Joint Projects and Other Co-operative Projects

NUCLEAR SAFETY RESEARCH

The Halden Reactor Project

The Halden Reactor Project has been in operation for 47 years and is the largest NEA project. It brings together an important international technical network in the areas of nuclear fuel reliability, integrity of reactor internals, plant control/monitoring and human factors. The programme is primarily based on experiments, product development and analyses, carried out at the Halden establishment in Norway, and is supported by approximately 100 organisations in 20 countries.

Halden, Norway

Entrance to the Halden reactor, Norway.

The 2005 programme of work in the fuel area included the continuation of the in-pile loss-of-coolant accident (LOCA) test programme aimed at assessing high burn-up fuel behaviour under accident conditions. A test using high burn-up fuel was carried out, but did not perform as expected and will need to be repeated early in 2006. Properties of UO_2, gadolinia and MOX fuels in a variety of conditions relevant to operation and licensing were investigated. Corrosion and creep behaviour of various alloys were studied. The experimental programme on the effect of water chemistry variants on fuel and reactor internals material continues to be of great interest. Tests to investigate the cracking behaviour of reactor internals material in BWRs and PWRs continued, with the aim of characterising the effect of hydrogen addition to the coolant water. The programme on human factors focused on tests and data analyses carried out in the Halden man-machine laboratory. The work on human factors also encompasses new designs and evaluations of human-system interfaces and control rooms. This involves *inter alia* the use of the Halden Virtual Reality Facility. Progress has been made in the area of human reliability assessment, aiming to provide data suitable for probabilistic safety assessments. The work on cable ageing has yielded a promising technique for online cable assessment.

An Enlarged Halden Programme Group Meeting (bringing together both programme representatives and participating country experts) was held in October 2005, with almost 300 participants attending. The main results of the joint programme were reported on that occasion. A number of international workshops were also organised, mainly with the purpose of discussing new programme items and goals.

The Halden Reactor Project operates by way of three-year renewable mandates. The project's signatory organisations confirmed their strong support of the Halden activities for the 2006-2008 period during a meeting held at NEA headquarters in December 2005.

The Cabri Water Loop Project

The Cabri Water Loop Project is investigating the ability of high burn-up fuel to withstand the sharp power peaks that can occur in power reactors due to rapid reactivity insertion in the core (RIA accidents). It involves substantial facility modifications and upgrades and consists of 12 experiments to be performed with fuel retrieved from power reactors and refabricated to suitable length. The project began in 2000 and will run for eight years. The experimental work is being carried out at the Institute for Radiological Protection and Nuclear Safety (IRSN) in Cadarache, France, where the Cabri reactor is located. Programme execution also involves laboratories in participating organisations for fuel preparation, post-irradiation examinations and test channel instrumentation. Organisations in 12 countries, including regulators, industry and research organisations, participate in the project.

The examination and analyses of the tests that have been carried out so far were completed in 2005. The planning of future tests continued, with the aim of developing a consistent set of objectives and identifying suitable fuel specimens. Further progress was made on the refurbishment of the Cabri test facility and the preparation of the water loop installation.

Two meetings of the Technical Advisory Group (TAG) took place in 2005, during which the programme results and plans for future activities were reviewed. Related analyses were also presented and discussed in a Cabri seminar held in conjunction with a TAG meeting. The TAG also addressed technical issues related to the water loop design. One meeting of the project Steering Committee was held in 2005: the extension of the Cabri Umbrella Agreement to 2010 was finalised, and considerable progress was made in relation to the future participation of Japan in the project. The French *Commissariat à l'énergie atomique* became a new participant in the project in 2005.

The MASCA Project

The first phase of the Material Scaling (MASCA) Project investigated the consequences of a severe accident involving core melt. It started in mid-2000 and was completed in July 2003. The second phase of the project started thereafter, upon request of the member countries and recommendation of the CSNI. The programme, to last three years, is supported by organisations in 17 countries. It is based on experiments that are mainly carried out at the Kurchatov Institute in the Russian Federation, and that make use of a variety of facilities in which corium compositions prototypical of power reactors can be tested.

The tests in the first phase of the programme were primarily associated with scaling effects and coupling between thermal-hydraulic and chemical behaviour of the melt. The tests of the second phase seek to provide experimental information on the phase equilibrium for the different corium mixture compositions that can occur in water reactors. This determines the configuration of materials in the case of stratified pools, and thus the thermal loads on the vessel. In order to extend the application of MASCA results to reactor cases, the influence of an oxidising atmosphere and the impact of non-uniform temperatures (presence of crusts or solid debris) will be addressed in addition to scaling effects. The programme is also intended to generate data on relevant physical properties of mixtures and alloys that are important for the development of qualified mechanistic models.

Two meetings of the project steering bodies, supported by the NEA, were held in 2005. During these meetings, the results obtained to date and plans for future tests were reviewed. Discussions were also held to assess the possible need for a new programme at the Kurchatov Institute facilities after completion of the MASCA programme in June 2006. These discussions are expected to be completed during 2006.

The MCCI Project

The Melt Coolability and Concrete Interaction (MCCI) Project is managed by the US Nuclear Regulatory Commission (NRC), carried out at the Argonne National Laboratory (USA), and has participants from 13 countries. It was started early in 2002 and was completed at the end of 2005. It addressed ex-vessel phenomena which occur in the hypothetical case that the molten core is not retained inside the reactor vessel and is spread in the reactor cavity where it can interact with the concrete structure.

The MCCI Project provided experimental data of relevance to the type of severe accident mentioned above and to resolve two important accident management issues. The first one concerned the verification that the molten debris that has spread on the base of the containment can be stabilised and cooled by water flooding from the top. The second issue concerned the two-dimensional, long-term interaction of the molten mass with the concrete structure of the containment, as the kinetics of such interaction is essential for assessing the consequences of a severe accident.

The experiments on water ingress mechanisms showed that cooling of the melt by water is reduced at increasing concrete content, i.e. cooling by water flooding is more effective in the early phase of the melt-concrete interaction. The effect of concrete type, i.e. siliceous and limestone types (used respectively in Europe and the United States), was also addressed. Material properties such as porosity and permeability were derived. After a first melt-concrete interaction test, which produced unexpected results (i.e. a strong asymmetry in concrete ablation), two new tests were carried out in 2004 and 2005. The tests were successful and provided excellent data on axial and radial concrete ablation. They also pointed out appreciable differences in ablation rate for siliceous and limestone concrete, although this aspect requires confirmation. Analytical exercises were organised among participants as blind predictions of the test results, which were very valuable in order to understand code capabilities and shortcomings. The strength of the solid upper crust, a parameter that is of great interest for modelling and understanding MCCI at plant scale, was also determined during these experiments.

Two Programme Review Group meetings and two Management Board meetings were held in 2005, both supported by the NEA. On these occasions the scope of a possible extension of the project was discussed, leading to the formulation of a new programme proposal that was brought to the CSNI in December 2005. On that occasion, the decision was taken to extend the MCCI work programme for approximately three years.

The PKL Project

This project started in 2004 and consists of experiments carried out in the *Primär Kreislauf* (PKL) thermal-hydraulic facility, which is operated by Framatome ANP in its establishment at Erlangen, Germany. Organisations from 14 countries participate.

The PKL experiments focus on the following PWR issues that are currently receiving great attention within the international reactor safety community:

- boron dilution events after small-break, loss-of-coolant accidents (LOCAs);
- loss of residual heat removal during mid-loop operation with a closed reactor coolant system in context with boron dilution;

- loss of residual heat removal during mid-loop operation with an open reactor coolant system;
- an additional test to be defined in agreement with the project partners according to the state of open issues such as:
  – boron precipitation during large-break LOCAs, or
  – boron dilution after steam generator tube rupture.

Two tests were carried out in 2005. Their preparation and the first test outcome were extensively discussed at the two meetings of the project steering bodies that took place during the year. A workshop covering an analytical exercise with code predictions related to the PKL tests was also conducted in 2005. The project is set to continue until the end of 2006.

The PSB-VVER Project

The objective of the PSB-VVER Project is to provide experimental data of relevance to the validation of safety codes in the field of VVER-1000 thermal-hydraulics. The project, in which seven countries participate, started in 2003 and will be completed at the end of 2006. It consists of five PSB-VVER experiments addressing:

- scaling effects;
- natural circulation;
- small, cold leg break LOCAs;
- primary to secondary leaks;
- 100% double-ended, cold leg break (indicative, actual size to be agreed upon).

Extensive pre- and post-test analyses are to accompany the experimental programme throughout the experimental series. The possibility of setting up sets of international standard problems – either limited to project participants or with broader attendance – will also be considered in light of the resources that this effort requires.

Four project tests have been successfully carried out and reported upon thus far. The features of the final test were discussed and revised by members. This test will simulate thermal-hydraulic conditions arising after a large-break LOCA in a VVER-1000 reactor, and will be the first one run under these very demanding conditions. Two meetings of the project's Programme Review Group were held in 2005 with NEA support.

The ROSA Project

The ROSA Project was launched in 2005 to resolve issues in thermal-hydraulics analyses relevant to LWR safety, and makes use of the ROSA (Rig-of-safety assessment) large-scale test facility of the Japan Atomic Energy Agency (JAEA, formerly JAERI). It intends to focus on the validation of simulation models and methods for complex phenomena that may occur during safety transients. The project is supported by safety organisations, research laboratories and industry from 13 countries and is set to run from April 2005 to December 2009. The overall objectives of the ROSA Project are:

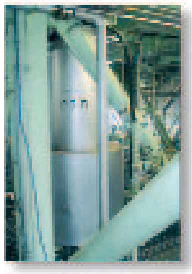
- To provide an integral and separate-effect experimental database to validate code predictive capability and accuracy of models. Phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows are to be studied in particular.
- To clarify the predictability of codes currently used for thermal-hydraulic safety analyses as well as of advanced codes presently under development, thus creating a group among member countries who share the need to maintain or improve technical competence in thermal-hydraulics for nuclear reactor safety evaluations.

The project consists of the following six types of ROSA large-scale experiments:

- temperature stratification and coolant mixing during emergency coolant injection;
- unstable and disruptive phenomena such as water hammer;
- natural circulation under high core power conditions;
- natural circulation with superheated steam;
- primary cooling through steam generator secondary depressurisation;
- two open tests defined by participants (one on pressure vessel upper-head break LOCA and another on pressure vessel bottom break LOCA, combined with accident management measures with symptom-oriented operator actions).

The first two tests were carried out as scheduled in 2005. Two meetings of the project steering bodies were held.

ROSA gamma-ray densitometer: from top to bottom view of the 9th, 8th, 7th and 6th floors.



JAEA, Japan

The SCIP Project

The Studsvik Cladding Integrity Project (SCIP) started in July 2004 and aims to utilise the hot cell facilities and expertise available at the Swedish Studsvik establishment in order to assess material properties and to determine conditions that can lead to fuel failures. The project has the following general objectives:

- to improve the general understanding of cladding integrity at high burn-up;
- to study both BWR and PWR/VVER fuel cladding integrity;
- to complement two large international projects (Cabri and ALPS), which focus on fuel behaviour in design basis accidents (notably RIA), where some of the mechanisms are similar to those that may occur during normal operational transients or anticipated transients;

- to achieve results of general applicability (i.e. not restricted to a particular fuel design, fabrication specification or operating condition), so that they can consequently be used in solving a wider spectrum of problems and be applied to different cases;
- to achieve experimental efficiency through the judicious use of a combination of experimental and theoretical techniques and approaches.

Although the primary concern of this project is the integrity of LWR cladding during reactor operation, a number of closely related areas of relevance to water reactors in general may also be addressed. In addition, some of the results will be able to be used in relation to cladding behaviour of discharged fuel during handling, transport and storage.

Organisations from ten member countries participate in the project. As recommended by the CSNI, comprehensive industry participation was sought in the project establishment phase. Two meetings of the project steering bodies were held with NEA support in November 2005.

The SETH Project

The SESAR Thermal-hydraulics (SETH) Project, which is supported by 14 NEA member countries, began in 2001 with a four-year mandate. It consists of thermal-hydraulic experiments in support of accident management, which are carried out at facilities identified by the CSNI as those requiring international collaboration to sponsor their continued operation. The tests carried out at Framatome's *Primär Kreislauf* (PKL) in Germany, which were completed in 2003, investigated boron dilution accidents that can arise from a small-break, loss-of-coolant accident (LOCA) during mid-loop operation (shutdown conditions) in PWRs. The final report of the PKL tests was completed in 2004.

The experiments being carried out at the Paul Scherrer Institute (PSI) PANDA facility in Switzerland are to provide data on containment three-dimensional gas flow and distribution issues that are important for code prediction capability improvements, accident management and design of mitigating measures. After an extensive preparation phase, the experimental series started in 2004 and continued in 2005. Due to the complexity of the PANDA experiments, some delays were encountered. The Project Board therefore decided to extend the programme's time frame to mid-2006.

Measurement of local gas concentrations inside PANDA vessels and components (mass spectrometer) at the PANDA experimental facility in Switzerland.

Paul Scherrer Institute, Switzerland

NUCLEAR SAFETY DATABASES

The COMPSIS Project

The Computer-based Systems Important to Safety (COMPSIS) Project was undertaken in 2005 by ten member countries with an initial mandate of three years. To the extent that analogue control systems are being replaced by software-based control systems in nuclear power plants worldwide, and that the failure modes of both hardware and software in these new systems are rare, there is a considerable advantage in bringing the experience of several countries together. By doing so, it is hoped to contribute to the improvement of safety management and to the quality of software risk analysis for software-based equipment.

Work during the first year of the project has concentrated on the development of the COMPSIS data collection guidelines, quality assurance and data exchange interface. Two meetings of the COMPSIS steering body were held in 2005 with NEA support.

The FIRE Project

The Fire Incidents Records Exchange (FIRE) Project started in 2002 and its mandate was renewed at the end of 2005 for a further three-year period. The main purpose of the project is to collect and analyse data related to fire events in nuclear environments, on an international scale. The specific objectives are to:

- define the format for, and collect fire event experience (by international exchange) in, a quality-assured and consistent database;
- collect and analyse fire events data over the long term so as to better understand such events, their causes and their prevention;
- generate qualitative insights into the root causes of fire events that can then be used to derive approaches or mechanisms for their prevention or for mitigating their consequences;
- establish a mechanism for the efficient feedback of experience gained in connection with fire events, including the development of defences against their occurrence, such as indicators for risk-based inspections; and to record event attributes to enable quantification of fire frequencies and risk analysis.

After having established the project quality guidelines and the quality-assurance procedure, data acquisition has proceeded according to plans. In addition, in 2005 Canada and the Netherlands joined the project, bringing the number of countries that participate to 11. Two meetings of the project steering body were held during the year.

The ICDE Project

The International Common-cause Data Exchange (ICDE) Project collects and analyses operating data related to common-cause failures (CCF), which have the potential to affect several systems, including safety systems. The project has been in operation since 1998, and a new agreement covering the period April 2005-March 2008 has come into force. Eleven countries participate.

The ICDE Project comprises complete, partial and incipient common-cause failure events. The project currently covers the key components of the main safety systems, such as centrifugal pumps, diesel generators, motor-operated valves, power-operated relief valves, safety relief valves, check valves, control rod drive mechanisms, reactor protection system circuit breakers, batteries and transmitters. These components have been selected because several probabilistic safety assessments have identified them as major risk contributors in the case of common cause failures.

Qualitative insights from data will help reduce the number of failure events that are risk contributors. Reports have been produced for pumps, diesel generators, motor-operated valves, safety and relief valves, check valves and batteries. Data exchange for switchgear and breakers, reactor level measurement and control rod drive component exchange is ongoing. Two meetings of the project steering body were held in 2005 with NEA support.

The OPDE Project

The Piping Failure Data Exchange (OPDE) Project started in 2002. The first phase of the project period was successfully completed in mid-2005. The project was then renewed for another three-year period until mid-2008. Currently, 12 countries participate. The project goals are to:

- collect and analyse piping failure event data to promote a better understanding of underlying causes, impact on operations and safety, and prevention;
- generate qualitative insights into the root causes of piping failure events;
- establish a mechanism for efficient feedback of experience gained in connection with piping failure phenomena, including the development of defence against their occurrence;
- collect information on piping reliability attributes and factors of influence to facilitate estimation of piping failure frequencies.

The OPDE Project is envisaged to include all possible events of interest with regard to piping failures in the main safety systems. It will also cover non-safety piping systems that, if leaking, could lead to common-cause initiating events such as internal flooding of vital plant areas. Steam generator tubes are excluded from the OPDE project scope. Specific items may be added or deleted upon decision of the Project Review Group. Two meetings of this body were held in 2005 with NEA support.

RADIOACTIVE WASTE MANAGEMENT

The Co-operative Programme on Decommissioning (CPD)

The Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD) is a joint research project which has been operating under Article 5 of the NEA Statute since its inception in 1985. A revised Agreement between participants came into force on 1st January 2004 for a period of five years. Currently, 20 organisations from 11 NEA member countries and one non-member economy participate in the CPD, providing experience from 41 decommissioning projects (26 reactors and 15 fuel cycle facilities). Altogether 49 decommissioning projects have benefited from the information exchange framework provided by the CPD. The information exchange includes biannual meetings of the Technical Advisory Group (TAG), during which the site of one of the participating projects is visited, and where positive and less positive examples of decommissioning experience are openly exchanged for the benefit of all. In 2005, TAG meetings were held in Tsuruga, Japan, and in Cadarache, France.

Cross-membership of some of the programme's Management Board in the RWMC Working Party on Decommissioning and Dismantling (WPDD) ensures that insight gained within the CPD can also benefit the work of the NEA standing technical committees. In this context, the CPD collected information amongst its members on the state of the art in measuring contamination levels of materials designated to be released from regulatory control. The CPD provided this information to the RWMC decommissioning group for review and publication in order to make its experience available to all NEA member countries.

UKAEA, United Kingdom

Decommissioning activities in the United Kingdom.

The Sorption Project

The NEA Sorption II Project was launched in October 2000 with the objective of demonstrating the applicability of different chemical thermodynamic modelling approaches to support safety assessments of geological repositories. To enable an evaluation of the respective merits and limitations of different thermodynamic sorption models, the project was implemented in the form of a comparative modelling exercise based on selected datasets for radionuclide sorption by both simple and complex materials. These were organised into seven test cases that were prepared and distributed to participating organisations. A Technical Direction Team evaluated the existing database, developed test cases for sorption modelling, and carried out the subsequent analysis and interpretation of modelling outcomes. Eighteen funding organisations from thirteen countries joined phase II of the Sorption Project, and in total, twenty modelling teams participated in the exercise, making it possible to base the conclusions of the project on a broad range of experience and expertise.

The findings of the project have been published as an NEA report addressing an audience of radioactive waste management organisations and regulators, as well as modellers and experimentalists who are involved in performance assessment. The report summarises the main results and identifies the strengths and drawbacks of various typical approaches. The results show that:

- the conceptual and methodological tools needed for characterising, interpreting and justifying the equilibrium distribution coefficients (Kd values) provided for performance assessment needs, are largely available;
- with regard to complex materials, the main need is for good quality and more complete sets of pertinent sorption data.

Phase II of the Sorption Project was completed with a final workshop in Paris in October 2005. The workshop provided an overview of the main project results, with emphasis on the merits and limitations of thermodynamic sorption models (TSMs) and recommendations on their use.

The Thermochemical Database (TDB) Project

The Thermochemical Database (TDB) project aims at meeting the specialised modelling requirements for safety assessments of radioactive waste disposal sites. Chemical thermodynamic data are collected and critically evaluated by expert review teams and the results are published in a book series edited by the Data Bank. The French *Commissariat à l'énergie atomique* recently joined the TDB project, bringing the number of TDB participants to 17 organisations in 12 member countries.

Four new reviews were published in 2005, bringing the total number of volumes in the TDB series to nine. The new reviews contain inorganic chemical thermodynamic data of nickel, selenium and zirconium, as well as organic compounds and complexes of uranium, neptunium, plutonium, americium, selenium, nickel, technetium and zirconium with oxalate, citrate, EDTA and iso-saccharinate ligands.

Work also continued on the reviews of thorium, tin and iron. The first two reviews are scheduled for peer review in 2006 and the latter in early 2007. A state-of-the-art report on chemical thermodynamics of solid solutions was prepared in 2005. A final version, containing precise scientific guidelines on the subject, is envisaged for publication in 2007.

RADIOLOGICAL PROTECTION

The Information System on Occupational Exposure (ISOE)

Since its creation in 1992, the Information System on Occupational Exposure (ISOE) has been facilitating the exchange of data, analysis, lessons and experience in occupational radiological protection at nuclear power plants worldwide. Jointly sponsored by the IAEA, the ISOE programme includes 478 reactor units (403 operating and 75 in cold-shutdown or some stage of decommissioning) operated by 71 utilities in 29 countries. ISOE databases cover 91% of all nuclear power reactors (442) in commercial operation throughout the world. In addition, the regulatory authorities of 25 countries participate actively in ISOE. Utilities and authorities continue to join the ISOE programme, notably the new units that have recently come on line in Korea and Japan, as well as several recent participants from the United States.

The database and information exchange mechanism used initially was the floppy disk, which then evolved to the CD. With the increasing use and flexibility of the web, it was recommended in 2003 that the ISOE programme should migrate its data exchange/assessment processes, as well as its information and experience sharing to the web. Following a pilot study in 2004, and assessment by the ISOE Working Group on Data Assessment, it was agreed that the ISOE databases should be transferred to a web-compatible database system, in order to create a natural network for the online exchange of information and experience. During 2005, the data viewing and analysis component was successfully transferred to the web as part of the new ISOE network information portal. The databases will continue to be maintained on CD for those with specific national requirements or without access to the web.

In substantive terms, the ISOE programme continued to concentrate on the exchange of data, analysis, good practice and experience in the area of occupational exposure reduction at nuclear power plants. The four regional ISOE Technical Centres continued to support their regional members though specialised data analyses and benchmarking visits. ISOE information and experience exchange continued through the international and regional ISOE ALARA symposia, including the first Asian ALARA Symposium in Japan. Finally, a new initiative was launched in 2005 to improve the usefulness and accessibility of the ISOE programme, with a goal of it becoming a primary source of information when topics of occupational radiological protection are discussed. This will be facilitated through the proactive identification of user needs and the ongoing migration of resources to the unified, web-based ISOE network portal.