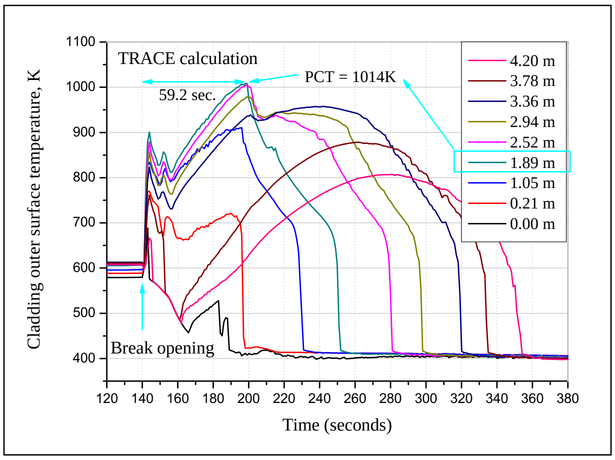

Figure 2: **Results of TRACE EPR™ LB-LOCA analysis.**

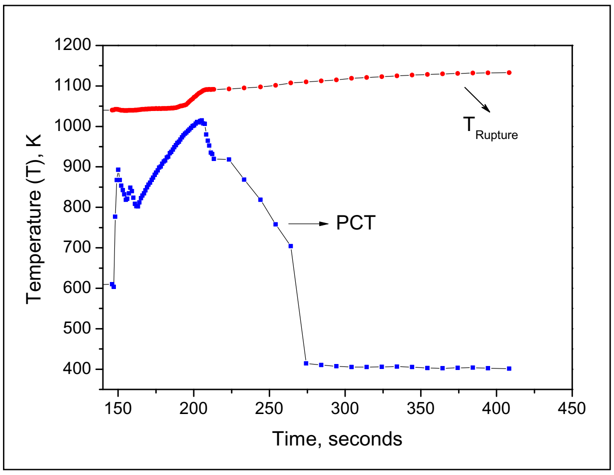

Figure 3: **Results of FALCON EPR™ LB-LOCA analysis.**

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Computational analysis of selected Phénix End-of-Life tests with the FAST code system

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Selected experiments performed in the frame of the Phénix End-of-Life (EOL) programme have been used to further validate the FAST code system. In particular, appropriate ERANOS and PARCS/TRACE models have been developed for the analysis of the Control Rod Shift (CRS) test. Comparing calculated results with the experimental data, it has been demonstrated that the FAST code system can accurately simulate the control rods and the core power distribution. A Natural Convection (NC) test has also been studied, using a Phénix primary circuit model for use with the TRACE thermal-hydraulics code. Predicted temperature fields from the model compare well with the experimental data, thus confirming that the FAST code system is able to simulate the establishment of natural circulation following a pump trip in a sodium-cooled fast reactor.

Phénix is a prototypic-scale (250 MWe) pool-type, sodium-cooled fast breeder reactor [1], which operated for over 35 years in France. Prior to its final shut-down in 2009, the Commissariat à l'Energie Atomique (CEA) performed a series of End-of-Life (EOL) tests. Two of these tests, namely, the *Control Rod Shift* (CRS) and *Natural Convection* (NC) tests, were chosen for analysis in two IAEA* Coordinated Research Projects (CRPs).

PSI participated in these two benchmarks in the framework of the FAST project, for which numerical analyses were undertaken of: (i) asymmetrical control rod (CR) configurations and, through sodium heating measurements, the impact of their positions on the local power distribution; and (ii) the establisment of natural circulation in the primary circuit following a pump trip. By comparing calculated and measured data, the tests aim to improve the neutronics and thermal-hydraulics models in system codes.

The FAST group at PSI are among seven international participants in the two CRPs. By participation in these tests, PSI gain access to very important plant data which can be used to (further) validate the FAST code system [2] for the study of Sodium-cooled Fast Reactor (SFR) systems.

A schematic of the Phénix primary system is shown in Fig. 1. The configuration consists of a large sodium pool containing the reactor core. Attached are three secondary sodium loops, each with one primary pump and two Intermediate Heat Exchangers (IHXs). The Phénix core consists of 110 hexagonal mixed-oxide (UO_2-PuO_2) fuel assemblies with 23% to 28% plutonium content.

* International Atomic Energy Agency

Analysis of the Control Rod Shift test

The power distributions in the core corresponding to the different CR positions during the test were calculated using the neutronics code ERANOS [3] together with a coupled 3-D neutronics/thermal-hydraulic model based on PARCS/TRACE [3,4]. The results of these blind calculations are presented in Fig. 2. As can be seen, results match very well the experimental data. The coupled model has then been applied beyond the benchmark requirements to study 3-D thermal-hydraulics effects and the transient behaviour of the core, each aimed at improving general understanding of SFR behaviour under the imposed conditions.

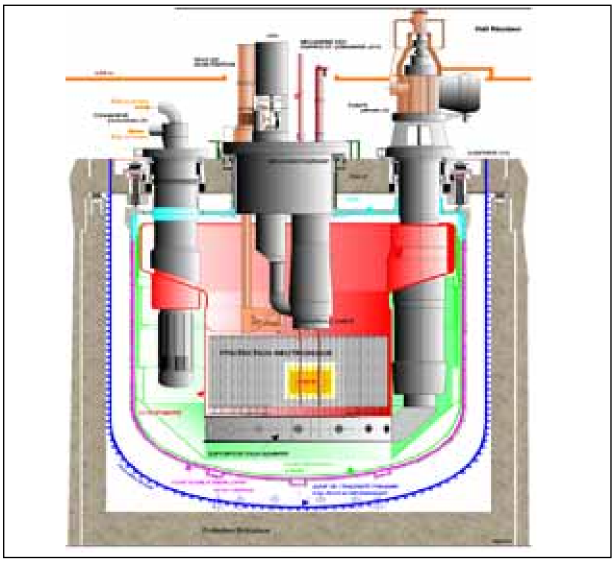

Figure 1: **Global view of the Phénix primary system.**

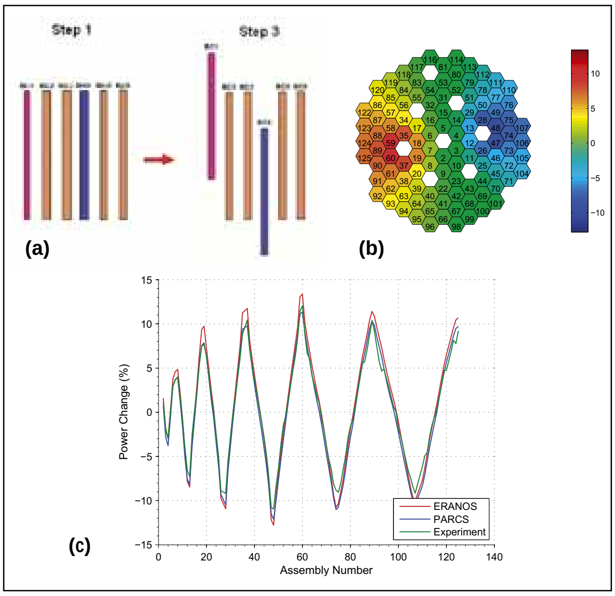

Figure 2: **Results from the ‘blind’ CRS simulations: (a) change in the CR positions; (b) map of the resulting power deformation; and (c) assembly-wise comparison of the calculated and measured power changes.**

Analysis of the Natural Convection test

Starting from a steady-state condition at reduced power (120 MWth) with two primary pumps in operation, the NC test was initiated by dryout of two of the four operating steam generators (SGs). The resulting cooling deficiency led to a reduction of the temperature difference between the primary and secondary sides of the IHXs. When this temperature difference reached 15K (after about six minutes) a scram was manually initiated and the primary pumps shut down.

To analyse the test, a TRACE model of the primary circuit was constructed (Fig. 3) using appropriate boundary conditions for the evolution of the secondary circuit temperature and core power.

Figure 4 shows comparisons between a *blind* calculation of the temperature in the primary circuit and the experimental data as a function of time. Largest discrepancies are seen at the beginning of the transient. These are considered to be a consequence of not taking into consideration the initial thermal inertia of the reactor structures. In addition, a fundamental limitation of the I-D TRACE model has been exposed, in that it is not capable of representing the strong temperature stratification in the pool. Nevertheless, the reactor behaviour and trends in the temperature plots match very well with the measured data towards the end of the transient, so that overall the TRACE predictions are considered satisfactory.

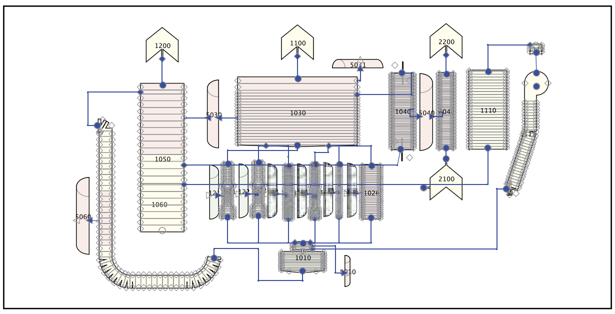

Figure 3: **TRACE model of the Phénix primary circuit.**

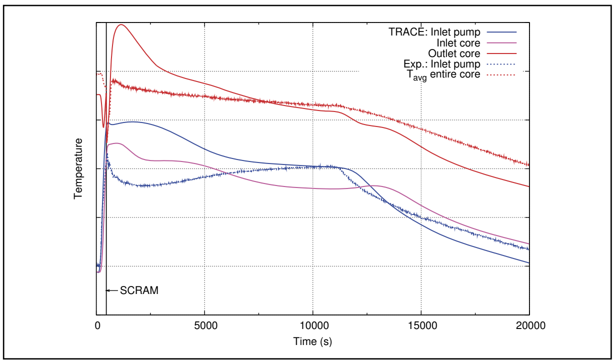

Figure 4: **Evolution of the primary circuit temperatures: a comparison between blind predictions and experimental data.**

Conclusions

The analyses presented here constitute an important further step in the validation process of the FAST code system. Participation in the IAEA Control Rod Shift benchmark constitutes a reference calculation for modelling SFR control rod assemblies in ERANOS, and has validated the neutronics modules (ERANOS/PARCS) in the FAST code system against (static) reactor data. The establishment of natural circulation in the primary pool following a pump trip has been successfully predicted by the TRACE model, which was specifically constructed for the present study. However, comparisons between calculational results and test data in this case have also highlighted some discrepancies which represent important challenges for future SFR transient analyses.

Acknowledgements

The authors sincerely thank the CEA for releasing the experimental data of the CRS and NC tests, and the IAEA for organising the CRPs.

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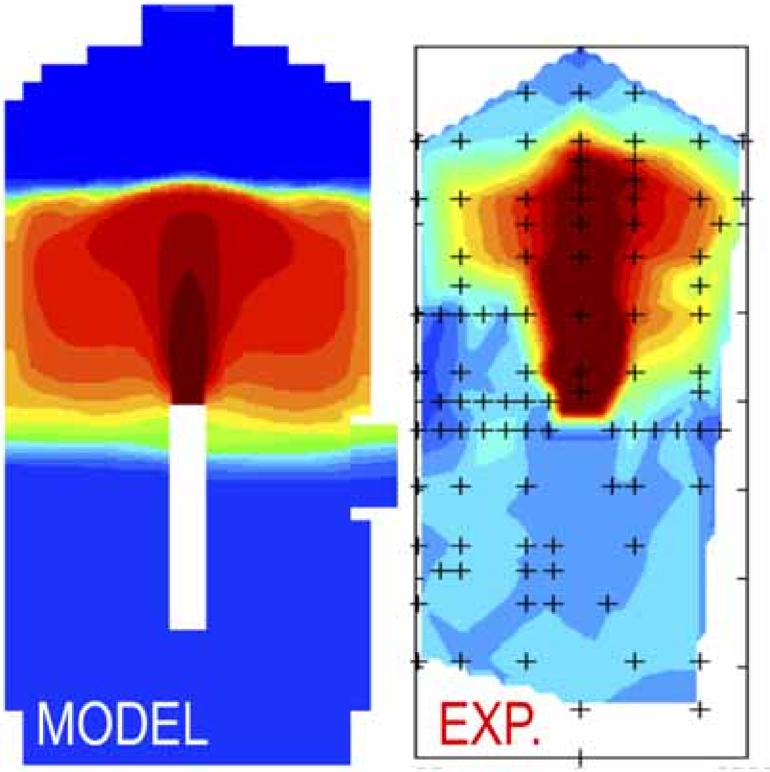
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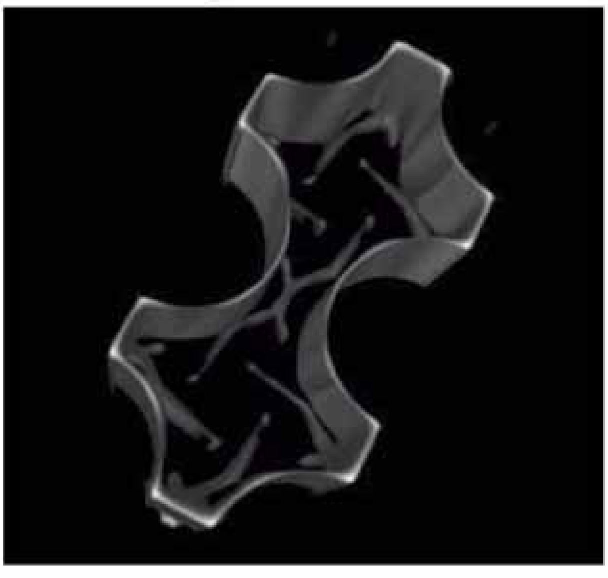

Laboratory for Thermal Hydraulics (LTH)

The Laboratory develops, validates and applies simulation tools at the cutting edge of modelling technology, with a focus on multi-phase and multi-component flows. Efforts focus on boiling and condensation heat transfer, turbulent mixing, aerosol physics and iodine chemistry. The work is oriented towards transition from empirical to fundamental modelling.

Efforts fall broadly into two main project areas.

ALPHA
The project addresses safety-relevant, thermal-hydraulics issues common to Generation II, III and IV Nuclear Power Plants (NPPs) in respoect to accident situations, efficiency and plant life-time management. Originally, work focused on the passive safety systems of innovative Light Water Reactors (LWRs). More recently, the experimental base has been broadened to encompass fundamental phenomena in primary circuits and containments. The PANDA facility, which remains the experimental backbone of LTH, has evolved into one of the best instrumented and most versatile containment test facilities in the world. Currently, PANDA is being used to study hydrogen generation, its accumulation and removal from the containment atmosphere. A number of small-scale, special-effects tests complement the integral PANDA facility by providing essential validation data for the Computational Fluid Dynamics (CFD) activity within LTH, including the development of innovative measuring techniques, such as tomography using cold neutrons, and a feasibility study on the use of fast neutrons for flow imaging.

Source Term Evaluation
Activities here are centred around the ARTIST test facility, providing unique data on aerosol transport and retention during severe accidents, particularly on aerosol retention in Steam Generator Tube Rupture (SGTR) scenarios. Studies on iodine chemistry have resulted in novel, highly efficient methods being proposed for filtered venting of containments. Findings from the tests have been incorporated in state-of-art, severe-accident analysis codes, such as MELCOR.

◀ **Advanced experiments in the field of nuclear safety: large-scale containment tests on hydrogen flow dynamics (top), the behaviour of liquid films in BWR fuel rod spacers (bottom left), and aerosol deposition in a PWR steam generator tube bundle (bottom right).**

SUBFLOW: high-resolution liquid mixing and two-phase flow studies in a 4×4 rod bundle

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The SUBFLOW experimental facility is providing valuable data for the characterisation of flow mixing in a rod-bundle geometry. Central to the measurements is use of the wire-mesh sensor technique, which enables the details of the flow characteristics to be portrayed simultaneously over an entire cross-section of the flow. For single-phase situations, electrical conductivities are measured to monitor the degree of inter-channel mixing, while under two-phase flow conditions, the conductivity measurements reflect the void-fraction distribution. The data are valuable for the validation of Computational Fluid Dynamics (CFD) codes.

Ability to predict flow behaviour in fuel rod bundles is an important consideration in the safety and efficiency of the power production in the cores of nuclear plants. A common approach to the research being conducted in this area is study of two-phase flow in a rod bundle under adiabatic and atmospheric test conditions. Typical instrumentation used for void fraction measurements are multiple-tip probes, a method based on the optical or electrical properties of the coolant material. However, in order to determine the void fraction distribution over an entire cross-section of the flow, such a probe has to be moved transversely, and the measurements taken in different locations. An effective alternative is to use wire-mesh sensors (WMSs).

The main benefit of the SUBFLOW experiments is the application of the WMS measurement technique, previously developed for open-channel flows [1], to a rod-bundle geometry, to study both cross-mixing of single-phase coolant between parallel sub-channels and two-phase flows.

Wire-mesh sensors consist of two sets of parallel wires placed perpendicularly to each other at a small distance (~ 2 mm) apart. The conductivity of the fluid is then measured at each cross-over point with high time resolution, providing two-dimensional images of the flow characteristics over an entire cross-section. This makes possible a new quality of information for code validation purposes.

Single-phase cross-mixing tests

Single-phase, cross-mixing experiments constituted the first test series in the SUBFLOW experimental campaign [2]. A schematic of the test rig is given in Fig. 1. The flow of water is upwards, with flow mixing in the channel monitored by injecting a salt water tracer into the main flow, the differences in conductivity being measured by a WMS located at the top of the test section. To characterise turbulent mixing with distance, 10 injection capillaries are placed at different

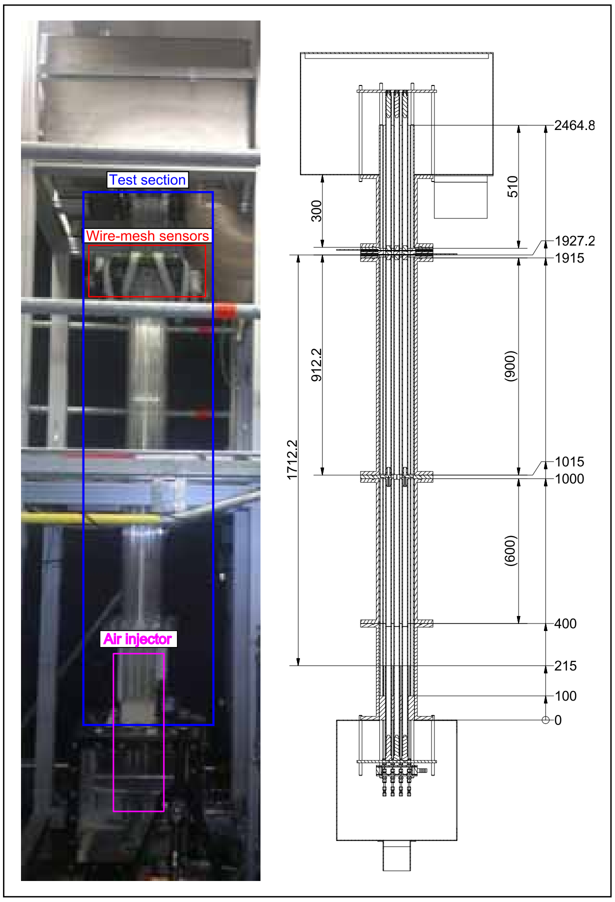

Figure 1: **Overall view of the SUBFLOW test section.**

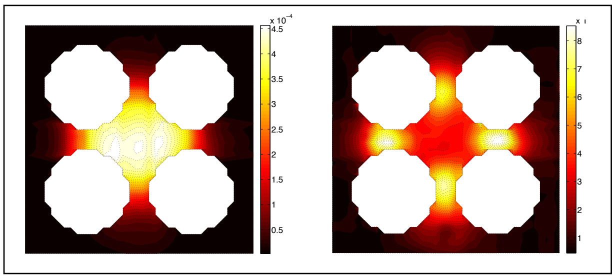

Figure 2: **Sample results from a single-phase mixing test: time-averaged concentration distribution (left) and standard deviations of the concentration (right).**

elevations beneath the WMS. The sensor has a spatial resolution of 2.125 mm and a measurement frequency of 2.5 kHz. Of the possible 64×64 cross-over points of the wires, some lie inside the rods, so that conductivity measurements are actually available at 2304 positions in the plane of the sensor. Using calibration values, the measured conductivity is converted into a dimensionless scalar, its value representing the degree of mixing taking place (1 for pure tracer, 0 for no tracer). Time-averaged values of the normalised concentration (left) and its standard deviation (right) are presented in Fig. 2 for a selected test with tracer injection 1295 mm below the WMS. The plots show that at this distance downstream of the injector the tracer has spread everywhere in the sub-channel, but also that mixing has already begun in the parallel sub-channels. Standard deviations are a maximum in the rod gaps, where flow mixing is most intense.

Two-phase flow tests

The wire-mesh-sensor technique can also be employed under two-phase flow conditions. Since all measuring points in the plane of the sensor are scanned quasi-simultaneously, and at high frequency, measured signals can be converted to represent distributions of the gaseous phase, i.e. bubbles, since they are essentially non-conducting. In addition, with two WMSs placed a short distance apart, it is possible to estimate the velocities of the bubbles using cross-correlation methods. A large body of information, such as bubble-size distributions, can be obtained in this way.

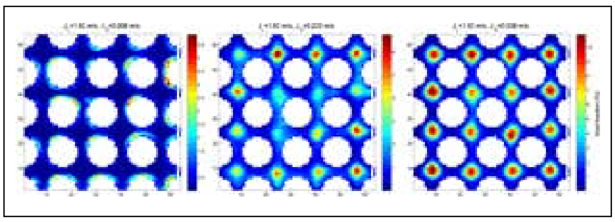

Figure 3: **Time-averaged void fraction distributions from a series of two-phase flow tests.**

Time-averaged void fraction distributions from a set of two-phase flow tests are presented in Fig. 3. The results show how air bubbles form wall peaks for the lowest air flow rate (J_G) case (left), and central peaks for the highest (right). The reason for this behaviour is the change of sign of the lift force on the bubbles, a sideways force acting perpendicularly to its main flow direction [3]. Due to this force, bubbles with diameters smaller than a critical value (~5.8 mm under atmospheric conditions) travel towards the walls, while larger bubbles congregate in the central regions of the flow. The bubble injector used in the SUBFLOW tests is designed in such a way that the initial bubble diameter can be varied, while keeping the air and water flow rates the same.

Data for CFD code validation

Results from the single-phase, cross-mixing tests have already been used for the validation of the in-house CFD code PSI-Boil [4]. Good agreement between measured values and code predictions has been achieved. Two-phase flow data from SUBFLOW could also be used for the validation of CFD codes which incorporate models for bubble-size classes, since the void fraction distributions measured in SUBFLOW can be grouped according to bubble diameters.

Future prospects

During 2011, the SUBFLOW facility will be used to study the effect of fuel rod spacers on the flow. The experiments aim to quantify the effect of their presence on liquid mixing, bubble break-up and coalescence, and void fraction distributions in general. Results will be useful for the optimisation of rod-spacer design.

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Bubbly-flow simulations with high accuracy using an extension of the CIP-CSL2 method

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At present, thermal-hydraulic studies of nuclear reactor safety rely largely on system codes, in which approximations such as the two-phase volume-averaged model are employed. The models are appropriate in estimating global quantities, but in order to improve accuracy where it is needed more precise models, not based on coarse-mesh approximations, have to be employed. We report here the development of a highly accurate interface tracking method which explicitly calculates moving phase boundaries. The advantages of the method are that mass conservation is strictly preserved, and the interface shape is exactly captured. The effectiveness of the method is demonstrated through a number of validation tests for bubbly flows.

To simulate boiling flow in fuel bundles precisely, without using coarse-mesh approximations such as the two-phase, volume-averaged model, a surface-tracking model should be used together with the conservation equations and a phase-change model. In this paper, a recently developed surface-tracking model is briefly described, and its use illustrated by means of sample calculations for bubbly flows.

The shape of a bubble in a continuum liquid is determined from a balance between buoyancy, viscous and surface tension forces. These forces need to be computed accurately in order to reproduce the correct bubble shape, and its evolution with time. In addition to this, mass conservation must be ensured to high accuracy, otherwise the bubble volume changes, and errors are produced in predicting its shape and rise velocity.

To fulfil these requirements, we have developed a new surface-tracking model which couples the **C**onstrained **I**nterpolation **P**rofile–**C**onservative **S**emi-**L**agrangian (CIP-CSL2) method [1] with an interface-sharpening algorithm [2].

Numerical method

In common with other interface-tracking schemes, CIP-CSL2 utilises a 'colour function' f to delineate the regions occupied by liquid and gas (or vapour). In infinitesimal form, $f = 1$ in the region occupied by the gas, and $f = 0$ in the liquid. As the flow field evolves, f changes as a function of space and time, and a transport equation for the colour function needs to be solved. The CIP-CSL2 method uses the derivative of the colour function in addition to its scalar value within the solution procedure. The combination results in low diffusion and enhanced numerical stability. Nonetheless, in the discretised form of the equations, in which f represents the volume fraction of gas in a particular control volume, the colour function distribution tends to smear out as time advances. To prevent this, we have coupled an interface-sharpening algorithm with CIP-CSL2. The coupled approach involves solving an additional 'sharpening equation', but in such a way as to satisfy mass conservation exactly in a cell-wise sense.

Condition	σ	(N/m)	0.072	0.072	0.072	0.072	0.032
	ρ_l	(kg/m3)	1350	1350	1350	1350	876
	μ_l	(Pa·s)	2.73	2.04	1.28	0.775	0.118
Computed							
	$V_{term.}$	(m/s)	0.22	0.25	0.28	0.30	0.22
Experiment							
	$V_{term.}$	(m/s)	0.19	0.21	0.26	0.29	0.22

Figure 1: **Shapes and terminal velocities of rising air bubbles in a liquid pool for different surface tensions (σ), liquid densities (ρ_l) and liquid viscosities (μ_l).**

This new surface-tracking algorithm has been implemented into the in-house Computational Fluid Dynamics (CFD) code PSI-Boil [3]. The code, which is able to simulate fluid flow as well as heat and mass transfer phenomena, employs a staggered finite-volume approach similar to that used for system codes, but at a micro/meso-scale rather than at the macro-scale.

Simulations

By way of illustration of the method's potential, three bubbly-flow simulations are reported here. The first example features a single air bubble rising in a stagnant liquid pool for varying viscosity and density ratios. Figure 1 shows qualitative comparisons of computed bubble shapes against experimental observations [4-5], and quantitative comparisons of bubble terminal velocities. Good agreement with the measured quantities is seen in all cases.

The second example consists of a single air bubble rising in stagnant water. Here, the emphasis is on terminal velocity as a function of bubble diameter (and shape, as seen in Fig. 2), measured in different experiments [6]. The computed terminal velocities are shown as red circles. The peak around diameter 1 mm is well captured by the computations, and the terminal velocities for larger diameters agree exactly with the experiments. However, for bubble diameters smaller than 1 mm, terminal velocities are over-predicted by the model. This discrepancy is considered to be a consequence of bubble surface contamination present in the experiments. These contaminants obstruct the water flow over the bubble surface, resulting in no circulation being generated inside the bubble. In contrast, in the computations, contaminants are not modelled, and a circulation is predicted inside the bubble, driven by the surface shear.

The third case considers multiple bubbles rising in a vertical channel filled with water. Fifty air bubbles are initially placed in the water in a random manner, as shown in Fig. 3a; the diameter of each (spherical) bubble being set at 5 mm. The computational domain has dimensions 40×40×80 mm in length, breadth and height, respectively. The side boundaries are modelled as rigid walls with zero-slip, while the top and bottom boundaries are assumed to be periodic. Computed results are shown in Figs. 3b to 3d. Initially, the bubbles start to rise under buoyancy, deform, and sometimes coalesce. As a consequence, larger bubbles are formed, and the total number of bubbles decreases. Although results cannot be compared against experimental data in this case, the calculation demonstrates the robustness of the method, and the ability to compute flows with many bubbles simultaneously.

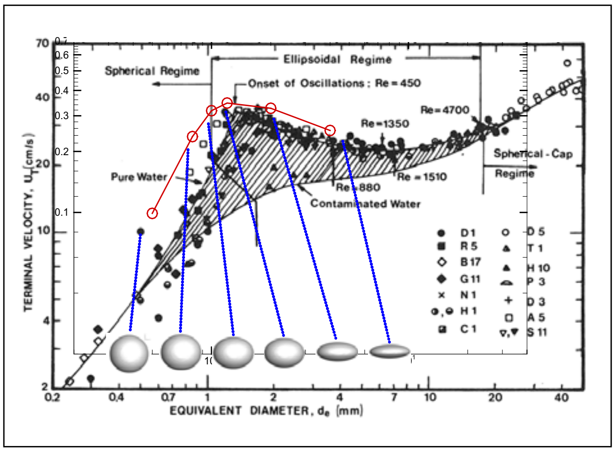

Figure 2: **Terminal velocity of air bubbles rising in water. Red circles represent computed values.**

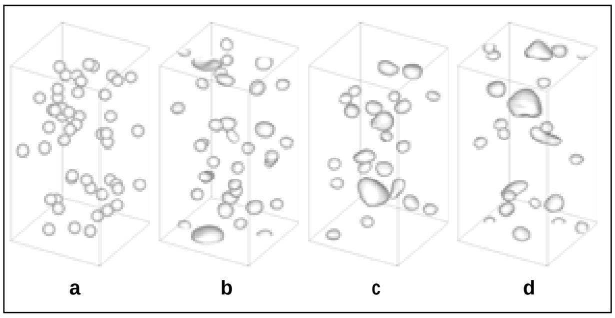

Figure 3: **Multiple-bubble simulation at successive times: (a) t=0; (b) t=0.12; (c) t=0.24; (d) t=0.36. All times in seconds.**

Conclusions

A new interface-tracking algorithm has been developed within the in-house CFD code PSI-Boil and applied to a number of bubbly-flow configurations. The advantages of the method are: (1) exact mass conservation, and (2) high precision interface capture. Comparisons have been made against experimental data for single bubbles rising in stagnant oil and water baths. The simulation of multiple bubbles demonstrates the ability of the method to compute bubbly flows with coalescence.

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Improved retention of radioactive aerosols during a steam generator tube rupture

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A Steam Generator Tube Rupture (SGTR) in a Pressurised Water Reactor (PWR) may lead to release of radioactive aerosol particles to the environment. Though an event of very low probability, its significance lies in the potential to bypass the containment barrier, thus providing a direct pathway from the (radioactive) primary circuit to the (normally inactive) secondary circuit, and thence to the environment. It is shown in this work that if the leak is covered with water a large proportion of the aerosols would be retained, and the potential release of radioactivity to the environment substantially reduced. Consequently, active flooding of the secondary side of the steam generator represents a powerful accident management procedure, allowing reactor operators to mitigate the consequences of such an accident. However, no reliable models exist to quantitatively describe the aerosol retention mechanism, which has then to be determined experimentally. The ARTIST facility at PSI supplies unique data in this regard.

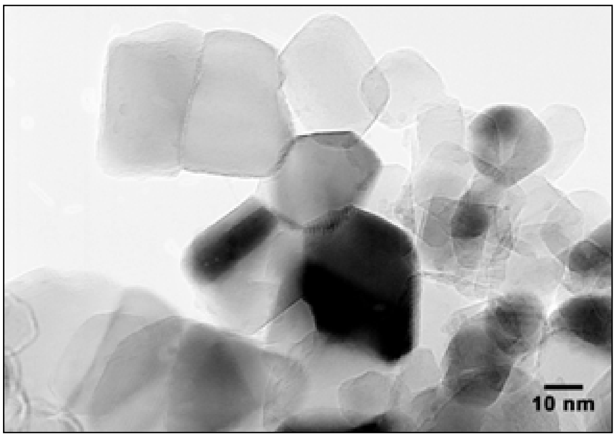
In the case of an SGTR in a PWR, radioactive aerosols released from the reactor core can pass to the secondary circuit, thereby creating a direct pathway to the environment, bypassing the engineered containment barrier. The steam generator (SG) offers ample surface area upon which fission products may be deposited, such deposition having the potential to significantly reduce the final emission to the environment. However, due to the complicated physical processes occurring, such as particle impaction, thermophoresis and turbulent deposition, it has not yet been possible to quantify the retention potential by calculation alone. Consequently, PSI has initiated, and coordinated, the international research programme ARTIST [1], in which experiments are carried out to investigate aerosol retention in a steam generator during an SGTR event.

In certain severe accident scenarios, it is suspected that if aerosols are released from the core, the steam generator can be (at least partially) flooded with water as part of a defined accident management procedure. The presence of the water would greatly affect aerosol deposition, and it remains an open question to what extent this would mitigate radioactive release to the environment, and whether the operators should be instructed to flood the secondary side if such an accident should occur [2]. Due to the complex interaction of the aerosols with the steam and water in the SG, and the complexity of the geometry, no reliable models currently exist for estimating the degree of aerosol retention. Experiments are needed, and the ARTIST II experimental programme has been launched as a consequence.

The ARTIST facility

ARTIST is a scaled-down version of the FRAMATOME 33/19-type steam generator in operation at the Beznau (1136 MWth PWR) Nuclear Power Plant in Switzerland. The ARTIST facility (Fig. 1)

Figure 1: **ARTIST: a stream generator mock-up facility.**



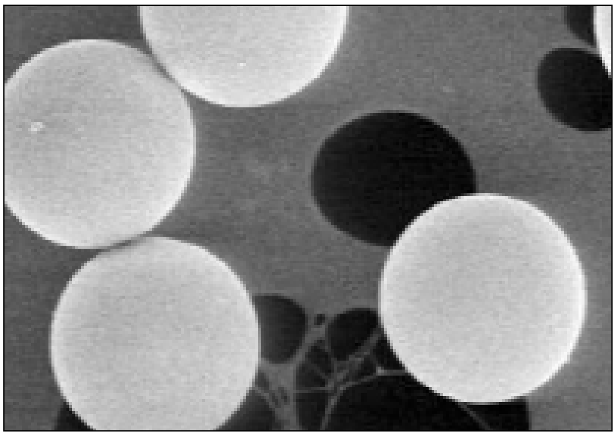


Figure 2: **Electron micrographs of test aerosols: agglomerates (top) and spherical particles (bottom).**

comprises a tube bundle, a separator unit (at 1:1 scale) and a dryer cell (at 1:1 scale). The bundle section is scaled-down in terms of height (3.8 m) and diameter (0.57 m), but the 270 straight tubes of 19.05 mm diameter are of prototypic dimensions.

Aerosol retention in the flooded bundle

Since the year 2000, several experimental campaigns have been carried out in ARTIST to determine the level of aerosol retention in a steam generator under different accident conditions. In particular, the effects of variation of different parameters on particle retention have been investigated, such as particle type – spherical or agglomerate (as depicted in Fig. 2), particle size, gas mass flowrate and water level [3].

Experiments carried out to date have confirmed that the existence of water in the secondary side significantly increases the degree of aerosol retention. Under certain conditions of the aerosols entering the flooded secondary side, less than 0.1% were ultimately transported out of the bundle. Figure 3 gives a quantitative comparison under otherwise similar conditions of the aerosol transport efficiency through a steam generator bundle under dry secondary-side conditions and for two water levels above the break.

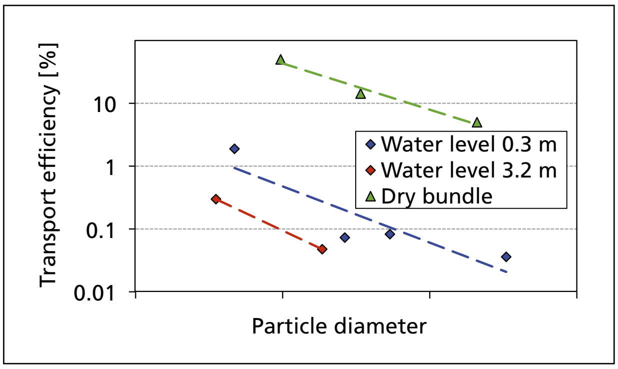


Figure 3: **Aerosol transport efficiency in a stream generator bundle under dry and flooded conditions, as measured in the ARTIST facility.**

Results

Increasing particle inertia, i.e. particle size and velocity, generally increases the retention capacity following a SGTR. Evidently, inertial mechanisms ultimately determine particle retention in the bundle close to the tube break, where most aerosol retention takes place. Even with the very low water level of 0.3 m above the break, only about 1% of the aerosols released through the break would be transported out of the bundle compared to dry-bundle conditions. Under conditions of high gas flowrate, and for large aerosol particles, aerosol transport efficiency through a flooded bundle would be just 0.04%.

At high gas flowrates, droplets are entrained from the water surface, and carry aerosol particles with them. Once the droplets evaporate, the particles are released into the gas flow. However, compared with particle retention in the water due to inertial effects close to the tube break, the effect of droplet entrainment on particle transport is shown to be minimal.

Increasing the submergence depth of the break increases the level of particle retention in the water, and the effect is much stronger in a flooded SG bundle than in open pools [3], presumably due to the jet/bundle interactions, which create very complex, two-phase flow conditions. Current indications are that models developed for open pools are not suitable for calculating aerosol retention in a flooded bundle, even for regions far from the tube break.

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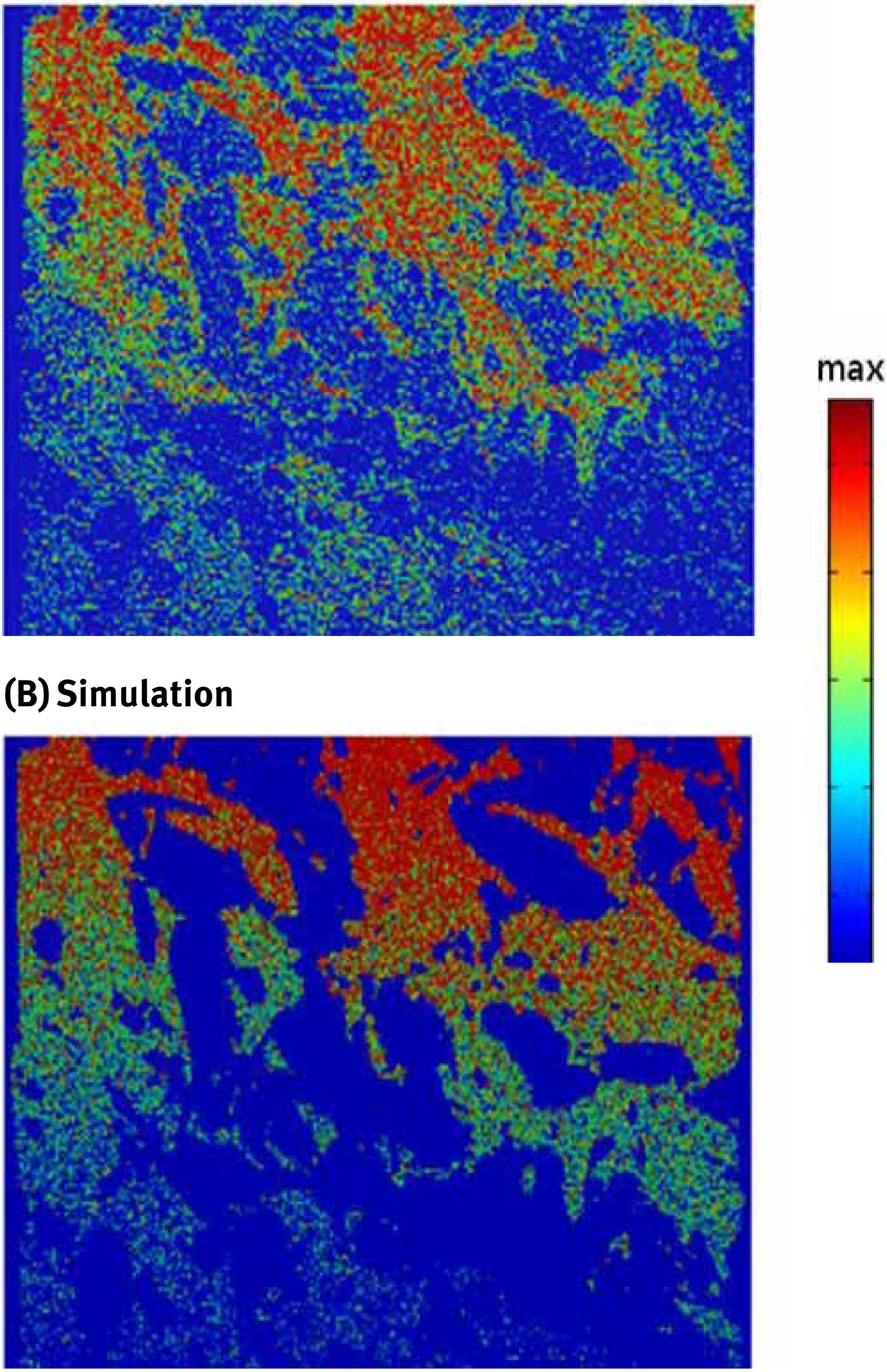
(A) Experimental



(B) Simulation

Laboratory for Waste Management (LES)

The mission of the Laboratory for Waste Management (LES) is to carry out a comprehensive research and development (R&D) programme in support of Swiss radioactive waste disposal options. The Federal Government is responsible for the disposal of radioactive waste products arising from medicine, industry and research in Switzerland, and on its behalf LES carries out an R&D programme in collaboration with the National Cooperative for the Disposal of Radioactive Waste (Nagra), the organization charged with the disposal of all Swiss radioactive waste. LES serves national needs, present and future, in respect to the safe disposal of radioactive waste in the fields of geochemistry and the transport mechanisms of radionuclides, including geochemical retardation and immobilisation.

LES makes use of some of the unique infrastructure within Switzerland available at PSI, in particular the A-Laboratory radioactive facilities, (micro-) X-Ray Adsorption and Fluorescence Spectroscopy (XAS, XFS) beam lines at the Swiss Light Source (SLS) and the Spallation Neutron Source. Research activities within LES are performed on very different spatial and temporal scales, to take advantage of all the information that can be obtained from the different facilities: i.e. from the nano-scale (molecular modelling) to the micro-scale (XAS), on to the cm scale (laboratory), and up to the field and regional scales (Mont Terri Rock Laboratory at St Ursanne).

◀ **The effect of spatial mineralogical heterogeneities on caesium (Cs) diffusion in Opalinus Clay. A comparison of the measured and simulated Cs distributions in two dimensions in the clay mineral fraction for a radial slice through the sample: (A) Cs transport monitored in-situ using micro-tomography at the PSI-TOMCAT beam line; (B) transport simulations performed using an in-house random walk code. (with the transport and sorption parameters estimated from the experimental results).**

A key point in the future strategy of LES is that the Laboratory continues, and, wherever possible and appropriate, intensifies its integration within scientific and waste management communities through cooperation and joint projects, including participation in EU-Framework programmes. Also, an important aim of LES is to contribute to “educating the next generation” in the area of waste disposal, and as a consequence of our connections to universities enables us to provide doctoral and post-doctoral positions within the Laboratory.

In 2008, the Federal Office of Energy (BFE) approved guidelines for the site selection of radioactive waste repositories sited within Switzerland, as defined in a Sectoral Plan (Sachplan Geologische Tiefenlagerung). Participation in the safety analyses carried out by Nagra during the site selection process for SMA and HAA repositories, and the preparation of licensing applications (to the year 2018), will be a central function of LES in the coming years.

Predictive sorption modelling of Ni(II), Co(II), Eu(IIII) and Th(IV) on MX-80 bentonite and Opalinus Clay

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The uptake of radionuclides from the aqueous phase onto immobile solid surfaces (mineral phases in the backfill materials and host rock) is one of the main pillars of safety in the performance assessment of deep geological repositories in Switzerland. An in-house, quasi-mechanistic sorption model, with associated non-adjustable parameters, has been used to make blind predictions of sorption isotherms of radionuclides measured on bentonite and Opalinus Clay under chemically realistic conditions. The procedures are described for Ni(II), Co(II), Eu(III) and Th(IV), and the calculated curves correspond well, or very well, with the measured data.

The disposal of radioactive waste in deep geological repositories is aimed at isolating the radionuclides from the biosphere for many hundreds of thousands of years. Radionuclide transport will most probably occur via the aqueous phase in the backfill and sealing materials, and in the host rock. In this respect, the partitioning of the dissolved species between the aqueous phase and immobile solid surfaces (sorption) is a primary consideration. It is currently common practice to treat sorption in terms of a solid/liquid distribution ratio: $R_d = C_{sorb}/C_{eq}$, in which C_{sorb} is the quantity of metal sorbed per unit mass of sorbent (mol kg^{-1}) at an equilibrium aqueous sorbate concentration of C_{eq} (mol L^{-1}). Sorption is a complex process, and it is certainly rather unsatisfactory in that the entire complexity of the radionuclide/porewater/rock interacting system for each radionuclide is contained in the single, lumped parameter: R_d. Such a purely empirical approach has a clear and significant disadvantage, in that it has no predictive capabilities. Consequently, there is a pressing need to better understand sorption, and the factors which influence it. This need has led to the development of sorption models based on mechanistic approaches, which aim to understand and quantify the processes controlling the uptake (sorption) of aqueous species.

Sorption model for clay minerals

The surfaces of clay mineral platelets carry a permanent, negative electrostatic charge arising from the isomorphous substitution of lattice cations by cations of a lower valency. Charge neutrality is maintained by electrostatically bound cations which can undergo stoichiometric exchange with the cations in solution; this process is called 'cation exchange'. A second category of reactive sites associated with clay minerals is perceived as being that of surface hydroxyl groups (≡SOH) situated along the edges of the clay platelets, *edge* or *broken-bond* sites, and which can protonate and deprotonate as a function of pH to form surface complexes with aqueous metal species.

In radioactive waste management, the 2:1 clay minerals montmorillonite (Fig. 1) and illite are important major components of bentonite and argillaceous rocks, respectively, and for this reason were chosen to be studied. A **2**-**s**ite, **p**rotolysis, **n**on-**e**lectrostatic, **s**urface **c**omplexation and **c**ation **e**xchange sorption model (2 SPNE SC/CE) has been developed to describe the uptake of radionuclides on the different site types [1].

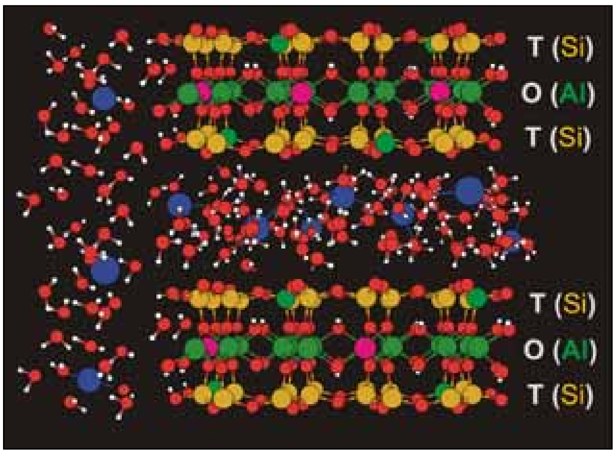

Figure 1: **Structure of the clay mineral montmorillonite.**

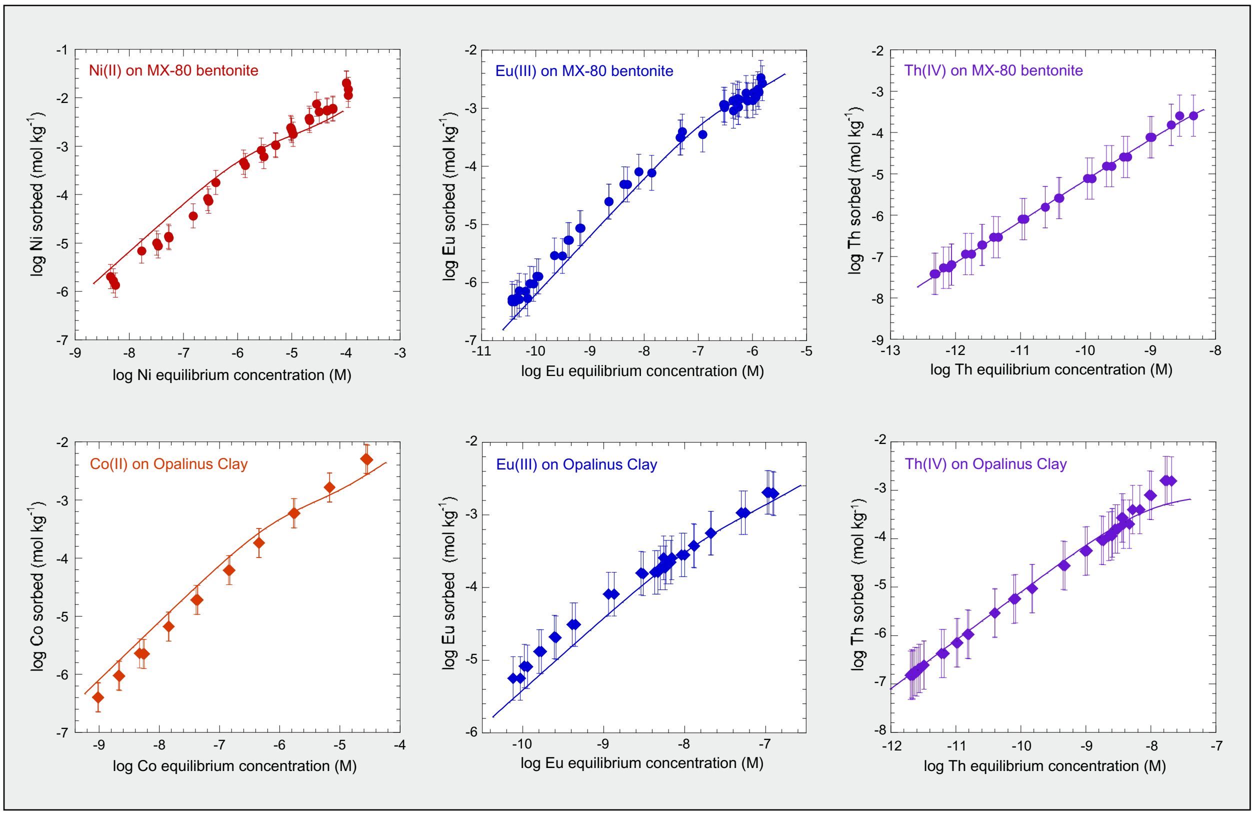

Figure 2: **Sorption isotherms on MX-80 bentonite and Opalinus Clay: experimental results and modelling.**

The bottom-up approach

The *bottom-up* approach is based on the premise that radionuclide uptake in complex mineral/groundwater systems can be quantitatively predicted from knowledge and understanding of the mechanistic sorption processes on single minerals, and the models developed to describe them. The average 2:1 clay mineral contents of bentonite and Opalinus Clay are 75 wt. % (montmorillonite) and 40 wt. % (illite and illite/smectite mixed layers), respectively. These are assumed to be the minerals predominantly responsible for sorption. The sorption values have been calculated as functions of concentration using the MINSORB code, which incorporates the 2SPNE SC/CE model and the Nagra/PSI 01/01 thermodynamic database [2], and assumes that only the free-metal and hydrolysed species are sorbing. These values are then scaled by 0.75 and 0.40, which are the respective clay mineral fractions in bentonite and Opalinus Clay.

The predicted sorption isotherms were then compared against measured data for MX-80 bentonite and Opalinus Clay. The results for Ni(II), Co(II), Eu(III) and Th(IV) are shown in Fig. 2. The metals were chosen to provide a broad range of chemical behaviour and valence states: bivalent transition metals, lanthanide/trivalent actinides, and tetravalent actinides. In most cases, measurements cover more than four orders of magnitude of the metal aqueous equilibrium concentration. The agreement between experiment (solid circles and diamonds) and the blind predictions (solid lines) is good to very good, and in most cases the calculated values fall within the given experimental error bars.

Conclusions

On the basis of the modelling results presented here, the *bottom-up* approach, used in conjunction with some simplifying assumptions and the procedures described, may be regarded as a very promising method for quantitatively calculating radionuclide uptake in complex geochemical systems. This has been demonstrated for two important barriers in the current concepts for deep geological disposal: bentonite (MX-80) and argillaceous host rocks (Opalinus Clay).

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X-ray absorption investigations of sorption sites for metal uptake by clay minerals

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Clay minerals are major components of the back-fill materials and argillaceous host rocks associated with potential high-level, radioactive waste repository sites. One of the main properties of clay is its characteristically strong sorption of radionuclides. The present study examines a major assumption made in a widely accepted sorption model used to predict the uptake of radionuclides in clay-containing systems: namely, the existence of *strong* and *weak* sorption sites. In this study, different, carefully prepared Zn/montmorillonite samples have been investigated using X-ray absorption spectroscopy.

The sorption of radioactive elements on the immobile material components of the near and far fields of a deep geological radioactive waste repository is an important process in retarding their ultimate aqueous phase transport, and as such plays a significant role in safety assessment studies. The development of robust and well-founded mechanistic sorption models to predict the uptake of radionuclides under different geochemical conditions greatly enhances the justification and defensibility of the sorption values selected for performance assessments, and thereby represents a substantial contribution to the scientific basis for long-term radioactive waste disposal.

For a high-level-waste repository, clay minerals are major components of the back-fill material (e.g. bentonite) and the host rock (e.g. Opalinus Clay), and serve to significantly retard the release of radionuclides into the biosphere by uptake processes and/or the formation of new phases. Clays possess a large specific area and a high structural charge (up to 1000 meq per kg). These characteristics are ideal for their utilisation as geochemical safety barriers in radioactive waste repositories.

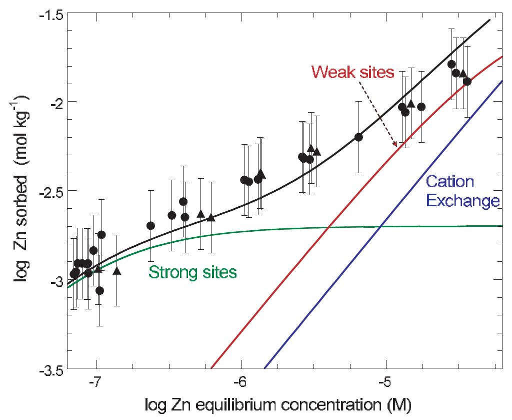

Figure 1: **Zn sorption isotherms on montmorillonite: measured data (symbols) and model predictions (lines).**

Zn sorption on montmorillonite

Montmorillonite is a dioctahedral 2:1 clay mineral consisting of an octahedral Al sheet sandwiched between two tetrahedral Si sheets. Isomorphic substitutions in the octahedral sheets generate a permanent charge on the basal plane of montmorillonite. Cations can form outer-sphere complexes at these surfaces, which are, relatively, easily interchangeable with other cations from solution: the so-called *cation exchange* process. Metals can also sorb on the pH-dependent amphoteric hydroxyl sites (≡SOH) exposed at the edges of montmorillonite by forming inner-sphere surface complexes. The two-site, protolysis, non-electrostatic surface complexation and cation exchange (2SPNE SC/CE) sorption model has been used over the past decade or so [1,2] to quantitatively describe the sorption edges and isotherms of metals with valences from II to VI in 2:1 clay minerals. One of the main assumptions of the model is that there are two broad categories of edge-sorption sites: the so-called *strong* ($\equiv S^{S}OH$) and *weak* ($\equiv S^{W1}OH$) sites. In Fig. 1, an example of the uptake of Zn on montmorillonite is shown, together with the contributions of Zn sorption on the three different types of sites, as obtained from calculation.

It was considered important to verify (or not) the strong/weak site assumption embodied in the sorption model. Because of their different sorption characteristics, the expectation was that the coordination environments of the surface complexes on the two-edge site types would be different. In this study, Zn was selected as a representative divalent transition metal cation, because it allows extended X-ray absorption fine-structure (EXAFS) measurements to be performed at environmentally relevant concentrations.

EXAFS investigations

EXAFS spectroscopy was employed to gain insight into the sorption processes at the molecular level. Initial EXAFS characterisations were carried out on the *as-received* montmorillonite samples in order to determine the speciation of the incorporated Zn species. In the corresponding radial structure functions (RSFs), obtained by Fourier transformation of the EXAFS data, there is a RSF peak (Zn-O contribution) at R+ΔR = 1.54 Å, which is typical for Zn-O back-scattering pairs (Fig. 2). Beyond the first shell, there are three further RSF peaks in the range between R+ΔR = 2-4 Å, characteristic for back-scattering pairs from the montmorillonite surface. Data analysis has shown that the incorporated local environment of Zn consists of 3 Al at 3.02 Å and 4 Si at 3.22 Å. These parameters are consistent with Zn located in the octahedral position (Zn1 in Fig. 3), in which the OH groups are transverse to the site.

The RSF of the EXAFS data for Zn sorption under conditions representative for the *strong sites* (Zn loading ≤ 2 mmol/kg) show distinct differences from the spectrum of the incorporated Zn, indicating a molecular environment differing from the incorporation on octahedral sheets. Data analysis indicates that the local Zn environment consists of ~1.5 Al at 3.02 Å, ~1 Si at 3.07 Å, and ~2 Si at 3.26 Å. These parameters indicate that a mixture of sorption complexes has formed at

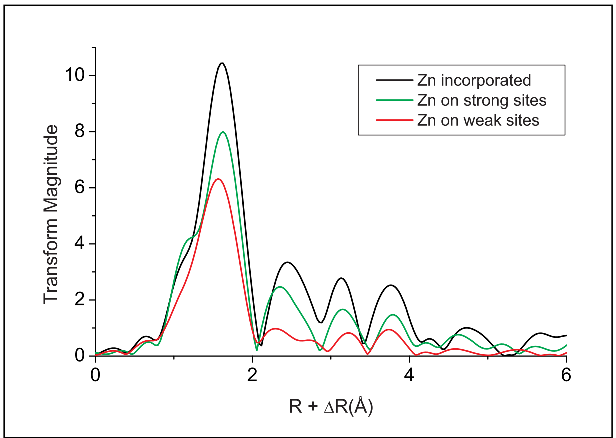

Figure 2: **Radial structure functions of Fourier-transformed EXAFS data indicating the different structural environment of Zn complexes.**

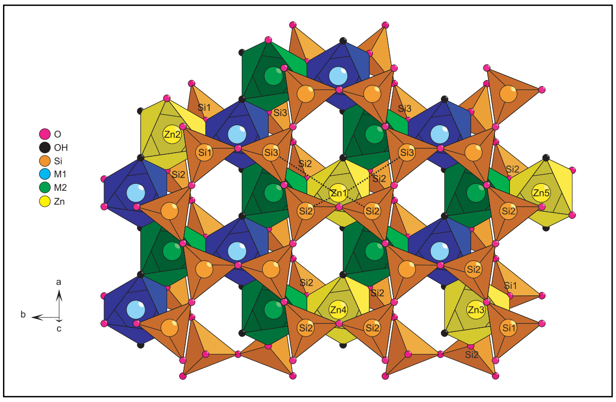

Figure 3: **Illustration of Zn incorporated in the montmorillonite clay (Zn1) and Zn complexes sorbed onto the amphoteric hydroxyl sites (Zn2–Zn5).**

the montmorillonite edge sites, as illustrated by the Zn complexes Zn2 – Zn5 in Fig. 3.

The RSF of the sample prepared with Zn loadings characteristic of *weak sites* (i.e. Zn loading ≤ 30 mmol/kg) differ significantly from those of Zn incorporated and sorbed at low concentrations. The lower amplitude in the RSF peaks indicates that the complexes are less ordered than in the previous cases. The data could be well fitted with ~1 Al at 3.04 Å and ~1 Si at 3.33 Å. Whereas the spectra of the incorporated Zn, and the Zn on the *strong sites,* exhibited a pronounced polarised EXAFS dependency, and formed surface complexes in the continuity of the Al octahedral sheets at the montmorillonite edges, the spectra of Zn *weak sites* show only minor angular dependency. The weak P-EXAFS dependency, and the longer Zn-Si bond length of 3.33 Å, indicate that the Zn complexes are not strongly bound in octahedral positions, but are rather randomly distributed on the surface.

Conclusions

The study has clearly demonstrated that there is a pronounced difference between the local environment of the Zn surface complexes at low and medium loadings, indicating that different crystallographic sites are involved in the Zn sorption behaviour on the edge sites of montmorillonite. Furthermore, the spectroscopic evidence is consistent with a multi-site sorption model, and in particular with the strong/weak site concept intrinsic to the 2SPNE SC/CE sorption model.

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Up-scaling of molecular diffusion coefficients in clays

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Up-scaling of pore diffusion coefficients for water and ions in clays from the molecular to the continuum scale is particularly challenging since the scales differ by several orders of magnitude. To address the up-scaling problem, a two-step simulation approach has been developed which enables macroscopic diffusion coefficients for water and ions for use with continuum equations to be derived from atomistic simulations, at comparably low computational costs.

Materials with low permeabilities, such as clays, are important components in the engineered and natural barriers for the planned deep geological repositories for radioactive waste in Switzerland. Slow diffusion, and the strong sorption properties of such clay-like materials greatly retard the migration of radionuclides from the repository to the environment. The macroscopic diffusive transport of dissolved species through the barrier materials originates from the Brownian motion of molecules and ions in solution, and their interaction with the surfaces of the solid medium. Up-scaling of such molecular-based phenomena to the macroscopic scale is very challenging, since the scales differ by several orders of magnitude. To address the up-scaling issue, a two-step simulation approach has been developed to derive the macroscopic diffusion coefficients for water and ions for use with the continuum equations from pore-scale molecular diffusion coefficients [1].

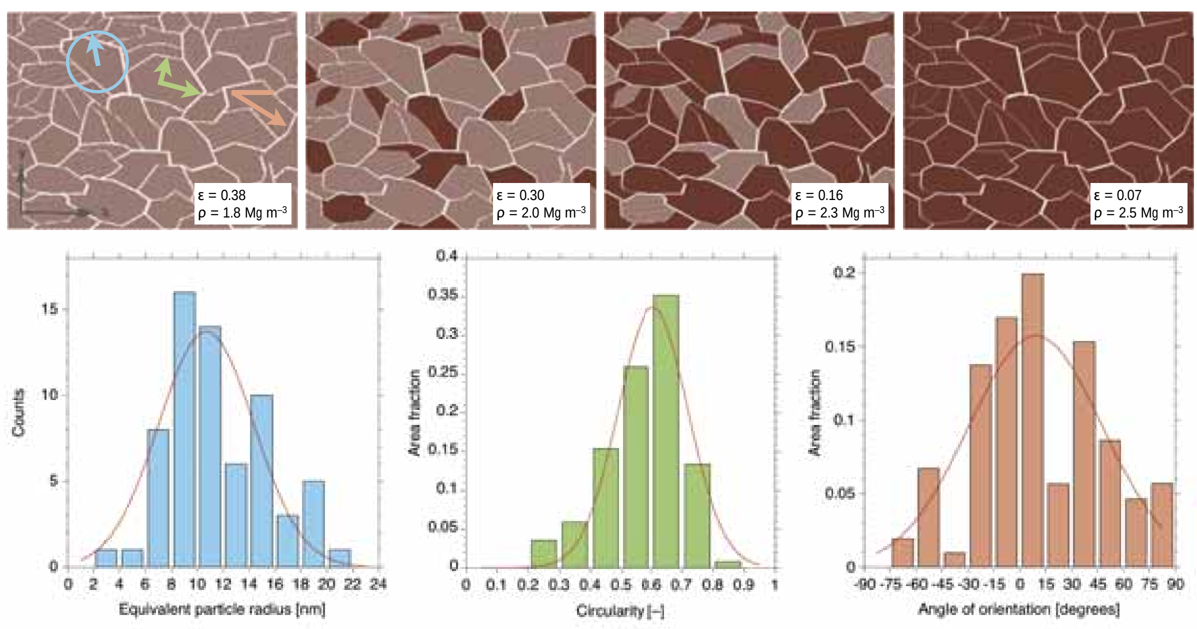

Figure 1: **Clay structural maps used for the simulations representing a heterogeneous mineralogical composition (top). From left to right: 100-0, 70-30, 30-70 and 0-100% Smectite-Illite particles. The interlayer porosity accessible to cations and water is shown by the thin white stripes. Illite particles (dark brown) are so-called non-swelling clays, i.e. those without interlayer porosity. The structural characteristics of the mineral particles in the clay structural map are shown underneath. The definition of equivalent particle radius (blue), circularity (green) and angle of orientation (red) are indicated by the lines in the top left image.**

The starting points for the up-scaling procedure are the local pore diffusion coefficients derived from molecular dynamics (MD) simulations for specific local confinements of the pore solution: i.e. the interlayer and edge regions of clay particles [2,3].

Diffusion of water and ions in such environments is strongly influenced by the composition and bulk density of the clay. Local diffusion parameters in a single pore were first obtained from molecular simulations. These were then assigned to the different types of porosity in a model clay structure (Figs. 1 and 2), and the structure-averaged diffusion coefficient was subsequently derived from random walk (RW) simulations.

The model clay structure was generated in 2D, and approximated the basic structural properties of the clay materials, such as particle-size distribution, geometry, orientation and porosity (Fig.1). The effects of mineralogical heterogeneities and of anion exclusion on the larger-scale diffusion coefficients for the model clay structure were investigated by varying the fraction of the smectite clay particles and other non-swelling phases.

The validity of the approach has been demonstrated by comparing the results of direct molecular simulations for a stack of pyrophyllite nanoparticles using a two-step up-scaling algorithm (Fig. 2). Whereas the same diffusion coefficients were obtained, the computational costs of the up-scaling approach were several orders of magnitude lower than those of the direct molecular simulation.

Future Prospects

The proposed up-scaling approach is very robust, versatile, and capable of unravelling the relationship between structural and diffusion properties of very complex porous media. In the future, the focus will be on developing an algorithm for the automatic generation of structural maps. This will enable the simulation set-up to be tailored to specific clay samples. Investigating larger samples, possibly including heterogeneities, is one option for future development, but further work on improving the high-resolution and pore-scale description will also be required.

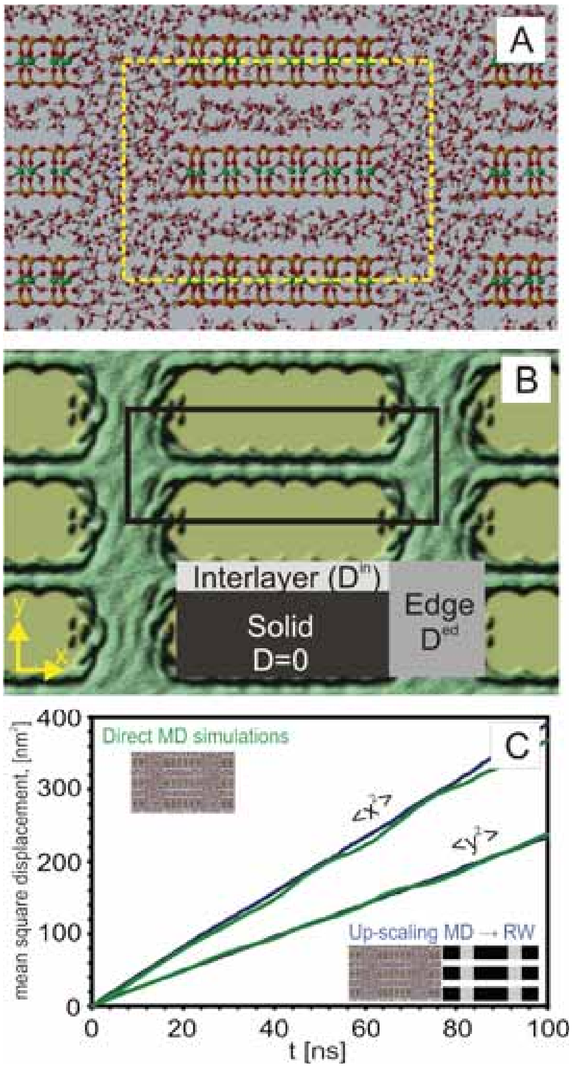

Figure 2: **(A) Snapshot from MD simulations of pyrophyllite particles separated by two water layers. Oxygen, hydrogen, silica and aluminum atoms are shown as red, grey, yellow and green spheres, respectively. The simulation supercell is shown by the yellow dashed line. (B) Surface plot of the 2D probability density profile of water molecules determined from MD simulations. Peak areas correspond to long residence times. In the overlay, light grey and grey areas indicate interlayer space and interparticle (edge) porosity, respectively, as assigned to the RW simulations. The diffusion coefficients D^{in} and D^{ed} for these pore spaces were obtained from short (200 ps) MD trajectories resident in the corresponding domains. The inaccessible solid phase is shown by the solid black line delineating the simulation cell for the RW simulations. (C) Mean square displacement of water molecules in direct MD simulations (green lines) and the results of RW simulations (blue lines). Good agreement between the slopes of the green and blue curves indicates that the structure averaged diffusion coefficients obtained by the two methods are the same.**

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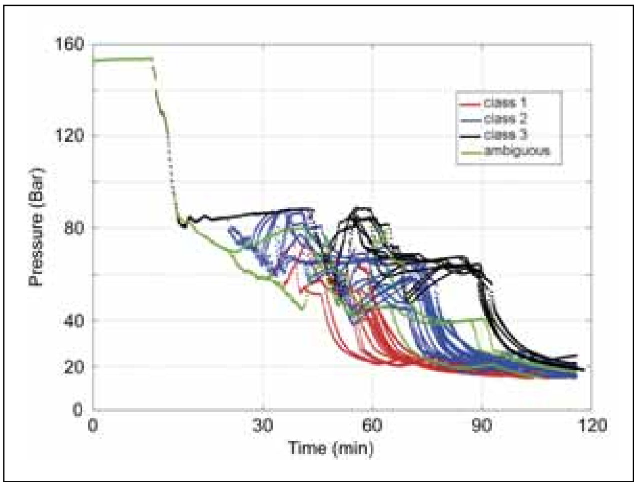
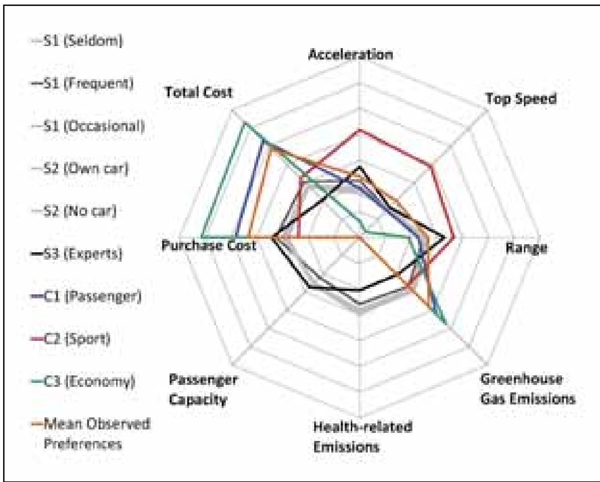
Safety assessment

Effect of crew responses on plant behavior



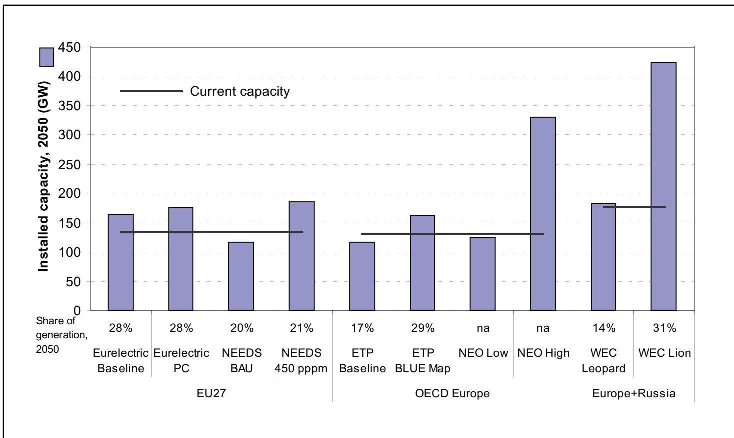
Technology and choice

Preferences in transport



System scenarios

Nuclear deployment in 2050



Laboratory for Energy Systems Analysis (LEA)

LEA is an interdisciplinary Laboratory supporting both NES and the General Energy Department (ENE). The Laboratory aims to contribute to effective decision-making on long-term technology strategies in energy supply and demand, ensuring full integration of all environmental, economic and social factors. LEA also develops methodologies, and carries out the associated risk analyses, within the framework of Human Reliability Assessment (HRA).

The activities within LEA, in cooperation with its various external partners, cover the following three project areas.

Technology Assessment (GaBE)
The project involves analyses of fossil, nuclear and renewable energy technologies. It is based on an interdisciplinary framework, thus enabling comparisons to be made between current and future options for the electricity, heating and transport sectors.

Energy Economics
Analyses are undertaken of energy systems, and associated technological changes, at the Swiss, European and global levels, all aimed at improving understanding of available options for the realisation of more sustainable energy mixes for the future.

Risk and Human Reliability
Main contributions here are to the solution of current and future issues relating to the handling of human factors in the context of Probabilistic Safety Assessment (PSA).

◀ **The safety, environmental impacts and economics of technologies are in focus in the work of the Laboratory for Energy Systems Analysis (LEA). A comprehensive perspective arises from analyses at multiple levels. These are, from top to bottom: human-technical facility-level simulations, preferences and criteria for technologies and sectors, and the economics and development of energy systems.**

Heuristic design for the technical characterization of passenger vehicles

Erik Wilhelm, Warren Schenler, *Laboratory for Energy Systems Analysis, PSI*

Light-duty vehicles account for roughly 10% of the total CO_2 emitted in the developed world, and many advanced technologies are currently under development to reduce the environmental impact of this form of personal transportation. We report here on methods and tools being developed to help policy makers, manufacturers and consumers to make the best possible decisions about how to form policy, what to build, and what to buy.

A revolutionary (albeit gradual) process of electrification and fuel diversification is underway in personal transportation, with manufacturers choosing from a variety of technology options to simultaneously satisfy consumer demand and regulatory requirements. A comprehensive overview of the impact that technologies could have on various stakeholder criteria is required to enable regulators to implement the most effective policies, for manufacturers to develop the most appropriate products, and to help consumers understand the choices they have. Special emphasis is being placed on fully electric and fuel-cell vehicles, which are among the most radical (though as yet least market-ready) technologies under development, but which are of particular interest to industrial and government stakeholders.

Methodology overview

To achieve the level of detail required to fully evaluate the technology options available, while still examining as broad a range of technologies as possible, three methods have been developed, and applied, at PSI. Heuristic vehicle design techniques draw upon historical design rules to combine and size the various technology options into self-consistent vehicle designs. Simulations based on quasi-static, linear dynamics models are being used to evaluate design feasibility. In order to objectively compare various hybrid vehicles with two or more energy sources against both each other and conventional vehicles, it is crucial to ensure that an optimal hybrid control strategy is used for the simulation. Deterministic dynamic programming methods have been developed which optimize the power split between electrical and chemical energy at each moment in the New European Driving Cycle. Once a set of virtual vehicles has been fully characterized, a multi-criteria, trade-off and decision analysis is applied to evaluate the influence of stakeholder preferences, and to input assumptions concerning the robustness and suitability of the various advanced technology options.

Survey and Multi-Criteria Analysis (MCA) results

Three separate surveys have been performed to estimate stakeholder preferences. Figure 1 shows results of the survey, together with the observed Swiss sales preferences for eight important technology indicators.

The disparity between the predicted and observed preferences seen in Fig. 1 indicate that survey respondents generally understate the financial, and overstate the environmental, criteria relative to actual car buyers. This observation has led to an investigation of how buyers' preferences could be derived from actual purchase data. The preferences calculated for the top 30 vehicles sold in Switzerland during 2010 are given in Table 1, which shows how the MCA algorithm, tested using real preferences, compared with the top-ranked

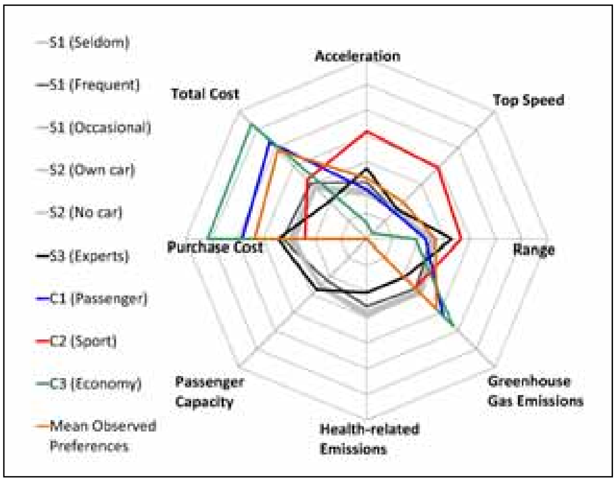


Figure 1: **Stated (S) and observed (C) preferences**

	Hybrid	Fuel (G-gasoline, D-diesel)	Displacement (L)	Electric Power (kW)	Battery charge (Ah)	Acceleration 0-100 kph (s)	Top speed (km/h)	Range (km)	CO_2 (g/km)	NO_x (g/km)	Capacity (m3)	Purchase Cost (€)	Total Cost (ct/km)
Average criteria values for observed Swiss vehicle sales data clusters													
Passenger (C1)	-	G	1.4	-	-	11.9	177	789	133	n/a	n/a	16750	35
Sport (C2)	-	G/D	1.8	-	-	9.5	202	906	167	n/a	n/a	25600	46
Economy (C3)	-	G	1.2	-	-	13.5	160	778	124	n/a	n/a	9770	29
All	-	G/D	1.4	-	-	11.2	184	820	138	n/a	n/a	19102	38
Simulation models and MCDA top vehicle designs: only ICE gasoline vehicles													
Passenger (C1)	-	G	1.9	-	-	6.7	253	642	151	0.3	2.4	15965	24
Sport (C2)	-	G	2.9	-	-	6.2	269	719	288	0.2	1.7	26882	32
Economy (C3)	-	G	1.9	-	-	10.0	217	917	110	0.2	2.4	15309	21
Simulation models and MCDA top vehicle designs: All vehicles													
Passenger (C1)	Mild	Bio-Diesel	2.7	3	4.5	6.7	255	716	138	0.2	2.5	16959	24
Sport (C2)	Parallel	G	1.5	60	60	7.6	268	1197	175	0.2	1.7	37007	38
Economy (C3)	Mild	Bio-Diesel	1.7	3	4.5	10.0	219	1008	102	0.2	2.5	16303	22

Table 1: **Real and simulated vehicle characteristics, coloured according to model predictions and real consumer preferences: red = low, yellow = mid, green = high, bordered = disagreement.**

'virtual vehicle' consumers actually purchased. Whenever the order of the 'green-yellow-red' triplets matches by column the three datasets, the MCA algorithm accurately reflects the real-world trend. As can be seen, the algorithm performs very well for most criteria, consistently matching vehicle performance ranking with observed stakeholder preference, except for the range criterion (bold border), which is difficult to determine due to the lack of fuel-tank capacity data.

The influence of varying input assumptions on vehicle design choice was investigated for Switzerland using six scenarios representing current and future technology and fuel costs (with economies of scale applied to electric and fuel cell vehicle technologies to generate reasonable manufacturing cost estimates). In Fig. 2, the observed preference clusters, C1, C2, C3, are plotted against the first- and second-ranked vehicles from the MCA according to changes in fuel cost and fuel-chain emissions. Data are taken from the US GREET database [3]. Bio-diesel and hydrogen fuel-cell vehicles dominate the ranking, due in part to their low emissions and range performances (the all-electric vehicles suffer in this regard). In general, a deviation between consumer preference and vehicle behaviour still exists. The MCA algorithm is shown to accurately predict which vehicles correspond to actual car sales, based on preferences representing observed clusters of customers. For stakeholder profiles representative of current vehicle consumers, it is seen that bio-diesel and hydrogen-fuelled, non-hybrid, parallel and fuel-cell hybrids are preferred over a broad range of Swiss technology assumptions.

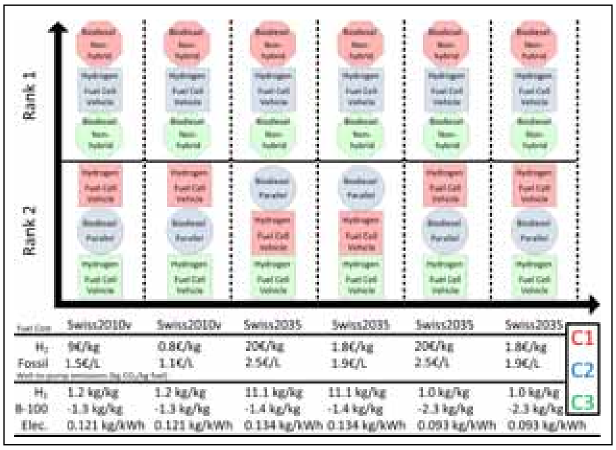


Figure 2: **Multi-criteria decision analysis on observed preferences over various current and future scenarios.**

Conclusions

It is important to emphasise that no technology satisfies all the user criteria. In addition, there remain uncertainties in future fuel and technology costs. In order to reconcile uncertainties with multiple stakeholder preferences in an interactive and meaningful way, a web-tool based on the concept of an 'heuristic vehicle design' has been created. This tool allows users to explore how their preferences and predictions of future technology characteristics influence which vehicle is top-ranked for them.

This work reports on the final stages of the CCEM HyCHANGE project, which will be expanded in the CCEM THELMA project (thelma-emobility.net), with the goal of assessing the need for electrification of personal vehicles.

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Review of scenarios for nuclear generation in Europe

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Laboratory for Energy Systems Analysis

Nuclear energy has the potential to play an important role in the future, but faces a number of uncertainties. The Laboratory for Energy System Analysis (LEA) has recently completed a timely review of leading scenario studies presenting possible futures for nuclear energy in Europe. This review compared different scenarios – from the International Energy Agency (IEA), the World Energy Council (WEC), the Nuclear Energy Agency (NEA), the European Commission (EC), and others – and sought to understand the factors driving the deployment of nuclear generation up to the year 2050, thereby identifying conditions that support nuclear energy. This work was supported by FORATOM for their 2050 Roadmap.

A number of authoritative organisations have developed scenarios of the European energy system. Such scenarios are important for policy and business decision-making, are used to explore possible developments in the future, and to test strategies against those potential developments. In the case of nuclear energy, such scenarios illustrate how the level of nuclear generation may change, and provide a consistent storyline and quantification of the driving forces and conditions affecting deployment of nuclear against alternative energy sources.

LEA has reviewed the role of nuclear generation in Europe in five leading scenario studies:

- Eurelectic's Power Choices [1];
- the IEA's Energy Technology Perspectives (ETPs) 2010 [2];
- the WEC's Energy Policy Scenarios to 2050 [3];
- the NEA's Nuclear Energy Outlook (NEO) 2008 [4]; and
- the EC's New Energy Externalities Development for Sustainability (NEEDS) project [5].

For each study, a baseline scenario was reviewed, along with a stringent climate change mitigation scenario.

Nuclear deployment in scenarios

The selected scenarios exhibit a range of outcomes for the future of nuclear energy in Europe in 2050, as shown in Fig. 1. The deployment of nuclear energy ranges from 117 GW to 424 GW across the scenarios (taking account of different regional definitions), although for most climate change mitigation scenarios a nuclear share of roughly 30% is foreseen.

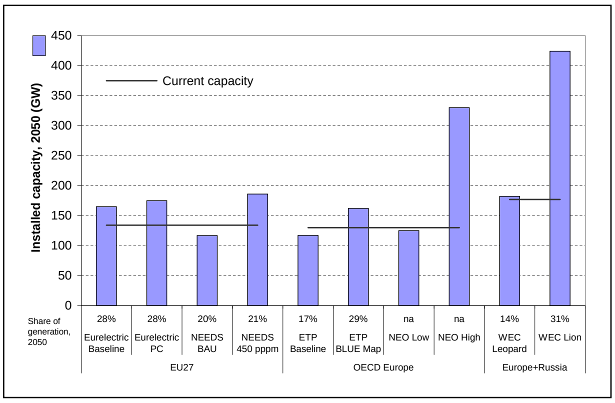

Figure 1: **Scenario estimates of European nuclear deployment for 2050.**

The review sought to determine the factors driving these scenario outcomes for the future of nuclear electricity generation in Europe. These factors could be delineated into two main groups:

1. those determining the size of the electricity market;
2. those determining the relative competitiveness, availability, and acceptability of nuclear energy vis-à-vis other electricity generation options.

Scenario driving forces

Table 1 provides a summary of the factors contributing to the observed levels of nuclear deployment for the selected scenarios. For factors affecting the size of the electricity market:

- The scenarios exhibit a range of future economic growth trajectories, and a range of estimates of future energy intensity. However, economic growth and energy intensity reductions generally correlate in the scenarios (with the exception of those from the NEEDS project). This is consistent with much of the scenario literature, and with the notion that faster economic growth equates to greater innovation, faster replacement of capital stock, and structural changes that reduce energy intensity. As a consequence, within this subset of scenarios, rates of economic development alone are not a strong determinant of the size of the electricity market.

- The scenarios also exhibit a wide range of estimates for the future level of electrification in Europe, which is important for the size of the electricity market. This divergence depends to a large extent on the assumptions built in to the scenarios regarding success of electric mobility and the large-scale electrification of industry and buildings that lead to electrification levels above 40%.

Whether nuclear is successful in the electricity market in the scenarios depends on a second set of assumptions determining the relative competitiveness, availability and acceptability of nuclear:

- Nuclear generation is assumed to be relatively cost-effective in all reviewed studies. Realising these cheap costs is likely very important for achieving the projected levels of deployment, requiring success in controlling capital expenditure costs, and realising long operating lifetimes and high load factors.
- Politically imposed limits on deployment play a large role in constraining nuclear in all scenarios (with the possible exception of WEC Lion). Moreover, sensitivity analyses presented in the scenario reports suggest that these constraints come into play before competition from other low-carbon energy sources (renewables, combined heat and power (CHP), or carbon capture and storage) has a significant impact under the technologies assumptions applied in the scenarios.
- Climate policy is also important for the deployment of nuclear. Without strong climate policy, coal-fired generation can displace nuclear when coal prices are low.

Summary

What do the findings tell us about the future of nuclear energy? They provide some insights concerning the conditions most likely to support the nuclear energy option: increasing electrification; the realisation of cheap generation costs; stringent climate policy; and, critically, a high level of political support.

Importantly, the selected scenarios only illustrate a limited number of possible future trajectories for the European energy system, and other scenarios are possible. More information is available in a forthcoming PSI Report and in [6].

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Scenarios (not all shown)	Market size				Nuclear competitivemess and availability			Overall deployment in 2050 (GW)
	GDP growth (pa)	Energy intensity (pa)	Electrification, 2050	2050 (TWh)	Limits to nuclear deployment	Technology costs	Policy	
Power Choices	1.8%	-2.5%	45%	5200	Phase out in 2 countries, restrictions in 10, limited sites	Cheap, but costs increasing for new sites	Strong climate policy (€$_{08}$103/t)	175 (28%)
NEEDS 450 ppm	1.7%	-1.4%	40%	6100	Phase out in 4 countries; restricted to current users	Cheap, Gen IV available early	Strong climate policy (€$_{00}$850/t)	186 (21%)
ETP BLUE Map	1.1%	-1.8%	30%	4300	Phase out in 4 countries, restrictions in others, global cap	Cheap	Strong climate policy (US$$_{08}$175/t)	162 (29%)
WEC Lion	2.4%	-2.2%	33%	7800	Not discussed. Strong government support, and cooperation	Cheap, low discount rate	Strong interest in CC and energy security	424 (31%)
NEO High		NA, not developed as an integrated scenario.			New capacity, public acceptance		High concern, carbon trading	330

Table 1: **Comparison of factors driving the deployment of nuclear power.**

Dynamic operator-plant simulation

Vinh N. Dang, Davide Mercurio*
*Laboratory for Energy Systems Analysis (LEA), PSI; *Now at BKW FMB Energie AG, Switzerland*

The aim of the safety analysis of nuclear power plants is to determine whether plant systems and planned personnel responses would adequately maintain safety during a spectrum of postulated emergencies, ranging from Loss-of-Coolant-Accidents (LOCAs) to earthquakes. Combining a plant model with a plant operator model in a joint simulation enables a systematic analysis to be made of how the plant response, automatic systems, and operator response, can interact during accident sequences. Such a model is able to predict a variety of performance parameters for operator action, and enables the effectiveness of different responses to be evaluated.

In the safety analyses of nuclear power plants, many accident scenarios need to be examined, given the ways in which equipment may fail to perform as required, and operator action be deemed to be inadequate. Probabilistic Safety Analysis (PSA) aims to quantify the probabilities of different failure scenarios. The results of a PSA analysis include quantification of the likelihood of undesired incidents occurring, insight into potential response weaknesses, and identification of possible safety-improvement measures.

Within a PSA, plant behaviour during an accident, and the outcomes of the various scenarios, are evaluated using plant simulation models. The evolution and outcome of accident scenarios result from a combination of the dynamic interactions of plant behaviour and the accompanying equipment and operator responses. To analyse a broad range of accident scenarios involving such interactions, a joint simulation is required, combining a plant simulation model with an operator-response model, determining what actions the operator may take, and when.

Simulating operator response

A first approach to simulating operator response is to define a series of tasks to be performed, and the time required for each task. Since the necessary operator action will depend on the particular accident scenario, and how it evolves, this type of model is limited, its main emphasis being on the time-response aspect only. The time to perform specific tasks may vary in different circumstances: for instance, extra time may be needed to assess whether or not a particular system is functioning correctly. Concerning the series of tasks, it may turn out that additional tasks are required, or that some tasks may not be needed at all. Determining the actions that operators are likely to perform in a given situation requires a different approach.

The required response of the personnel to (postulated) emergencies is planned in advance, in detail, and is subject to thorough analysis and review. These plans are the plant's emergency operating procedures, which are practised by the operators during simulator training. As a result, the operators are able to assess and respond to most scenarios without the need for fundamental analysis of specific situations.

Consequently, the operator model in this approach [1,2] dynamically generates operator action by interpreting the procedures they follow, in combination with a small set of rules for selecting response strategies or specific actions. These rules reflect training and experience, and circumvent the need for developing a detailed model of the operators' knowledge, and of their cognitive processes for situation assessment and response planning. The simulation model is complemented by a physical and thermal-hydraulic model of the plant – in many cases, the RELAP5 transient and accident analysis code [3] is employed.

Operator response in a Loss of Coolant Accident (LOCA)

The application of an operator-plant response simulation to the analysis of a LOCA scenario in a Pressurised Water Reactor (PWR) is presented here by way of illustration. In a Small-Break LOCA (SB-LOCA) – i.e. one in which a small leak is postulated in the Reactor Coolant System (RCS), the primary

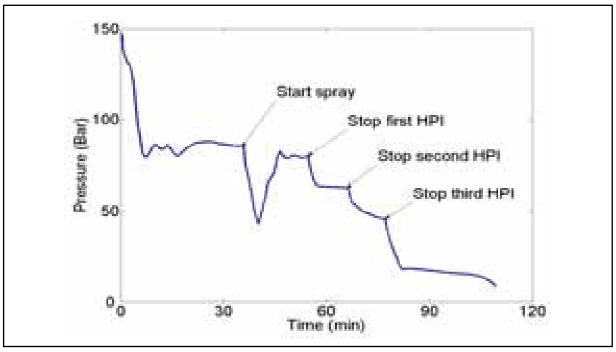

Figure 1: **Pressure response during an SB-LOCA with operator actions.**

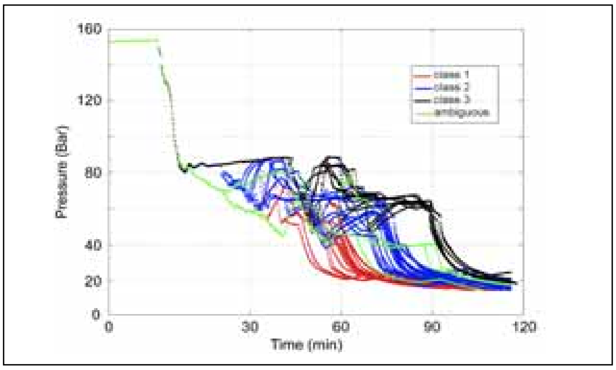

Figure 2: **Primary pressure evolutions.**

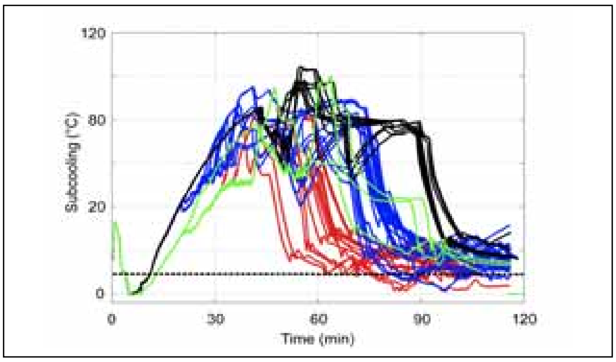

Figure 3: **Margin to saturation conditions.**

loops which remove heat from the core to generate steam in the steam generators of the secondary loop. In this scenario, the primary pressure remains relatively high at first, and the high-pressure Emergency Core-Cooling (ECC) systems actuate automatically to inject water into the primary loop, to replace the coolant lost through the leak and maintain the coolant inventory in the core. Next, the operators reduce the RCS temperature and pressure in order to achieve low-pressure conditions, and a stable, safe state for the core. In this process of cool-down and depressurisation, the operators must maintain feedwater supply to the steam generators, monitor the rate of cool-down, and control the water level in the RCS. As shown in Fig. 1, the operators eventually need to shut off the high-pressure injection (HPI) pumps: this is done progressively. The time needed to reach a low-pressure condition is one of the important parameters in the safety analysis.

The RCS pressure and subcooling margin are shown for a set of responses in Figs. 2 and 3. The operators can influence the time to reduce pressure to 20 bar, the condition for low-pressure injection (Fig. 2), through the time over which they execute the depressurisation procedure, as well as through the strategies they follow to cool down the core, ultimately control the pressure level, and the shut-off of the high-pressure injection pumps. For instance, more frequent and shorter actuation of the pressuriser spray (used to increase the coolant level in the core) can be simulated, as well as longer, more extended, spray operation. Results of such analyses indicate that the conditions under which low-pressure conditions tend to be reached the earliest (shown in red in Fig. 2) increase the risk of having saturation (i.e. boiling) conditions in the RCS. These analyses suggest that restarting the HPI pumps to avoid boiling conditions can be a cause of delay in the completion of depressurisation.

Outlook

Operator-plant simulation models have become a useful tool for Human Reliability Analysis (HRA), which deals with the assessment of the human contribution to plant safety. The models provide deeper understanding of the human performance component and its dynamics: that is, response to available indicators, the procedures and training needed to support decision-making and performance, and the feedback indicators from the plant. While nuclear power plant (NPP) training simulators offer the best possibility for observing the responses of operators to emergency conditions in a highly realistic environment, simulation analysis can offer additional insights, by considering the many possible variants to the scenarios, and the consequences of diversity in operator response.

Acknowledgements

This work is supported by the Swiss Federal Nuclear Safety Inspectorate (DIS-Vertrag Nr. 82610).

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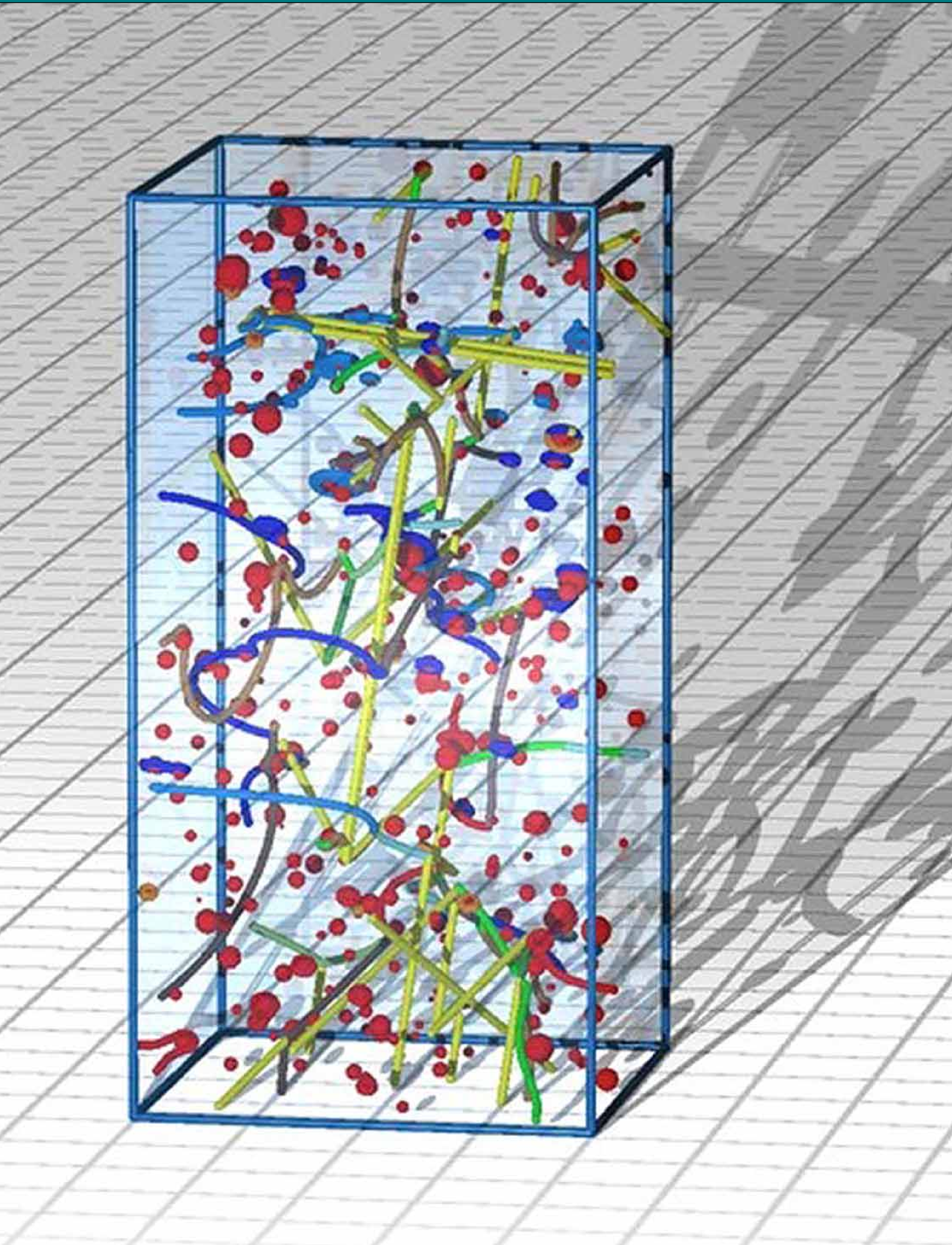

Laboratory for Nuclear Materials (LNM)

LNM addresses material-related, scientific issues in regard to safety, life-time (extension), performance and sustainability of current and future nuclear reactors.

Currently, LNM focuses on three main project areas.

High Temperature Materials

This activity involves characterisation of materials to be used in the future Generation IV reactors (particularly Gas-Cooled Reactors), which will operate at significantly higher temperatures than current Gen II reactors, and will be subject to a more intense radiation environment. Mechanistic models are being developed for the prediction of material behaviour, from the atomic level up to the scale of the continuum. Experimental validation of the models is also undertaken using advanced spectroscopic methods, and, in particular, synchrotron radiation.

Nuclear Fuels

This project focuses on micro-structural/micro-mechanical examination of the ageing of core internals (fuel rods and other structural materials), and the development of associated theoretical models. In particular, investigations of fuel damage, and identification of possible causes of failure, are carried out, and methods for the production of Gen IV fuels, and their associated fuel cycles, are also under consideration.

Component Safety (INTEGER)

This activity involves the experimental characterisation of important ageing mechanisms (stress corrosion cracking, thermal fatigue and irradiation embrittlement) in primary pressure boundary components, the development and validation of advanced, mechanistic material-ageing models, and the application of probabilistic methods for improved integrity assessments and life-time predictions. The work also encompasses the evaluation of advanced, non-destructive techniques for the early detection of fatigue, the initiation of stress corrosion cracking, and characterisation of the actual degree of embrittlement in reactor components.

◀ **Simulation of the mechanical behaviour of an oxide-dispersion, strengthened steel – a candidate structural material for Gen-IV systems – in a tensile test with a micron-sized specimen by the Discrete Dislocation Modelling (DDM) technique. The oxide particles (spheres) act as obstacles for the movement of individual dislocations (lines), thereby increasing the stress required to induce plastic deformation.**

In-situ irradiation set-up installed at the cyclotron site of CEMHTI/CNRS

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T. Sauvage, M.F. Barthe, *CEMHTI/CNRS, Orleans, France*

To support the development of high performance materials for advanced reactors, the PSI in-situ creep device, originally developed by the Research Centre Jülich (FZJ), and following a maintenance and update operation, was shipped to CNRS/CEMHTI during 2010. To expedite the installation and operation of the device, the target room at CNRS/CEMHTI had to be modified, and a new beam scanning system added. The present report briefly describes the beam parameters delivered by the cyclotrons, and the characteristics of the in-situ irradiation set-up. For illustration purposes, two application examples are also given.

The need to ensure a sustainable energy supply, with low emission of Green House Gases (GHGs), has re-stimulated the study of advanced nuclear reactor technologies. Several concepts for future nuclear fission plants have been proposed within the framework of the international Generation IV Initiative Forum (GIF) and the Sustainable Nuclear Energy Technology Platform (SNE-TP). Among the various reactor technologies being put forward for evaluation are the Very High Temperature Reactor (VHTR) and different Fast Reactor (FR) concepts.
In contrast to current Light Water Reactors (LWRs), the VHTRs and FRs would operate under much more extreme conditions: at high temperatures, at high mechanical stress levels, and with the risk of high irradiation doses to plant operators. Realisation of these advanced nuclear plants depends critically on the development of new high-performance materials. Basic understanding of material behaviour, in which parameters such as temperature, dose rate and applied specimen load can be well controlled, is at present incomplete, due to the lack of suitable high-flux, fast-neutron sources and high-energy, light-ion simulation experiments.

Experimental Set-Up

The in-situ irradiation facility was originally constructed in the late 1970s at the Institute of Solid State Research at FZJ Jülich [1] with the aim to irradiate/implant metallic and ceramic specimens using light ions (hydrogen and helium isotopes from a compact cyclotron) under uniaxial mechanical load conditions and elevated temperatures. As a result of the energetic ions first passing through a magnet scanning system and degrader wheel, the specimen can be homogeneously irradiated or implanted with continuous monitoring of the sample length and resistivity. A sketch of the facility is given in Figure 1.
The in-situ device was installed at the end of one of the cyclotron beam-lines at CEMHTI/CNRS in Orleans, France following a maintenance and update period at PSI. The target room was modified appropriately: in particular, to allow easy exchange of specimens by lifting the upper lid. In addition, the beam line was modified by adding new scanning dipoles to allow homogeous implantation/irradiation to be scanned by the beam over a large specimen area, and at high frequency.
The particles delivered by the cyclotron are listed in Table 1; these include H^+, H_2^+, D^+, $^3He^{++}$ and $^4He^{++}$, with energies up to 50 MeV. Those particles with a penetration range of 100–500 µm in most metals and alloys could be used for

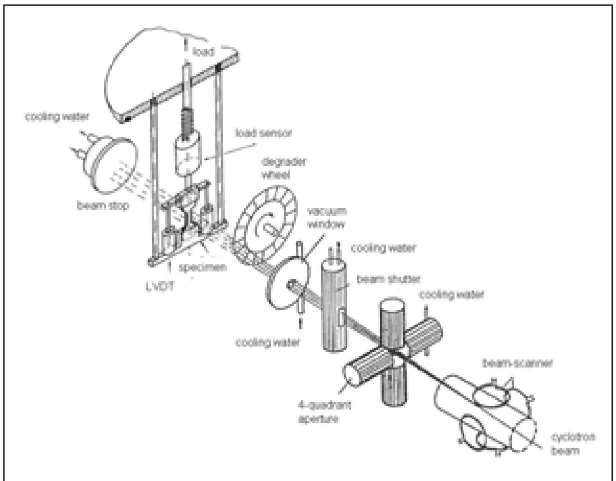

Figure 1: **Schematic view of the experiment set-up.**

Ions	Energy (MeV)	Maximum Intensity (µA)
p	5–38	40
H_2^+	5–25	40
D^+	5–25	40
α	10–50	30
He^3	10–50	15

Table 1: **The main parameters of the cyclotron beam at CEMHTI/CNRS**

homogenous implantation and/or irradiation of foil specimens. Using sub-millimetre samples, irradiation effects on the micro-structural and mechanical properties of the material could be studied with relevance to their bulk properties. The most interesting applications refer to irradiation creep, helium-embrittlement, and the stresses resulting from material swelling.

Applications

Irradiation creep observed at temperatures far below the typical thermal creep regime is an important consideration in regard to the fuel cladding in fast reactors. In most cases, the irradiation-creep strain rate $\dot{\varepsilon}$ is proportional to the irradiation displacement damage rate K, and to the stress σ (at least for low-to-medium applied stresses). That is:

$\dot{\varepsilon} = B \cdot \sigma \cdot K$

Strain rates determined from irradiation creep curves all showed a steady-state creep behaviour similar to thermal creep. The irradiation creep data from the present He- and H-implantations [2], and from previous proton and neutron irradiation of ferritic/martensitic and ODS steels, are compared in Figure 2. Strains and stresses in pressurised tubes in reactor experiments are effective values obtained from the Soederberg formalism [3]. After damage efficiency correction, it is seen that the compliances from neutron irradiations are only 2–3 times lower than those from light ions. This difference may be ascribed to two factors. First, various creep experiments performed in reactors have shown that creep rates decrease by factors of the order of 2 (or even above 5) dpa. This means that the present light-ion creep strains (at doses far below 1 dpa) may not be completely stationary. Second, all neutron data in Figure 2 are obtained for pressurised tubes: i.e. from tri-axially stressed samples, in contrast to the uniaxial stress state for the light-ion irradiations. There are some indications that irradiation creep rates in pressurised tubes may be slightly lower than those observed for uniaxial loading. Therefore, it can be concluded that light-ion irradiation gives similar irradiation creep compliance as in the case of neutrons.

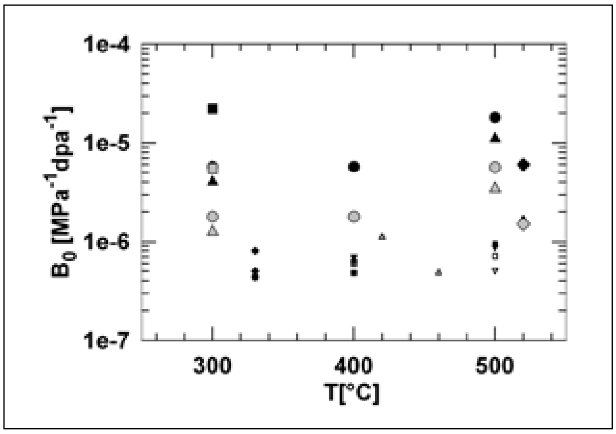

Figure 2: **Comparison of irradiation creep compliance B_0 as a function of irradiation temperature T. The large black and grey-filled symbols indicate light-ion irradiations before and after damage efficiency correction, respectively: He-implanted ODS PM2000 (●,●) and 19Cr-ODS (▲,▲), p-irradiated ODS Ni-20Cr-1ThO2 (■,■), p-irradiated martensitic DIN1.4914 (◆,◆). The small symbols indicate neutron irradiations to doses below 25 dpa (filled symbols) and above 25 dpa (empty symbols): ODS MA957 (▼,▽), HT9 (■,□), HT9 (•), F82H (△), Fe-16Cr (◆).**

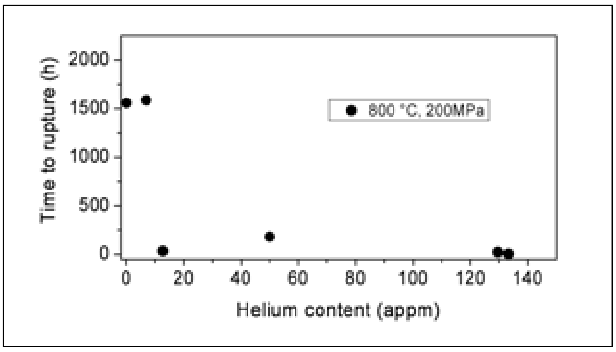

Figure 3: **Creep time-to-rupture for samples at 800°C and 200 MPa implanted at up to 1333 appm He.**

At high temperatures, helium reduces the creep lifetimes of some materials by several orders of magnitude, a phenomenon known as high-temperature helium embrittlement. Thus, the role of He effects on creep properties is a very important issue in assessing the benefits of utilisation of such materials in nuclear environments.

Figure 3 shows the time-to-rupture for TiAl alloy samples implanted over the range 0 to 1333 appm He. A clear loss in time-to-rupture is seen between the samples implanted with 6.9 and 12.7 appm He, as a result of the strong loss in ductility at fracture, indicating helium embrittlement [4].

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Plutonium characterisation in mixed-oxide nuclear fuel by synchrotron radiation

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Laboratory for Nuclear Materials, PSI

Mixed-oxide (MOX) fuels are currently being used in many reactors. As a consequence of the fission reactions taking place and the heat generated inside the fuel, chemical and structural changes occur which influence fuel performance and safety. The chemical and structural changes occurring in MOX fuel are studied at an atomistic level using synchrotron radiation from the Swiss Light Source (SLS) facility at PSI. The techniques employed will also be applied to the new types of commercial nuclear fuels to be used in Swiss nuclear power plants.

Plutonium-uranium mixed oxide (MOX) fuels are commonly used in many commercial nuclear power plants around the world. In Switzerland, MOX fuel has been used in the Beznau reactor (KKB), and, more recently, also in the Goesgen reactor (KKG). The use of MOX fuel helps to make fuel utilisation more efficient. With an increasing number of fission reactions, the chemistry in the nuclear fuel changes as a result of the production of fission products and minor actinides. Under high burn-up conditions, the actinides (U, Np, Pu, Am and Cm) in the fuel matrix undergo redox changes, which could make them more labile, as in the (hypothetical) case of water contact in a geological repository. For this reason, an atomic-level understanding of the actinide-based redox species is crucial, especially for MOX fuel.
The safety of fuel in its spent form, i.e. after its use in the reactor, is an ongoing commitment at PSI. In the present study, irradiated fuel was examined using X-ray fluorescence (XRF) and absorption fine-structure (XAFS) spectroscopy in the microXAS beam-line at SLS. The study was carried out to assess the redox stages of Pu (and thus its potential water solubility) in MOX fuels, as reported earlier [1], but also after irradiation [2]. The relevant charge distribution, local structure and speciation had already been investigated using XAFS [3], and the valence states of actinides examined by a combination of theoretical chemistry and data from XAFS experiments. These studies completed the XAFS background needed for the interpretation of the Pu atomic environment in MOX fuel.

Experiments

In the 1980s, MOX sphere-pac fuel had been produced in the Hotlab at PSI by a special process specifically developed internally at PSI. The fuel fabricated was homogeneous MOX, obtained by internal gelation and microsphere formation [2]. The MOX sphere-pac fuel segments prepared were then transferred to the Swiss Pressurised Water Reactor (PWR) Beznau-1 for irradiation. The fuel was then utilized through six reactor cycles, reaching a burn-up of 60 MWd/kg of fissile metal, corresponding to 6% Fission per Initial (heavy) Metal Atom (FIMA). Compared with conventional fuel in Swiss power plants, this level of burn-up is considered to be relatively high.
The more-recent synchrotron-based XRF and XAFS investigations have been performed at the microXAS beam line of SLS. It is worth noting that the use of radioactive material at the beamline is subject to stringent activity limits, with the consequence that only very small test specimens can be examined. Sample preparation was somewhat of a challenge, with specimens made by means of a replicate technique.
This involved applying a piece of adhesive Kapton tape to obtain a kind of 'fingerprint' of the fuel surface. It was possible to collect fuel particles from different positions over the fuel cross-section: from the centre of the fuel to its periphery, or rim, which is subject to higher burn-up. The X-ray data from the samples were collected at room temperature by monitoring the fluorescence emission using an Si solid-state fluorescence detector. The subsequent XAFS procedure focused on the L_{III}-edge absorption line, with energy calibration carried out for U and Pu using a Zr film, measured in transmission.

Results and outlook

Samples were first analysed in fluorescence mode using XRF (Figure 1) prior to XAFS examination of specific spots.

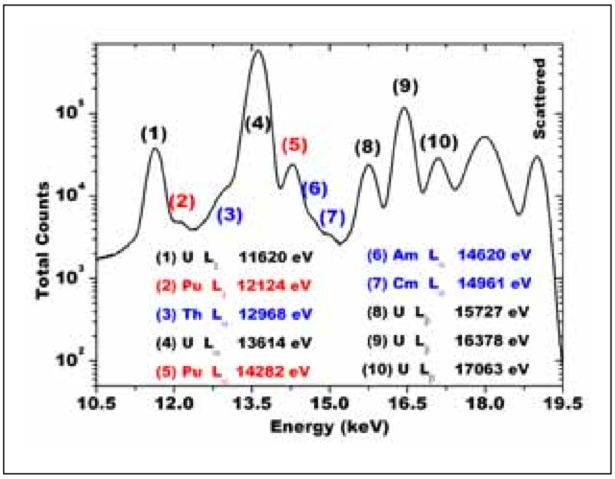

Figure 1: **Fluorescence spectrum of the irradiated MOX sample. Positions (3), (6) and (7) are indicated for Th, Am and Cm, respectively, but with signals too low to be detected.**

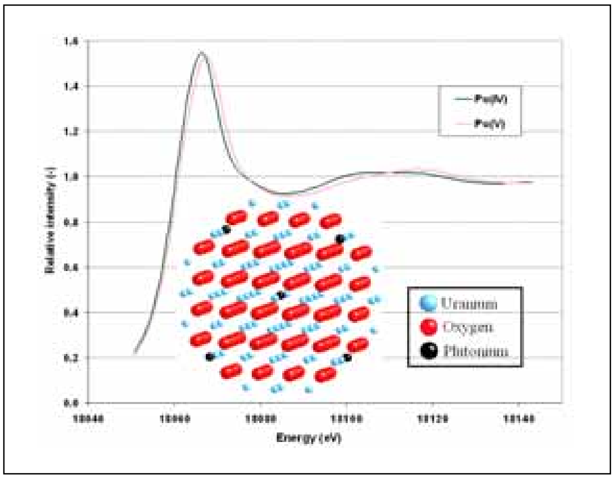

Figure 3: **Plutonium L$_{III}$ XAFS spectra calculated for the MOX solid solution.**

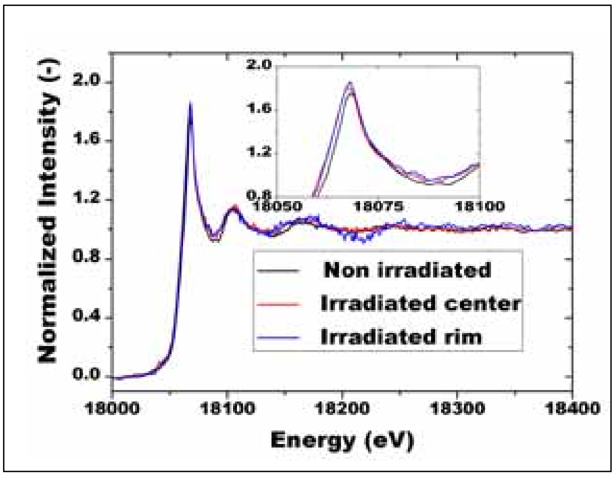

Figure 2: **Plutonium L$_{III}$ XAFS spectra for the non-irradiated and irradiated MOX fuel in the centre and rim zones.**

The environment of U and Pu can be influenced by the presence of fission products. However, for the given burn-up, the concentration of individual fission products does not reach 1 at.%, and should thus not *a priori* significantly affect the U or Pu XAFS data. For the plutonium L$_{III}$-edge, the E_0 was stable at 18062 ± 2 eV for all relevant cases; i.e. for the non-irradiated and irradiated samples at the rim and centre zones (Figure 2).

Comparison of the white-line factor for Pu in the non-irradiated and irradiated MOX samples reveals that the Pu remaining in the irradiated sample consisted of more than 95% Pu(IV), but no (i.e. less than 5%) Pu(V) or Pu(VI) could be detected, though the fuel did undergo slight oxidation in the rim zone.

This slight oxidation may have been buffered by the UO_2 matrix. The spectrum may be calculated using FEFF 8.4 for a cluster involving 247 atoms (82 U, 5 Pu, 160 O), as displayed in Figure 3. The smaller Pu(V)-O distance makes the Pu white line somewhat less distinct, and shifted towards higher energy (though still below the energy resolution level).

The results for Pu confirm that, with the given detection limit, irradiation-induced detrimental speciation changes – i.e. those affecting Pu water solubility – are not expected.

The methods and results described have evoked some industrial interest. As a consequence, a joint project with Swissnuclear, and other partners, has been launched for the fabrication of fuel with specific additives. The additives analysed should lead to better – especially gaseous – retention of fission products in the fuel matrix.

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Numerical simulation of stresses and micro-cracks in anisotropic polycrystals using XFEM

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Crack formation due to cyclic thermal shocks and/or thermal fatigue is of primary concern in the safety of nuclear power plants. It has been observed that under such loadings micro-cracks are initiated, either at the surface or inside the bulk material, depending on its micro-structure. The present study aims to understand the effect of material anisotropy at the microscopic scale on the resulting stress distribution. Use is also made of the eXtended Finite Element Method (XFEM) to calculate micro-crack formation.

Within the PLiM project [1], investigations are being conducted into the initiation and growth of cracks which may occur in the steel piping networks of nuclear power plants under cyclic thermal loading. The cracks occur on the inner surface of the pipes, and often result from turbulent mixing of hot and cold coolant streams.

Experiments indicate that fracture mechanics models based on continuum mechanics need to be refined in order to take into account the micro-structure of polycrystals [2,3]. Thus, increased attention is being paid to the stresses induced at the microscopic scale, to the anisotropic nature of the structural material, and to the influence of grain orientation and size.

In this work, crack growth is estimated using the eXtended Finite Element Method, or XFEM [4], recently implemented in the Abaqus code [5]. Using this approach, crack growth may be calculated without the need to modify the mesh as the crack grows. Analyses of two loading conditions of the same micro-structure of stainless steel AISI 321 are presented as illustration.

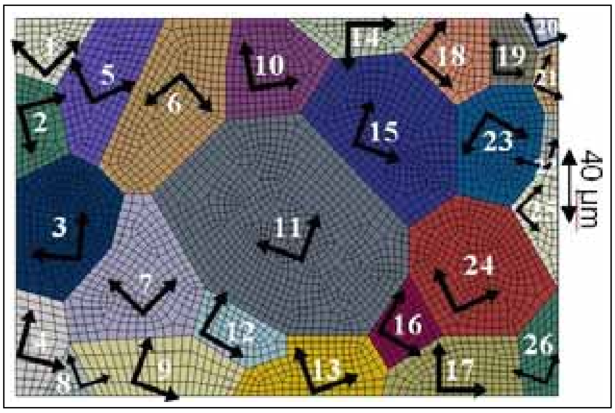

Figure 1: **2D polycrystal model: arrows indicate the orientations of the elastic properties of the grains.**

Problem description

A 2D representation of the structure is considered [6]. The grain shapes and sizes adopted for this study are shown in Fig. 1. The orthotropic thermal expansion coefficients and elastic constants are transformed to the given grain orientation, and assigned to each grain according to: $C_{11}=C_{22}=C_{33}=223$, $C_{12}=C_{13}=C_{23}=132$ and $C_{44}=C_{55}=C_{66}=116$; all units being expressed in GPa.

Stress distribution due to thermal loading

The model is assumed to have free boundaries, and to be heated uniformly from 20°C to 200°C. Figure 2 shows the von Mises stress distribution over the deformed structure. Increasing the temperature results in the stress concentrations appearing mainly at the grain boundaries – a consequence of the anisotropic material properties of the grains. The free boundary assumption implies that internal stresses would not exist if the grains had the same orientation, or the material was isotropic.

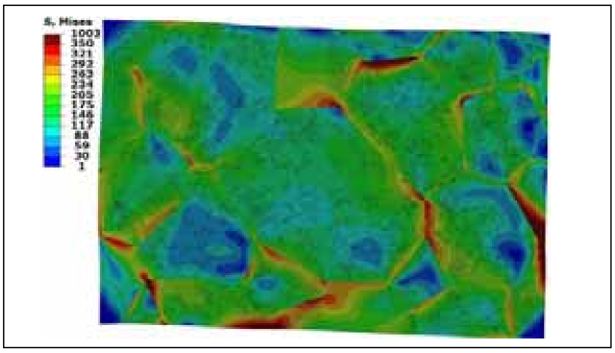

Figure 2: **Von Mises stress distribution (in MPa) as a consequence of pure thermal loading.**

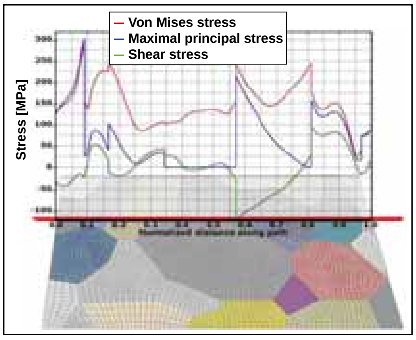

Figure 3: **Von Mises, maximum principal and shear stress variations due to thermal loading along a horizontal line crossing several grain boundaries.**

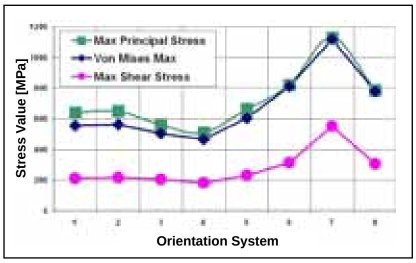

Figure 4: **Maximum von Mises, principal and shear stresses for different orientations and grain sizes.**

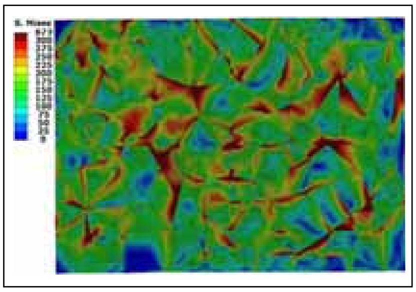

Figure 5: **Von Mises stresses [MPa] for small grains.**

Figure 3 shows the von Mises, maximum principal and shear stress variations along a horizontal line crossing several grains. The discontinuous change in the stress field at the grain boundaries is readily seen, the stress profiles within the grains remaining smooth.

Effects due to grain orientation and size

In order to study the effect of grain orientation on the stress field, eight other orientations were assigned to the same granular structure. Figure 4 shows the change in the maximum von Mises, principal and shear stresses for each of the orientations considered. Note that the maximum stress can increase by a factor two as a result of changes in grain orientation. The location of the maximum stress also changes. To analyse the influence of grain size on the predicted stress levels, the 26 grains considered earlier were further partitioned, resulting in 138 smaller grains, each with a different orientation. The corresponding stress field is displayed in Fig. 5. The maximum local von Mises stress is 673 MPa, while higher values (up to 1003 MPa) had been obtained for the larger grain sizes (Fig. 2).
Considering the maximum principal stress criterion for crack initiation and propagation, an XFEM calculation was performed to study the cracking mechanism within the granular structure. Here, a uniaxial vertical force was applied (red arrows in Fig. 6) in addition to the thermal loading. In this case, two transgranular cracks can be seen to initiate at the grain boundaries (denoted by GB in Fig. 6), but then grow inside the grains and not along the grain boundaries themselves. However, both cracks nucleate at the two critical stress concentration locations previously identified.

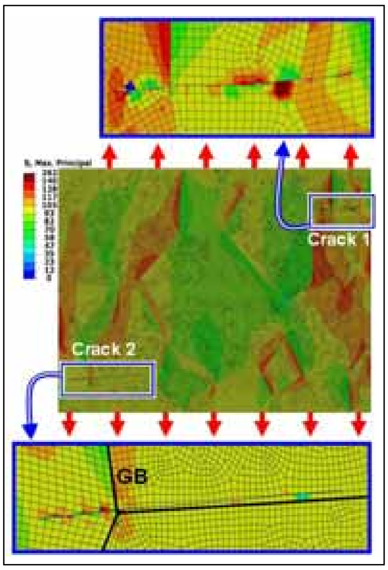

Figure 6: **Maximum principal stresses [MPa] and crack growth for combined thermal and (uniaxial) mechanical loading, as predicted using XFEM.**

Conclusions

By considering the anisotropic micro-structure of the test material, it has been shown that pure thermal loading can lead to internal stresses, even for the case when no external constraints exist. The influence of grain orientation and size on local stress levels is also a consideration, with smaller grain representations leading to a reduction in the predicted maximum stress levels. Within the XFEM methodology, crack initiation and growth are calculated using the maximum principal stress criterion determined experimentally. Future models will be based on measured grain sizes and orientation, together with their distribution. Ultimately, this should enable crack initiation probabilities to be estimated.

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ATOMIKA

Hot Laboratory Division (AHL)

The Hot Laboratory (Hot Lab) is the largest nuclear research facility under the supervision of the Swiss Federal Nuclear Safety Inspectorate (ENSI), and the only Swiss research facility capable of examining large quantities of radioactive materials. The Hot Lab incorporates a complex infrastructure to ensure that all radioactive materials inside the building are contained, and that a safe workplace for its staff is guaranteed. AHL is the operator of the Hot Lab, as well as being its main user. The two main tasks of the division are to ensure safe and efficient utilisation of its infrastructure, and to conduct state-of-the-art service work for the Swiss nuclear industry.

Highlights of current activities are listed below:

- AHL offers Hot Lab users modern analytical tools for the manipulation and investigation of radioactive materials. In particular, the laboratory is very well equipped to carry out structural and chemical analyses of materials used in Nuclear Power Plants (NPPs) and PSI's accelerator facilities.
- The Hot Lab is one of the nominated 'PSI User-Lab Facilities', and is responsible for the preparation and handling of radioactive specimens prior to their deployment in the large facilities at PSI: namely, SINQ, SLS and PROTEUS.
- AHL has strong links to the Swiss NPPs, and undertakes the necessary detailed material investigations for ensuring their continuing safe and economic operation. AHL also collaborates with several research projects concerned with the fuel and structural materials used in nuclear installations. Through this involvement, AHL has established recognition of its competence within the nuclear material research community worldwide.
- AHL continues to benefit directly from its very competent and experienced staff, and successfully develops new analysis methods, supplys the necessary infrastructure for tackling the challenging and ever-changing needs of the nuclear community, and is also able to undertake its own safety evaluation to ensure its continuing safe operation.

◀ **Details of the SIMS (Secondary Ion Mass Spectrometry) installation representing the manifold state-of-the-art instrumentation infrastructure of the Hot Lab at PSI.**

Determination of the instant release fraction of spent LWR fuel at elevated burn-up

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Lawrence Johnson, *NAGRA, Wettingen, Switzerland*

Safety studies of the long-term disposal of radioactive waste in deep geological structures require information regarding gap and grain-boundary inventories of the radionuclides in the spent fuel. These inventories determine the "instant-release fraction", i.e. the part of the inventory available for fast release once the spent fuel comes into contact with groundwater. In the framework of the *Gap Inventory* project between NAGRA and PSI, measurements of the rapid release of ^{14}C, ^{137}Cs, ^{129}I and ^{79}Se have been performed for a number of UO_2 and MOX fuel samples with burn-up levels exceeding 60 GWd/t U.

Assessment of the safety of the disposal procedures for spent nuclear fuel in underground repositories requires definitive information regarding the mechanisms and rates of release of the various radionuclides present in the fuel. Previous studies [1-3] have shown that release of radionuclides from spent fuel pellets is controlled by two processes: the slow dissolution of the UO_2 grains; and the rapid release of some elements, such as Cs, I and Cl, which occurs in the first weeks to months of contact with an aqueous solution. This instant-release fraction (IRF) is of particular interest in geological disposal of spent fuel, because some of the preferentially released radionuclides (e.g. ^{129}I, ^{36}Cl) are both long-lived and geochemically mobile. During recent years, there has been a worldwide trend towards increasing fuel burn-up, which means that the present database (of fuel burn-ups of 20-50 GWd/t) on the IRF of fission products is no longer appropriate for the safety analysis of future repositories. As a consequence, the *Gap Inventory* project has been initiated between NAGRA and PSI, which aims to determine IRF values for high burn-up fuels (> 60 GWd/t U), and to investigate correlations between the IRF of some fission products and the in-reactor release (FGR).

Experiments

Several well-characterized PWR and BWR UO_2 or MOX fuel samples from the Swiss Nuclear Power Plants (NPPs), each of length 20 mm, have been leached in glass columns with 28 mM borate buffer (of pH value 8.5) for up to 100 days at room temperature, each containing 20 ppm NaI as iodine carrier. The experimental set-up within the shielded dissolution box is shown in Fig. 1. In order to investigate if the fuel surface available for attack by the leaching agent has a significant impact on the leach rate itself, a number of samples were divided into two batches: in some, cuts were made in the cladding, the fuel pellet was broken, whereas the other samples were kept intact (Fig. 1). In order to investigate the leaching behaviour of the rim zone of the pellet in more detail, the inner part of one fuel pellet was removed mechanically by drilling, leaving only an outer fuel ring about 1 mm thick.

The FGR of the fuel rods selected for the Gap Inventory project had been determined earlier in the framework of other post-irradiation examination (PIE) projects, and were provided as valuable comparison data by the Swiss NPPs.

The leaching kinetics of several nuclides with long radioactive half-lives were measured in the leach solutions using different techniques: liquid scintillation (^{14}C); inductively coupled plasma mass spectrometry, ICP-MS (^{79}Se, ^{135}Cs); and gamma

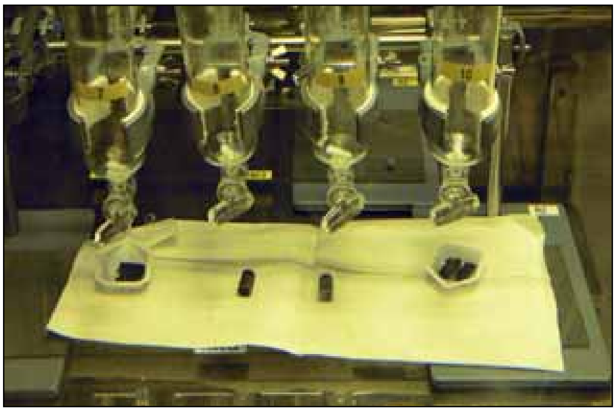

Figure 1: **Experimental set-up for leach experiments (picture taken before the start of the experiments).**

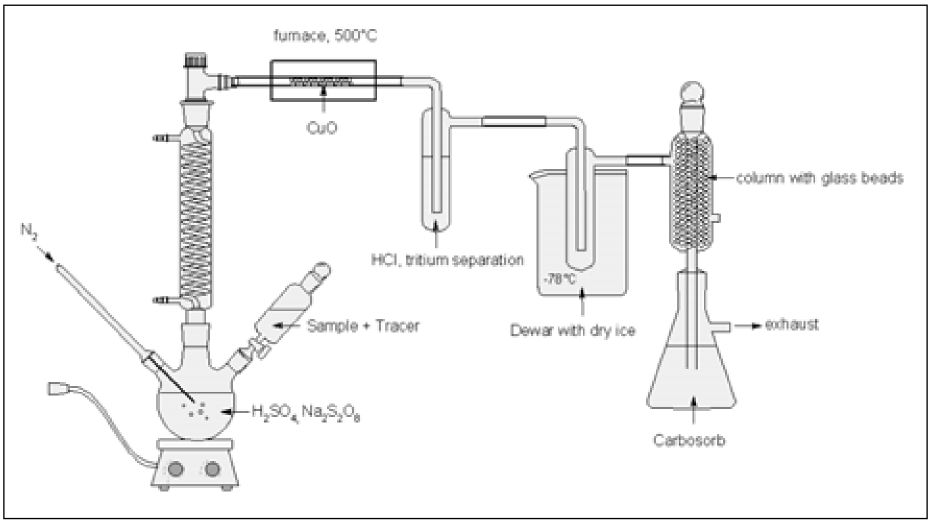

Figure 2: **Experimental set-up for ^{14}C extraction.**

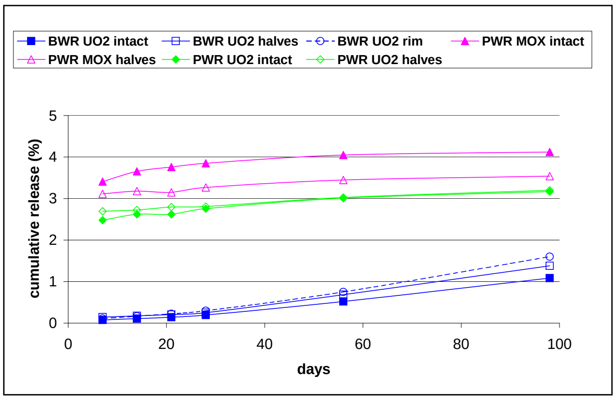

Figure 3: **Cumulative release of ^{137}Cs as percentages of the initial inventory.**

spectrometry (^{129}I, ^{137}Cs). The uranium content in the leach solutions was also measured using ICP-MS, to determine the dissolution rates of the spent fuel samples. The experimental set-up used for the extraction of ^{14}C from the leach solutions is shown in Fig. 2.

Results

The cumulative release of ^{137}Cs as a percentage of the initial inventory [4] for the various fuel samples is shown in Fig. 3. Releases from the MOX samples are highest, followed by those from the PWR UO_2 fuel samples. In both cases, the integrated release levels off at the end of the 100-day test period. Releases from the BWR UO_2 samples are the lowest, but show no indication of levelling off. The rim sample has slightly higher fractional release than that of both the intact segment and the samples from the same fuel rod cut into two.

Typically, in the case of ^{129}I, the leach solutions of all BWR samples had concentrations close to the detection limits, whereas for the PWR samples the release rates were in the range of 4-10% of the initial inventory. Based on the measured uranium concentrations in the leach solutions, it was calculated that the contribution of matrix dissolution to the release of Cs and I was less than about 0.01%, and thus negligible. The ^{79}Se concentrations in all leach solutions were below the detection limit of the ICP-MS equipment (around 0.5 ng/g of solution). Therefore, the leach fraction for ^{79}Se can be estimated to be less than 0.22% for the fuel samples investigated in this study.

The ^{14}C concentrations for all the BWR samples investigated were very close to 'blank levels' (i.e. about 80 Bq), and could therefore not be quantified. For the PWR samples, results indicate a slightly higher release of ^{14}C (per g of fuel) for the MOX samples compared to UO_2 fuel. Differences between the ^{14}C extracted from the fuel matrix and that from the cladding cannot be distinguished with the present experimental set-up, and remains a subject for investigation in detail in future studies.

Discussion

The data from the various high burn-up fuels investigated are broadly consistent with the study of Johnson *et al.* [2] in regard to comparisons between FGR and IRF. In most cases, ^{129}I releases are of the same order as, or somewhat less than, those for FGR, while ^{137}Cs releases are significantly lower [5].

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On-line monitoring of the thermal release of fission products from nuclear fuel using ICP-MS

Natalia Shcherbina, Niko Kivel, Ines Günther-Leopold, *Hot Laboratory Division, PSI*

An inductive vaporization (InVap) device has been designed in the Hot Laboratory to fulfil requirements to define advanced reprocessing concepts for irradiated nuclear fuel at high temperatures. With this set-up, it is possible to heat irradiated fuel samples up to 2300°C, thereby inducing the release of fission products (FPs). Using the InVap device in combination with an Inductively Coupled Plasma Mass Spectrometry (ICP-MS) system enables the kinetics of FP release at specific temperatures and redox conditions to be monitored online. Theoretical modelling in support of the experimental work is also being undertaken.

Reprocessing of irradiated fuel has been the subject of intense investigation over the past two decades. Activities are primarily motivated by the growing interest in fast breeder reactor (FBR) technology, and the new fuels materials required for Gen IV reactors, both based on the assumption of a closed nuclear fuel cycle. One of the practical problems associated with reprocessing is that U, Pu and minor actinides (MAs) must be recovered completely, totally separated from the fission products (FPs). Their accumulation during reactor operation is also a critical issue from the point of view of fuel economy and recycling. FPs generally have a negative impact, since they degrade the mechanical and thermo-physical properties of irradiated fuel, and act as a poison due to their relatively large neutron cross-sections.

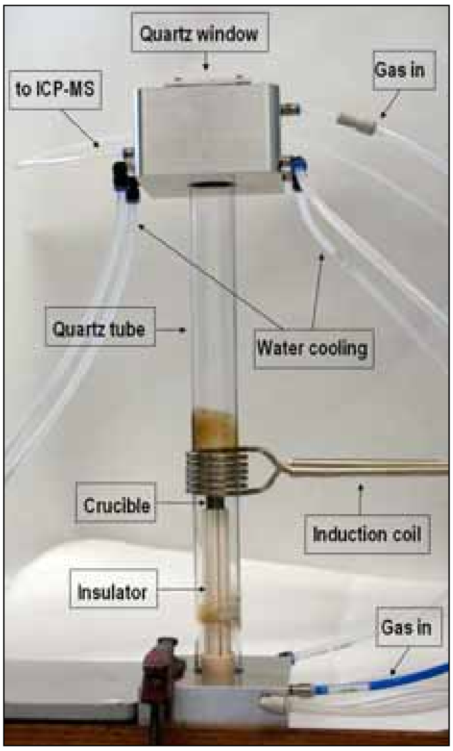

Figure 1:
InVap device

Experimental Procedures

In collaboration with a number of other laboratories in Europe, the Hot Lab at PSI performs studies of so-called 'head-end' steps in reprocessing technology. Two main strategies for reprocessing are pursued internationally, with U, Pu, MAs and FPs being separated either by aqueous or pyro-chemical techniques. In the aqueous reprocessing concept, the fuel is first dissolved in nitric acid and then, through a number of liquid-liquid extraction steps, separation of U and Pu (PUREX) [1], MAs and FPs (TRUEX, TALSPEAK, GANEX) is achieved [2,3]. In contrast, with the pyro-chemical reprocessing technique, the irradiated fuel is first dissolved in a molten salt solution, and the separation is then performed electro-chemically [4]. An alternative way of achieving FP separation is by voloxidation [5].

The activities of the Hot Lab in the general framework of the HERACLES project (head-end reprocessing studies by thermal and thermo-chemical treatment of fuels) are focused on advanced reprocessing concepts aimed at removing FPs by thermal treatment prior to fuel dissolution. FP removal can be beneficial to both the aqueous and pyro-chemical approaches in that it simplifies the reprocessing procedure.

The main idea is based on the chemical and physical properties of the FPs, these being either volatile (Cs, Rb) or semi-volatile (Sr, Sb). In order to separate them from U and Pu, a thermal treatment is applied using the InVap device designed at PSI [6], consisting of a quartz tube closed at both ends by water-cooled, aluminium lids (Fig. 1). The sample is placed in a carbon crucible placed on top of a ceramic insulator, and

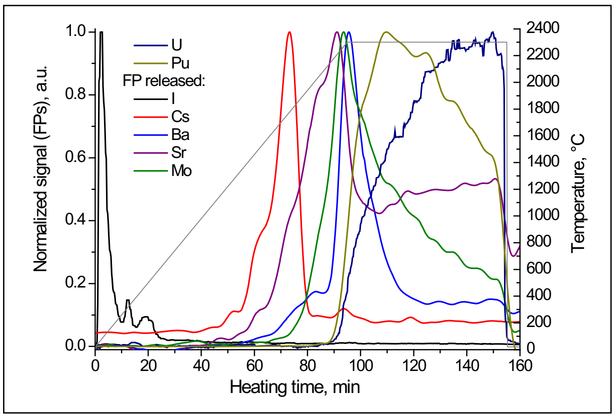

Figure 2: **Release of FPs from irradiated uranium oxide fuel (burn-up 3.8%) measured using ICP-MS.**

surrounded by an induction coil connected to a high-frequency generator to induce heating up to 2300°C. The carrier gas (Ar, Ar+H_2, Ar+CO_2) is introduced at the base of the InVap device (labelled in Fig. 1), and transports the FPs released at a specific temperature to the detection system (ICP-MS). The different compositions are used to examine the fuel samples under precise redox conditions; the heating regimes being controlled either manually or automatically. Thus, coupling of the InVap device to an ICP-MS system enables online monitoring of the kinetics of the FP release to be achieved over a wide mass range under controlled, dynamic conditions. The InVap device has been in operation since 2009, and successfully tested on a number of inactive samples (i.e. on materials simulating irradiated fuel) as well as on active (irradiated) fuel samples. As an example, Fig. 2 shows the release profiles of FPs from irradiated uranium oxide fuel with a burn-up of 3.8% fissions per initial metal atom (FIMA). Though FPs are known to accumulate in different amounts during reactor operation, here, for illustration, the transient signals have been normalized to the maximum of the corresponding FPs released during the entire experiment.

The experiments have increased understanding of the FP release mechanisms. As seen in Fig. 2, the release of different FPs begins at different times, and thus at different temperatures. Thermodynamically, the release of FPs into the gas phase is determined by their speciation in the solid phase. Consequently, one may expect release of I earlier than either Sr or Cs.

However, the formation of 'open porosity' in the fuel during reactor operation introduces another factor, which can be assumed to be a limiting step in the release process. In order to be released from the fuel matrix, the FPs first need to diffuse from the grain bulk to the grain boundaries. From there, they are released into the open inner porosity region created by the fission gases (Xe and Kr), and thence transported outwards through the fuel channels. Diffusion and interfacial reactions on the grain boundaries then define the kinetics of the FP release, i.e. the release depends on the FP inventory (irradiation history), their speciation, and the structure of the irradiated fuel. Figure 2 shows that the release of FPs such as I, Cs, Ba and Sr begins earlier than the thermal release of U and Pu. Consequently, the separation of a number of FPs from the fuel matrix (U and Pu oxides) in the InVap device can be achieved by applying appropriate temperature and redox conditions.

Future Improvements

So far, only qualitative analyses of the FP release using the InVap device have been possible. The major limitation for quantitative analysis of the FP release is the lack of an internal standard with a known concentration in the irradiated fuel samples. To establish a semi-quantitative procedure, further investigations are needed: e.g. the application of isotope dilution.

Theoretical Studies

Theoretical modelling is also being carried out. The codes utilized are the Gibbs Energy Minimization Selektor (GEMS) program, developed at PSI [7], and the Module of Fission Product Release (MFPR) program, developed at the Russian Academy of Science in collaboration with IRSN, Cadarache. Good agreement with the available literature data has been obtained. Further, the kinetics of the release simulated using MFPR agree with most of the profiles obtained using InVap. Overall, good understanding of FP behaviour at elevated temperatures in fuels has been achieved.

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SEM analysis of platinum deposits in a feed-water line of a BWR using noble metal injection

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In a Boiling Water Reactor (BWR), the water chemistry can be changed from normal to a reducing environment by the injection of hydrogen. To lower the hydrogen injection rate, while the reactor is online and at full power, platinum is added as a catalyst. A high flow resistance was noted at the two entry points into a feed-water line of a Swiss BWR. Subsequent inspection revealed the presence of solid deposits, or plugs, in the lines. Samples taken from the plugs were collected for detailed inspection in the Hot Laboratory at PSI using Scanning Electron Microscopy (SEM). The analyses revealed showed that the deposits were mainly composed of platinum, but with low concentrations of trace elements. The deposits are built up of very small particles (80 – 220 nm), sometimes in well-organised structures.

The water chemistry in a Boiling Water Reactor (BWR) is changed from normal to a reducing environment by the injection of hydrogen. Platinum is added as a catalyst to reduce the hydrogen injection rate. However, it was noticed at a Swiss BWR that in order to keep the injection rate constant, the pressure in the injection lines had to be steadily increased. As a consequence, the two entry points into the feed-water line were inspected using an endoscope, and solid deposits were found. The blockages (or plugs) were broken from the inside, and pieces collected for detailed analysis using Scanning Electron Microscopy (SEM) in the Hot Laboratory at PSI.

Sample preparation and analysis

The material was prepared on an aluminium sample holder, and fixed with a double-sided, glue-coated, conducting mounting tape. To prevent charging occurring during the subsequent analysis process, the material was further coated with a conducting layer of carbon using an evaporation process.
In order to obtain high-resolution pictures, two larger samples were also broken to expose virgin surfaces: i.e. ones not affected by the surface treatment. These surfaces were then coated with a 15 nm layer of platinum by means of a sputtering technique.
Sample analyses were performed using a SEM instrument manufactured by Zeiss (DSM 962) and equipped with the *System SIX* (from Noran Instruments) for the Energy Dispersive X-ray Spectrometry (EDS) measurements.

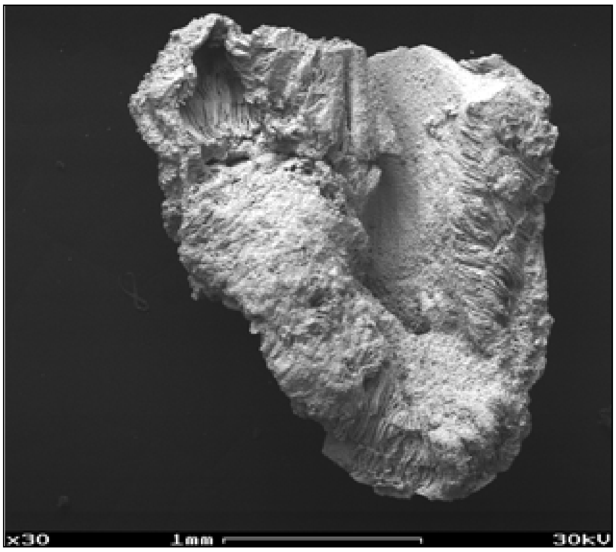

Figure 1: **Virgin surface of a freshly broken sample at 30x magnification.**

Scanning Electron Microscopy (SEM)

Figure 1 shows the freshly broken virgin surface of one of the samples taken from the plug. Even at low magnification, organised structures built from conglomerations of small particles can be found (Fig. 2). At high magnification, crystal-like structures can be observed (Fig. 3).
The sizes of the elementary particles were determined from high-resolution SEM pictures (Fig. 4) using digital imaging techniques: the sizes of the particles range from 80 nm up to 220 nm. Larger particles are normally formed from subsequent agglomerations of such small particles.

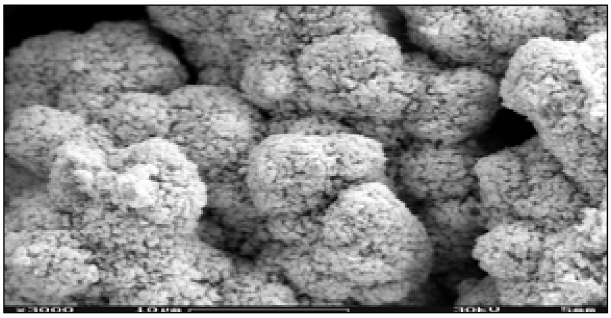

Figure 2: **Conglomerations of small primary particles at 3000x magnification.**

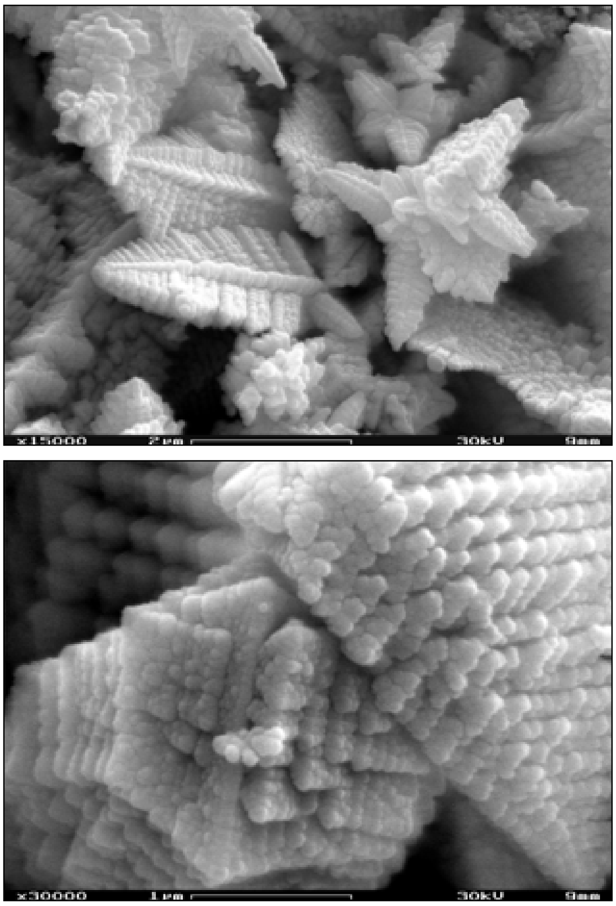

Figure 3: **Examples of well-organised structures of sub-microscopic particle conglomerations at high magnification: x15000 (above), x30000 (below).**

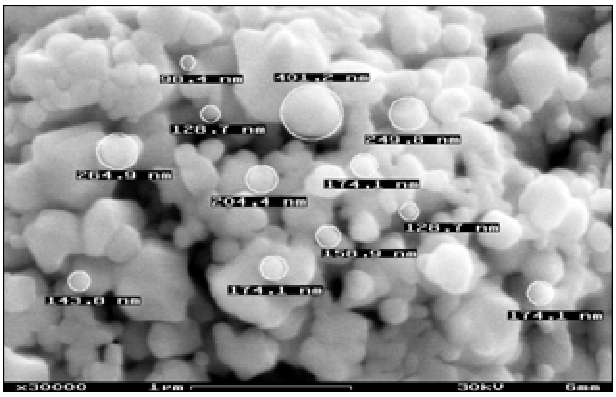

Figure 4: **Size determination of elementary particles.**

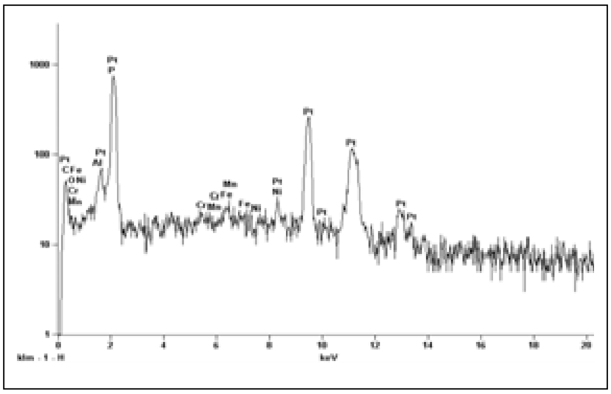

Figure 5: **Typical EDS spectrum.**

Energy Dispersive X-Ray Spectroscopy

In order to identify the main constituents of the plug material, EDS analyses were performed. Area- and point-wise analyses produced the same result: the deposits consist mainly of platinum particles with low concentrations of some trace elements, such as oxygen, aluminium, phosphorus and those elements originating from the steel of the pipework, namely iron, chromium, nickel and manganese.

Conclusions

SEM analysis of the structure and composition of the plugs taken from the injection entry points of the feed-lines of an operating BWR has revealed that they consist mainly of very small particle conglomerations of platinum, sometimes in highly organised structures, and with low concentrations of trace elements resulting from the injection process itself, though not from agglomerations of foreign materials.

Basic material analyses conducted as a result of unexpected operational conditions observed at Swiss power plants, as illustrated here, and realised at the Hot Laboratory at PSI, do not always lead to direct determination of the root cause of the problem. However, the present study has identified the different possibilities that can exist. Such information is invaluable in gaining further insights into more complex situations, and in better understanding of the phenomena. Ultimately, such in-depth analyses result in improved safety and the economy of the plant.

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◀ **Organisational chart of the Nuclear Energy and Safety Research Department.**

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"Corrosion product deposits on boiling water reactor claddings: experimental and theoretical investigations on the magnetic properties", Eur. Materials Research Society (E-MRS), Spring Meeting, Strasbourg, France, 7-11 June 2010

[1] KKL, Leibstadt, CH
[2] Westinghouse Atom AG, Västeras, SE

Perret G., Jordan K.A., Köberl O.
"PROTEUS zero-power reactor – future experimental programmes", Test Research and Training Reactors (TRTR), Int. Group on Research Reactors (IGORR), Knoxville, USA, 19-23 Sept. 2010

Pfingsten W., Baeyens B., Bradbury M.H.
"Modelling Ni diffusion in bentonite using different sorption models", 4th Int. Meeting on Clays in Natural & Engineered Barriers for Radioactive Waste Confinement, Nantes, France, 29 March – 1 April 2010

Pouchon M.A.
"Conversion processes: internal gelation and the sphere-pac concept", 1st Int. Workshop on Actinide Recycling by Separation and Transmutation (ACSEPT), 31 March – 2 April 2010, Lisbon, Portugal, 2010

Pouchon M.A.
"Advanced Internal Gelation for the Spherepac Concept", IAEA Technical Meeting on Manufacturing Methods for Advanced Nuclear Fuels, IAEA, Vienna, Austria, 18-21 May 2010

Pouchon M.A., Cozzo C., Vaucher S.[1], Ishizaki K.[1], Graule T.[2], Holzer L.[2]
"PINE – Platform for Innovative Nuclear Fuel", Energietagung, Villigen PSI, Switzerland, 11 June 2010

[1] EMPA, Thun, CH
[2] EMPA, Dübendorf, CH

Qian G., Hong Y.[1], Zhou C.[1]
"Investigation of very-high-cycle fatigue behavior of a structural steel with smooth and notched specimens", 18th Eur. Conf. on Fracture, Dresden, Germany, 30 Aug. – 3 Sept. 2010

[1] Chinese Academy of Sciences, CN

Seifert H.-P.
"Environmentally-Assisted Cracking of Light Water Reactor Structural Materials", CCMX Technology Apéritif "Mechanical Properties of Materials", Comptence

Centre for Materials Science and Technology (CCMX), EMPA, Dübendorf, Switzerland, 20 Oct. 2010

Seifert H.-P., Ritter S., Leber H.
"Corrosion Fatigue Behavior of Austenitic Stainless Steels under Simulated BWR/HWC & PWR Conditions", Joint FRI LNM Seminar on EAC, FRI University, Tohoku, Japan, 8 April 2010

Shcherbina N., Svedkauskaite-LeGore J., Kivel N., Günther-Leopold I., Kulik D.
"Thermodynamic modeling of fission product release during thermal treatment of spent oxide fuel", Eur. Materials Research Society (E-MRS) 2010 Spring Meeting, Strasbourg, France, 7-11 June 2010

Soler J.[1], Leupin O.[2], Gimmi T., Hirschorn S.[3]
"Planning of a new tracer and reactive diffusive transport experiment in the Opalinus Clay at the Mont Terri Underground Rock Laboratory", Eur. Geosciences Union (EGU) General Assembly, Vienna, Austria, 2-7 May 2010

[1] CSIC-IJA, Barcelona, ES
[2] NAGRA, Wettingen, CH
[3] NWMO, Toronto, CA

Tits J., Gaona X., Dähn R., Wieland E.
"Actinide sorption by cementitious materials: the case of neptunium", Plutonium Futures – The Science 2010, Keystone, CO, USA, 19-23 Sept. 2010

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"Experimental sorption studies on cement systems", Int. Workshop on Actinide and Brine Chemistry, Carlsbad, NM, USA, 15-17 Sept. 2010

[1] KIT, Karlsruhe, DE

Uldry A.C., Borca C., Idhil A., Victoria M., Hoffelner W., Samaras M.
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Valance S., Bertsch J., Alam A.
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[1] Hydrochemical Consultant, Amsterdam, NL
[2] NAGRA, Wettingen, CH

Van Loon L.R., Müller W.
"Diffusion of 60Co(II), 154Eu(III) and 134Cs(I) in Opalinus Clay: results from in-diffusion measurements combined with a high-resolution abrasive method", 4th Int. Meeting on Clays in Natural & Engineered Barriers for Radioactive Waste Confinement, Nantes, France, 29 March – 1 April 2010

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"Comparision of Neutron Data Libraries for the Analysis of the H.B. Robinson-2 Pressure Vessel Benchmark with the MCNPX Code", Int. Conf. on Nuclear Data for Science and Technology (ND2010), Jeju Island, Korea, 26-30 April 2010

[1] ETHZ/EPFL, CH

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"Computer simulations of $CaCO_3$-$MgCO_3$-$CdCO_3$ system and calculations of solid solution – aqueous solution equilibria", Jahrestagung der Deutschen Mineralogischen Gesellschaft, Muenster, Germany, 19-22 Sept. 2010

[1] University of Frankfurt, DE
[2] Curtin University, Peth, AU

Wang H., Grolimund D., Günther D.[1], Van Loon L.R., Borca C., Aeschlimann B.[1], Barmettler K.[1]
"Quantitative imaging of elemental diffusion into heterogeneous media using LA-ICP-MS and synchrotron microXRF", 10th Eur. Workshop on Laser-Ablation, Kiel, Germany, 29 June – 1 July 2010

[1] ETHZ, Zurich, CH

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Environmental Radiochemical Analysis, Chester, UK, 15-17 Sept. 2010

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"Post-Irradiation Analysis of a Pb/Bi Filled Ta Target Irradiated at Isolde", OECD/NEA Workshop on Technology and Components of ADS, Karlsruhe, Germany, 15-17 March 2010

1 University of Berne, CH
2 ESS, Lund, SE
3 CERN, Geneva, CH

Zimmermann J., Froideval A., Pouchon M.A., Van Petegem S., Chen J.C., Schmitt B., Leber H., Van Swygenhoven H., Hoffelner W.
"In-situ X-ray diffraction synchrotron study of an advanced ODS ferritic steel during tensile deformation", Materials Science and Engineering Conf. (MSE-2010), TU Darmstadt, Germany, 24-26 Aug. 2010

Conference Posters (without Proceedings)

Andreani M., Bertsch J., Churakov S., Curti E., Dähn R., Degueldre C., Froideval A., Hoffelner W., Kuri G., Lind T., Martin M., Niffenegger M., Paladino D., Patterson B., Pouchon M.A., Prasser H.-M., Valance S.
"Towards new opportunities in nuclear materials science and technology using Free Electron Laser", Joint Eur. XFEL and HASYLAB Users' Meeting, Hamburg, Germany, 27-29 Jan. 2010

D. Devynck, M. Krack
"Core-Shell vs. Rigid-Ion Model for the Study of Point Defects in UO_2", Nuclear Materials Conf. 2010 (NuMat2010), ZKM, Karlsruhe, Germany, 4-7 Oct. 2010

NES and ENE Colloquia

Berner U.
"Ein Solid Solution Modell für Montmorillonit", 21 Sept. 2010

Dai Y., Chen J.C.
"The behaviour of structural materials after irradiation in spallation targets and irradiation creep", 18 March 2010

Degueldre C., Lind T.
"Colloids go slow for generation and particles go fast for retention", 18 Nov. 2010

Diaz N.
"Modelling the Cs diffusion front in Opalinus Clay considering the 3D spatial mineral heterogeneities", 27 April 2010

Gaona X.
"Chemistry of neptunium in hyperalkaline systems", 26 Aug. 2010

Günther-Leopold I.
"Charakterisierung bestrahlter Brennstoffproben durch zerstörende und nicht-zerstörende Analysenverfahren", 22 April 2010

Kivel N.
"^{60}Fe half-life: ICP-MS measurements", 22 Oct. 2010

Prasser H.-M.
"Kernenergie: Sicherheit, Umwelt, Brennstoff", 17 Dec. 2010

Wohlmuther M., Smith B.L.
"MEGAPIE — a mega project", 27 Jan. 2010

Zimmermann M.A.
"Can we measure possible erosion of safety margins in nuclear power plants?", 28 Oct. 2010

University Level Teaching

Anklin H.
"Grundlagen der Kernphysik", "Kernspaltung und Kernbrennstoffe", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul A: Reaktorphysik, Paul Scherrer Institute, Villigen PSI, Switzerland, 11 Jan. 2010

Badillo A.
"Phase Field: I", "Phase Field: II", "Phase Field: III", Lectures given at the University of Sevilla, Spain, Feb. 22-26, 2010

Baeyens B.
"Modelling of sorption processes and sorption databases", International Training Course (ITC) on Transport and Retention of Radionuclides in Argillaceous and Fractured Media, Paul Scherrer Institute, Villigen PSI, Switzerland, 30 Nov. – 7 Dec. 2010

Cavedon J.-M.
"Einführung zum Kurs", Swissnuclear-PSI Fortbildungskurs Kerntechnik 2010, Paul Scherrer Institute, Villigen PSI, Switzerland, 11 Jan. 2010

Cavedon J.-M.
Course: From nuclear structure to nuclear energy (402-0627-00L), ETHZ, Zurich, Switzerland, Autumn Semester, 2010

Chawla R.
“Diffusionstheorie, Kritikalität, Leistungsverteilung”, “Reaktorkinetik Reaktivitätsänderungen”, “Einführung in das Reaktorpraktikum”, Swissnuclear-PSI Fortbildungskurs Kerntechnik 2010, Modul A: Reaktorphysik, Paul Scherrer Institute, Villigen PSI, Switzerland, 12 Jan. 2010

Churakov S.
“Molecular simulations of Zn^{2+} sorption on edges of montmorillonite”, Université Pierre et Marie Curie (UPMC), Paris, France, 29 Oct. 2010

Churakov S.
“Up-scaling molecular diffusion coefficients in clay-rich materials”, University of Zurich, Switzerland, 9 Dec. 2010

Degueldre C.
“Comportement des radionuclides dans l’environnement”, Lecture Course, Centre des sciences naturelles de l’environnement, University of Geneva, Switzerland, Autumn Semester, 2010

Degueldre C.
“Nuclear materials characterisation by advanced techniques”, Seminar, RWTH, Aachen University, Germany, 17 Dec. 2010

Gimmi T.
“Fluids in the Crust”, Masters Course in Environmental and Resource Geochemistry, University of Berne, Switzerland, Autumn Semester, 2010

Gimmi T.
“Upscaling from lab to field; Part I: Field experiments and upscaling from lab to field; Part II: natural tracer profiles”, International Training Course (ITC) on Transport and Retention of Radionuclides in Argillaceous and Fractured Media, Paul Scherrer Institute, Villigen, Switzerland, 30 Nov. - 7 Dec. 2010

Giust F.
“Neutron transport theory and Light Water Reactor (LWR) lattice calculations”; “LWR core modeling”; “Reactor shielding”, Lectures given in the Master Course: Special Topics in Reactor Physics, ETHZ, Zurich, Switzerland, Spring Semester, 2010

Glaus M.
“Diffusion of radionuclides in clays: experimental aspects”, International Training Course (ITC) on Transport and Retention of Radionuclides in Argillaceous and Fractured Media”, Paul Scherrer Institute, Villigen PSI, Switzerland, 30 Nov. – 7 Dec. 2010

Günther-Leopold I.
Lectures given in the Course: Nuclear Energy Systems, ETHZ, Zurich, Switzerland, Spring Semester, 2010

Günther-Leopold I.
“Kernbrennstoffe“, Strategic Exercise given in the Course: Analytische Chemie V, ETHZ, Zurich, Switzerland, 26 Oct. 2010

Hummel W.
Lectures given in the Course: Nuclear Energy Systems, ETHZ, Zurich, Switzerland, Spring Semester, 2010

Hummel W.
“Landfilling, nuclear repositories and contaminated sites”, Lectures given for the degree of Master in Biogeochemistry and Pollutant Dynamics, and for Master of Environmental Engineering, ETHZ, Zurich, Switzerland, Spring Semester, 2010

Kolbe E.
“Radioisotopes and Radiation Applications”, Lectures given in the Nuclear Energy Masters Program, EPFL, Lausanne, Switzerland, Autumn Semester, 2010

Kolbe E.
“Radioisotopes and Radiation Applications”, Lectures given in the Nuclear Energy Masters Program, ETHZ, Zurich, Switzerland, Autumn Semester, 2010

Kolbe E.
“Theoretische Kernphysik”, Lecture Course, University of Basel, Switzerland, Autumn Semester 2010

Kosakowski G.
“Statistics in Earth Sciences”, Lecture Course: Master of Applied Environmental Geoscience, University of Tubingen, Germany, Autumn Semester, 2010

Kosakowski G.
“Introduction to the reactive transport modeling with OGS-GEMS”, Special Lecture for Bachelor and Master Students, Gibbs Energy Minimalization Sector (GEMS) Teaching, University of Gunma, Japan, 21-22 Oct. 2010

Krack M.
“Basics of Electronic Structure Calculations”, Lecture given at the 1st F-BRIDGE School on Ceramic Nuclear Fuel and Cladding Materials, KIT, Karlsruhe, Germany, 28 Sept. – 2 Oct. 2010

Kulik D.
“From atomistic calculations to thermodynamic modelling”, DMG-DGK-Helmholtz-Virtual-Institute

Workshop, Gibbs Energy Minimalization Sector (GEMS) teaching, University of Frankfurt, Germany, 1-6 March, 2010

Kulik D.
"EMU-School 2010, Ion-partitioning in ambient temperature aqueous systems", Lecture given in Gibbs Energy Minimalization Sector (GEMS) teaching, EMU School, Oviedo, Spain, 27-30 June, 2010

Kulik D.
"Geochemical Speciation & Bioavailability of Trace Elements: Progress, Challenges and Future Trends", GeoSpec2010 Conference, Gibbs Energy Minimalization Sector (GEMS) teaching, University of Lancaster, UK, 7-8 Sept. 2010

Manera A.
Lectures given in the Course: Multiphase Flow (151-1906-00L), ETHZ, Zurich, Switzerland, Spring Semester, 2010

Manera A.
Lectures given in the Course: Nuclear Reactor's Laboratory Course (151-0162-00L), ETHZ, Zurich, Switzerland, Spring Semester, 2010

Manera A.
"Störfälle in Kernkraftwerken – Ursachen und Auswirkungen", "Sicherheit moderner Kernkraftwerke", Lectures given in the Course: SGK-Grundlagenseminar – Einführung in die Kernenergie und ihr Umfeld, SGK, Kurszentrum Bundesamt für Sport, Magglingen, Switzerland, 5-7 Oct. 2010

Mikityuk K.
"Multi-physics, coupled calculations for reactor dynamics", "Generation IV fast reactor systems", Lectures given in the Course: Special Topics in Reactor Physics (151-0166-00L), ETHZ, Zurich, Switzerland, Spring Semester, 2010

Niceno B.
"Essentials of Multiphase Flows", "Continuum Phase Modeling", "Wall Modeling", "Introduction to Multiphase Flow Modeling", "Two-Fluid Model", "Interface Tracking Techniques", "Mechanistic Modeling of Boiling", Lectures given in Master Course: Computational Multiphase Thermal Fluid Dynamics, ETHZ, Zurich, Switzerland, Spring Semester, 2010

Pelloni S.
"Fast reactor neutronics and perturbation theory", Lecture given in the Course: Special Topics in Reactor Physics (151-0166-00L), ETHZ, Zurich, Switzerland, Spring Semester, 2010

Pfingsten W.
Lectures given in the Course: Modelling of Processes in Soils and Aquifers (701-1334-00L), ETHZ, Zurich, Switzerland, Spring Semester, 2010

Pfingsten W.
"Transport processes and coupling of transport and chemistry", International Training Course (ITC) on Transport and Retention of Radionuclides in Argillaceous and Fractured Media, Paul Scherrer Institute, Villigen, Switzerland, 30 Nov. – 7 Dec. 2010

Prasser H.-M.
"Allg. Merkmale der thermodynamischen Zyklen von Kernanlagen", "Elemente der Thermohydraulik", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul B: Thermodynamik, Thermohydraulik & Sicherheit", Paul Scherrer Institute, Villigen PSI, Switzerland, 13 Jan. 2010

Prasser H.-M.
"Thermohydraulik bei Transienten und Kühlmittelverluststörfällen", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul B: Thermodynamik, Thermohydraulik & Sicherheit, Paul Scherrer Institute, Villigen PSI, Switzerland, 16 Feb. 2010

Prasser H.-M.
"Charakter und Arten von Störfällen der Sicherheitsebene 3", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul G: Sicherheit und Risiko, Paul Scherrer Institute, Villigen PSI, Switzerland, 8 March 2010

Seifert H.-P.
"Werkstoffalterung", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul E: Nukleare Werkstoffe und Wasserchemie, Paul Scherrer Institute, Villigen PSI, Switzerland, 1 March 2010

Seifert H.-P., Ritter S., Leber H.
"Werkstoffalterung im Primärkreislauf von LWR – aktuelle Risskorrosions-Forschungsaktivitäten aus dem INTEGER-Programm", SGK Herbstseminar, Langzeitbetrieb von KKW: Ausgewählte Beiträge zur Technik, Wirtschaftlichkeit und Aufsicht, Olten, Switzerland, 30 Nov. 2010

Smith, B.L.
"Introduction to Computational Fluid Dynamics", "Governing Equations, Turbulence Modelling and Numerical Procedures", "Identification of Nuclear Reactor Safety Issues where Single-Phase CFD can bring Real Benefits", "Error Control, Verification, Validation and Best Practice Guidelines", "Assessment Databases for Single-Phase CFD Applications with Emphasis on Nuclear Reactor Safety Issues", Lectures given at IAEA Regional Workshop on Application of Computational Fluid Dynamics Codes to Nuclear

Safety, University of Zagreb, Croatia, 22-25 March, 2010.

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Suter R.
"Thermohydraulik im Normalbetrieb von DWR und SWR", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul B: Thermodynamik, Thermohydraulik & Sicherheit, Paul Scherrer Institute, Villigen PSI, Switzerland, 18 Jan. 2010

Uldry A.C.
"Modelling Fe-Cr alloys by first-principles calculations and synchrotron X-ray measurements", Solid-State Physics Seminar, University of Zurich, Switzerland, 5 May 2010

Van Loon L.R.
"Diffusion and sorption of radionuclides in argillaceous materials: practical aspects and diffusion of radionuclides in argillaceous materials", International Training Course (ITC) on Transport and Retention of Radionuclides in Argillaceous and Fractured Media, Paul Scherrer Institute, Villigen, Switzerland, 30 Nov. – 7 Dec. 2010

Venz H[1]., Bart G.
"Wasserchemie (DWR, SWR) und Korrosion", Swissnuclear-PSI Fortbildungskurs Kerntechnik. Modul E: Nukleare Werkstoffe & Wasserchemie, Paul Scherrer Institute, Villigen PSI, Switzerland, 23 Feb. 2010
[1] KKB, Beznau, CH

Zimmermann M.A.
"Advanced Topics in Nuclear Reactor Materials", Lecture given in the Nuclear Energy Master Program, ETHZ, Zurich, Switzerland, Autumn Semester, 2010

Habilitation, Doctoral, Master and Bachelor Theses

Adams R.
"Coupled Neutronic/Thermal-Hydraulic Simulation of Selected Phénix End-of-Life Tests", EPFL/ETHZ MS Thesis, Sept. 2010

Epiney A.
"Improvement of the Decay Heat Removal Characteristics of the Generation IV Gas-Cooled Fast Reactor", Doctoral Thesis No. 4792, EPFL, Lausanne, July 2010

Feliciani C.
"Development of an isotope dilution technique for the quantitative analysis of fission gases in nuclear fuels", Master Thesis "nuclear engineering", EPFL/ETHZ MS Thesis, April 2010

Pittarello R.
"Monte Carlo Analysis of the H.B. Robinson II Reactor Pressure Vessel Dosimetry Benchmark", EPFL/ETHZ MS Thesis, April 2010

Pont Ribas A.
"Simulation of CABRI loss-of-flow experiments with the TRACE/FRED codes", EPFL/ETHZ MS Thesis, March 2010

Ramesh M.
"Thermomechanical Fatigue Behavior of Austenitic Stainless Tube and Pipe Steels under Light Water Reactor Relevant Temperature Conditions and Associated Microstructural Evolution", Doctoral Thesis No. 19137, ETHZ Zurich, Switzerland, June 2010

Rozov K.
"Stability and solubility of hydrotalcite-pyroaurite solid solutions: synthesis, characterization and thermo-dynamic modeling", Doctoral Thesis No. 07-106-321, University of Berne, Oct. 2010

Schreier T.
"Bau einer Thermoschockanlage", Bachelor Thesis, Fachhochschule Nordwestschweiz (FHNW), Windisch, Switzerland, Aug. 2010

PSI and Other Reports

Bradbury M.H., Baeyens B.
"Comparison of the reference Opalinus Clay and MX-80 bentonite sorption databases used in the *Entsorgungsnachweis* with sorption databases predicted from sorption measurements on illite and montmorillonite", PSI-Bericht Nr. 10-09, Nagra NTB 09-07

Bradbury M.H., Baeyens B., Thoenen T.
"Sorption data bases for generic Swiss argillaceous, crystalline and calcareous rock systems", PSI-Bericht Nr. 10-03, Nagra NTB 09-03

Dreier J., Smith B.L.
"NES Scientific Highlights 2009" (ISSN 1663-7380), Paul Scherrer Institute, July 2009

Güntay S.
"Implementation of Severe Accident Management Measures (ISAMM 2009)", PSI-Bericht Nr. 10-07

Hermann A.
"Autoclave Corrosion of Zircaloy-4 Cladding Samples in LiOH Solutions", PSI-Bericht Nr. 10-02

Tóth I.[1], Prior, R.[2], Sandervag O.[3], Umminger K.[4], Nakamura H.[5], Muellner N.[6], Cherubini M.[6], Del Nevo A.[6], D'Auria F.[6], Dreier J., Alonso J.R.[7], Amri A.[8]
"Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactors", OECD Nuclear Energy Agency Report, CSNI/R(2010)9, 2010

[1] AEKI, HU
[2] AREVA-NP SAS, Paris, FR
[3] SSM, SE
[4] AREVA, Erlangen, DE
[5] AEKI, HU
[6] University of Pisa, IT
[7] CSN, ES
[8] OECD-NEA, FR

General Communications and Public Relations

Cavedon J.-M.
"Evolution de la technologie nucléaire générations III et IV", Séminaire romand de lancement de la campagne nucléaire, economiesuisse, Lausanne, Switzerland, 1 Sept. 2010

Hardegger P.
"Die Sicht der Forschung: Konfrontation mit der Wirklichkeit", Energie Nachrichten, Schweizerischer Energierat, Sonderausgabe zum 21st World Energy Congr. (ISSN 1660-6833), Montreal, Canada, 11-16 Sept. 2010

Hardegger P.
"Berichterstattung über den Weltenergiekongress WEC 2010 Montreal aus der Sicht der Forschung: Traum und Wirklichkeit", Generalversammlung Energierat, Zurich, Switzerland, 25 Oct. 2010

Hardegger P.
"Berichterstattung vom Weltenergiekongress, WEC 2010 Montreal aus der Sicht der Forschung: Traum und Wirklichkeit", Podiumsdiskussion, Energy Science Center, ETHZ, Zurich, Switzerland, 11 Nov. 2010

Hardegger P.
"CME/WEC 2010 Montréal: Vue de la Recherche, Rêve et Réalité", Seminar Series, Energy Center, EPFL, Lausanne, Switzerland, 16 Nov. 2010

Pfingsten W.
"Neue analytische Forschungsmethoden am PSI für die sichere Tiefenlagerung radioaktiver Abfälle", SGK Apéro, Schweizerische Gesellschaft der Kernfachleute, Baden, Switzerland, 7 Sept. 2010

Awards

Abolhassani S.
"Corrosion", Swissnuclear Project of the Year 2009, Olten, Switzerland, 2 June 2010

Chenu A.
Best paper: Eur. Nuclear Conf. 2010 (4th ENEN PhD Event 2010), Barcelona, Spain, 3-6 June 2010

Froideval A.
Poster award (2nd prize): "Towards new opportunities in nuclear materials science and technology at the proposed SwissFEL X-ray laser", Eur. Materials Research Society (E-MRS), Spring Meeting, 7-11 June 2010, Strasbourg, France, 2010

Orlov A.
Poster Competiton Award: Nuclear Plant Chemistry Conf. (NPC 2010), Quebec City, Canada, Oct. 3-7, 2010

Membership of External Committees

Berner, U.
- Member of the International Scientific Committee of the Migration Conference on Chemistry and Migration Behaviour of Actinides and Fission Products in the Geosphere

Cavedon J.-M.
- Member of the KNS (Swiss Federal Nuclear Safety Commission)
- Member of the Advisory Board of the French Institut de Radioprotection et de Sûreté Nucléaire IRSN
- Member of the Advisory Board of OECD/NEA/CSNI (Committee on the Safety of Nucxlear Installations)
- Swiss representative to the Policy Group of the Generation IV International Forum
- Member of the Board of the Swiss Nuclear Forum

Chawla R.
- Vice Chairman of the OECD/NEA Nuclear Science Committee (NSC)
- Invited Expert on the Scientific Committee of the French CEA's Direction de l'Energie Nucléaire (DEN)
- Member of the Editorial Board of Annals of Nuclear Energy
- Member of the Training and Academic Affairs Committee of the Eur. Nuclear Education Network (ENEN)

Degueldre C.
- Member of the scientific review committee of the ESRF, Grenoble, Section: Applied Materials and Engineering
- Chairman of the Eur. Materials Research Society (E-MRS)

Hoffelner W.
- Invited expert of the ASME Section III Div. 5 Code development for gas and liquid metal cooled reactor
- Organizer of the ANS Nuclear Fuels and Structural Materials Symposium
- Leitung DGM Fachausschuss für Untersuchungen mit Strahllinien

Manera A.
- Member of the Editorial Board: Science and Technology of Nuclear Installations
- Vice-President of the Schweizerische Gesellschaft der Kernfachleute (SGK)

Ritter S.
- Vice-Chairman of the Executive Committee of the Eur. Cooperative Group on Corrosion Monitoring of Nuclear Materials (ECG-COMON)

Seifert H.-P.
- Member of the Executive Committee of the International Cooperative Group on Environmentally Assisted Cracking of Water Reactor Materials (ICG-EAC)

Smith B.L.
- Chairman of Governing Board, THINS, EU 7th FWP
- Chairman of the OECD/NEA Working Group on the Analysis and Management of Accidents (WGAMA) CFD Special Group

Streit M.
- Vice President of the European Nuclear Society
- Programm Committee Member of the European Nuclear Society
- Member of the Board of the Swiss Nuclear Society
- Member of the Board of Directors of the International Youth Nuclear Congress

Van Loon L.R.
- Member of the International Scientific Committee of the Migration Conference on Chemistry and Migration Behaviour of Actinides and Fission Products in the Geosphere
- Member of the Editorial Board: Transport in Porous Media

Wieland E.
- Member of the Editorial Board: The Open Waste Management Journal

Zimmermann M.A.
- Swiss representative to the Committee for the Safety of Nuclear Installations (OECD/CSNI)
- Chairman of the CSNI Expert Group Safety Margin Assessment and Application (SM2A)
- Member of the “Comité de Visite de l’Institut de Radioprotection et Sûreté Nucleaire (IRSN)”
- Swiss representative to the GIF International Expert Group

LEA – Laboratory for Energy Systems Analysis

The Energy Departments (NES and General Energy)

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Membership of External Committees

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