Joint Projects and Other Co-operative Projects

NUCLEAR SAFETY RESEARCH

The Halden Reactor Project

The Halden Reactor Project is operated by the Norwegian Institute for Energy Technology (IFE). It has been in operation since 1958 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 prototype developments and analyses carried out at the Halden establishment in Norway. It is supported by approximately 100 organisations in 18 countries. The Halden Project benefits from stable and well-experienced organisation and a technical infrastructure that has undergone substantial developments throughout the years. The project objectives have been continuously adapted to users' needs.

In 2008, work in the fuel area included continued testing under loss-of-coolant accident (LOCA) conditions, carried out with high burn-up fuel. These are the only LOCA tests that are currently performed in-pile worldwide, and complement the work done at laboratory scale in other institutions, notably in Japan and the United States. The tests carried out have provided valuable insights and have been the basis for benchmarking exercises carried out in the Working Group on Fuel Safety Properties of UO_2, gadolinia and MOX fuels under a variety of conditions relevant to operation and licensing. Long-term irradiations have been carried out with advanced and standard nuclear fuel at high initial rating conditions. Corrosion and creep behaviour of various alloys were studied. The experimental programme on the effect of water chemistry variants on fuel and reactor internals materials has been expanded. Tests to investigate the cracking behaviour of reactor internals materials in BWRs and PWRs continued, with the aim of characterising the effect of water chemistry and material ageing. The work on cable ageing has resulted in a technique that is being used for assessing insulation damage, and in those cases to determine the extent and location of the damage.

The programme on human factors has focused on experiments in the Halden man-machine laboratory, related data analyses, new control station designs, evaluations of human-system interfaces, process and instrumentation optimisation and digital instrumentation and control (I&C). This involves *inter alia* the use of the Halden Virtual Reality Facility. Progress has been made in the area of human reliability assessment (HRA), aiming to provide data suitable for probabilistic safety assessments and to improve validity of HRA methods.

To mark the 50th anniversary of the Halden Reactor Project, the CNRA and the CSNI summer meetings were held in Oslo, in conjunction with the Halden Board of Management. An Enlarged Halden Programme Group meeting (bringing together both programme representatives and participating country experts) was held in May 2008. The main results of the programme were reported on that occasion. Another Programme Group meeting was held in Sweden in October. The Halden Board also met twice in 2008. During the last Board meeting, which was hosted by Électricité de France in Lyon in December, the project's extension for the period 2009-2011 was decided, with all member countries confirming their participation.

The BIP Project

The Behaviour of Iodine Project (BIP), which is supported by 13 member countries, began in 2007. The work consists of separate effect and modelling studies that will augment and complement larger national and international experimental programmes. In addition, it will provide data and interpretation from three Radioiodine Test Facility (RTF) experiments. The proposed project for iodine experiments, to be hosted by Atomic Energy of Canada Limited (AECL), will combine international resources to achieve a consolidated understanding of the behaviour of iodine and other fission products in post-accident nuclear reactor containment buildings. This will be accomplished by:

- addressing technical issues and scientific gaps;
- optimising the use of existing data and test results to support common tools for predicting fission product behaviour.

Specific technical objectives that this programme hopes to achieve are:

- quantification of the relative contributions of homogeneous bulk aqueous phase processes, homogeneous aqueous phase processes in paint pores and heterogeneous processes on surfaces to organic iodine formation;
- the measurement of adsorption/desorption rate constants on containment surfaces as a function of temperature, relative humidity and carrier-gas composition;
- the provision of RTF data to participants, for use in collaborative model development and validation.

Two meeting of the project steering bodies were held in 2008 and were devoted to the presentation of the first test results and to discussing the parameters and boundary conditions to be chosen for the remaining tests.

The Cabri Water Loop Project

The Cabri Water Loop Project, which began in 2000 for an eight-year period, is investigating the ability of high burn-up fuel to withstand the sharp power peaks that can occur in power reactors due to postulated rapid reactivity insertions in the core (RIA accidents). The project participants, from 13 member countries, intend to determine the limits for fuel

failure and the potential consequences of possible ejection of fuel into the coolant environment. Different cladding materials and fuel types are being studied. Project execution involves substantial facility modifications and upgrades, and consists of 12 experiments with fuel retrieved from power reactors and refabricated to suitable length. The experimental work is being carried out at the Institut de radioprotection et de sûreté nucléaire (IRSN) in Cadarache, France, where the Cabri reactor is located. Programme execution can, however, involve laboratories in participating organisations, for instance, in relation to fuel fabrication and characterisation and instrumentation.

Two tests (still using the sodium loop) were carried out with high burn-up fuel having zirconium-niobium cladding material. Fuel that had been in service in Spanish and French reactors, respectively with ZIRLO and M5 cladding, and with burn-up in excess of 70 MWd/kg, was subjected to a ~100 cal/g energy injection during the transients. No fuel failure was registered. Appreciable progress has been made in the design of the water loop test facility and in the production of related components. It will take about three years for the water loop to be in place. The Cabri tests are being complemented by additional reactivity-induced accident (RIA) tests performed in Japan. These tests, which constitute the in-kind contribution from the Japan Atomic Energy Agency (JAEA) for its participation in the Cabri Project, will be carried out at both cold and hot coolant conditions and with both BWR and PWR fuel.

A meeting of the Cabri Technical Advisory Group was held in March. A meeting of the Project Steering Committee was held in December in Paris.

The MCCI-2 Project

The aim of the Melt Coolability and Concrete Interaction (MCCI) Project is to provide experimental data on relevant severe accident phenomena and to resolve two important accident management issues. The first one concerns 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 concerns 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 programme utilises the unique expertise and infrastructure that have been developed at Argonne National Laboratory (ANL) for conducting large-scale, high-temperature reactor materials experiments. The US Nuclear Regulatory Commission (NRC) acts as the project Operating Agent.

The first phase of the programme (MCCI-1) was completed in 2005. The experiments on water ingress mechanisms showed that cooling of the melt by water is reduced at increasing concrete content, implying that 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 in the first phase of the programme. Material properties such as porosity and permeability were derived. Tests also showed appreciable differences in ablation rate for siliceous and limestone concrete, which is a relevant finding that requires confirmation. A workshop on the results of MCCI-1 was organised in France in October 2007.

A second three-year programme (MCCI-2) started in 2006. Emphasis is being placed on 2D core-concrete interaction experiments, as they provide the integrated effect of many processes. The MCCI-2 Project involves organisations from 12 member countries. Two meetings of the project steering bodies were held in 2008. On these occasions, the tests results on core-concrete interaction and the test conditions for the molten core cooling test were discussed. The next meeting is planned for April 2009 to review new results and to specify the final integral tests with which the programme will be completed.

The PKL-2 Project

A first PKL project was performed from 2004 to 2007 and consisted of experiments carried out in the Primär Kreislauf (PKL) thermal-hydraulic facility, which is operated by AREVA NP in its establishment at Erlangen, Germany. Organisations from 14 countries participated. These PKL experiments focused on the following PWR issues that have been 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; and loss of residual heat removal during mid-loop operation with an open reactor coolant system.

A second phase of the project, using the same PKL loop together with the PMK loop in Hungary and the ROCOM facility at Dresden-Rossendorf (FZD), started in 2008 with the support of 14 countries. The PKL-2 tests will investigate safety issues relevant for current PWR plants as well as for new PWR design concepts. They will focus on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations.

Two meetings of the steering bodies were held in 2008 during which the test conditions for the first series of tests were discussed.

The PRISME Project

Fire is a significant contributor to overall core damage frequency for both new and old plant designs. Questions of fire probabilistic safety analysis (PSA) that still remain open are the following:

- the propagation of heat and smoke from the room in which the fire is located to other rooms;
- the impact of heat and smoke on safety critical systems;
- the role of the ventilation network in limiting smoke and heat propagation.

The objective of the PRISME Project, which began in 2006 and in which 13 member countries participate, is to answer questions concerning smoke and heat propagation inside a plant by means of experiments tailored for code validation purposes. In particular, the project aims to provide answers regarding the failure time for equipment situated in nearby rooms and the effect of conditions such as room-to-room communi-

IRSN, France

Cable fire testing, before and after the test.

cation and the configuration of the ventilation network. The results obtained for the experimentally studied scenarios will be used as a basis for qualifying fire codes (either simplified zone model codes or computational fluid dynamics codes). After qualification, these codes could be applied for simulating other fire propagation scenarios in various room configurations with a good degree of confidence.

Tests were carried out and reported upon as scheduled in 2008. Two meetings of the project steering bodies were held in April and October. The conditions for the integral test series were addressed, including ways to support the experimental projects with analyses and code assessments. As requested by the project members, the French IRSN also prepared and submitted the plans and conditions for the four tests to be carried out in 2008, which were circulated among participants and subsequently revised according to the input received. These tests will also involve facility modifications to meet specific members' requirements.

The PSB-VVER Project

The objective of the PSB-VVER Project was 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 was initially planned for a four-year period. It was to consist 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.

Extensive pre- and post-test analyses accompanied the experimental programme throughout the experimental series.

Four project tests were successfully carried out and reported upon. The features of the fifth test were discussed and revised by members. This test was to simulate thermal-hydraulic conditions arising after a large-break LOCA in a VVER-1000 reactor, and would have been the first one run under these very demanding conditions. Difficulties encountered by the Operating Agent led to the postponement of the last test. A reduced power (10%) large-break LOCA test was conducted in January 2008 and was the subject of a benchmark exercise. Later in 2008 attempts to perform the full power test were unsuccessful and much more additional time would have been needed to achieve it. Hence, participants agreed to end the project there with the commitment from the Operating Agent that it would nevertheless attempt to carry out the full power test and would provide the results to participants if and when available.

The ROSA Project

The ROSA Project is to address issues in thermal-hydraulics analyses relevant to LWR safety using the ROSA (Rig-of-safety assessment) large-scale test facility of the Japan Atomic Energy Agency (JAEA). In particular, it will 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 in 14 countries, and will be conducted between April 2005 and December 2009. The overall objectives of the ROSA Project are to provide an integral and separate-effect experimental database to validate the code predictive capability and accuracy of models. In particular, phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows are to be studied.

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;
- open tests: upper-head break and bottom break LOCA.

The programme includes a total of 12 tests, which had all been completed by the end of 2008 except one. Two project meetings were held in 2008. Project members discussed the issues to be addressed in a possible follow-up project, and the technical basis for a new 2009-2012 project period was defined. Based on this, a new project agreement was prepared.

The SCAP Project

The Stress Corrosion Cracking and Cable Ageing Project (SCAP), which is supported by 15 member countries, began in 2006. The International Atomic Energy Agency (IAEA) and the European Commission also participate as observers. The project's main objectives are to:

- establish two complete databases with regard to major ageing phenomena for stress corrosion cracking (SCC) and degradation of cable insulation respectively;
- establish a knowledge base by compiling and evaluating collected data and information systematically;
- perform an assessment of the data and identify the basis for commendable practices which would help regulators and operators to enhance ageing management.

The project is scheduled to last four years and the scopes and structures of the databases have been defined. The project is currently focusing on populating the data and

assessing the data. The assessment report will be published at the end of the project and provide the technical basis for commendable practices in support of regulatory activities in the fields of SCC and cable insulation.

The Management Board held its third meeting in June, approved the programme of work for 2008 and 2009, and decided to hold a workshop at the end of the project.

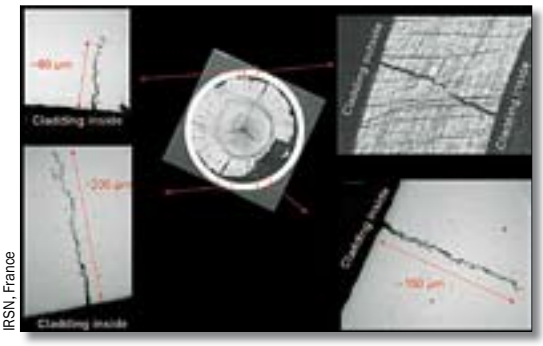
The SCIP Project

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 determine conditions that can lead to fuel failures. The project, in which 11 countries participate, has the main objective of improving the general understanding of cladding reliability at high burn-up through advanced studies on phenomena and processes that can impair fuel integrity during operation in power plants and during handling or storage. The project aims to achieve results of general applicability (i.e. not restricted to a particular fuel design, fabrication specification or operating condition). The results can consequently be used in solving a wider spectrum of problems and applied to different cases. It also aims to achieve experimental efficiency through the judicious use of a combination of experimental and theoretical techniques and approaches.

The SCIP Project has so far focused on the execution of several power ramps and on defining a hot cell programme addressing the various failure mechanisms which will be studied in the project. These are as follows:

- pellet-clad interaction (PCI): stress corrosion cracking initiated at the cladding inner surface under the combined effect of the mechanical loading and chemical environment caused by an increase in the fuel pellet temperature following a power increase;
- hydride embrittlement: time-independent fracture of existing hydrides;
- delayed hydride cracking (DHC): time-dependent crack initiation and propagation through fracture of hydrides that can form ahead of the crack tip.

The programme has been progressing very satisfactorily, producing evidence that is relevant for understanding the factors leading to cladding brittleness and the methods for reproducing in hot cell tests the stress-strain conditions that prevail in fuel power ramps. Two meetings of the project steering bodies took place in 2008. At the December 2008 meeting, all members clearly indicated their interest in continuing the project for a new five-year period. In addition, the project has been enriched with a new LOCA programme promoted by the USNRC on integral LOCA testing.



IRSN, France

Cross-section of ramp-tested BWR cladding showing incipient cracks on the outer and inner cladding surface.

The SERENA Project

The Steam Explosion Resolution for Nuclear Application (SERENA) Project was launched in 2007 with nine member countries participating. Its predecessor programme sought to evaluate the capabilities of the current generation of fuel-coolant interaction (FCI) computer codes in predicting steam-explosion-induced loads in reactor situations, and to identify confirmatory research that would be needed to bring predictability of FCI energetics to required levels for risk management. The programme concluded that in-vessel FCI would not challenge the integrity of the containment whereas this cannot be excluded for ex-vessel FCI. However, the large scatter of the predictions indicated lack of understanding in some areas, which makes it difficult to quantify containment safety margins to ex-vessel steam explosion. The results clearly indicated that uncertainties on the role of void (gas content and distribution) and corium melt properties on initial conditions (pre-mixing) and propagation of the explosion were the key issues to be resolved to reduce the scatter of the predictions to acceptable levels. Past experimental data does not have the required level of details to answer the question.

The present programme has been formulated to resolve the remaining uncertainties by performing a limited number of focused tests with advanced instrumentation reflecting a large spectrum of ex-vessel melt compositions and conditions, as well as the required analytical work to bring the code capabilities to a sufficient level for use in reactor case analyses. The objective of the SERENA experimental programme is threefold:

- to provide experimental data to clarify the explosion behaviour of prototypic corium melts;
- to provide experimental data for validation of explosion models for prototypic materials, including spatial distribution of fuel and void during the pre-mixing and at the time of explosion, and explosion dynamics;
- to provide experimental data for the steam explosion in more reactor-like situations to verify the geometrical extrapolation capabilities of the codes.

These goals will be achieved by using the complementary features of the TROI (Korea Atomic Energy Research Institute) and KROTOS (Commissariat à l'énergie atomique) corium facilities, including analytical activities. The KROTOS facility is more suited for investigating the intrinsic FCI characteristics in one-dimensional geometry. The TROI facility is better suited for testing the FCI behaviour of these materials in reactor-like conditions by having more mass and multi-dimensional, melt-water interaction geometry. The validation of models against KROTOS data and the verification of code capabilities to calculate more reactor-oriented situations simulated in TROI will strengthen confidence in code applicability to reactor FCI scenarios. Two meetings of the steering bodies of this project were held in 2008 and the results of the first two tests were presented. In parallel,

analytical activities were undertaken to prepare and then to assess these tests.

The SETH-2 Project

The SESAR Thermal-hydraulics (SETH) Project, supported by 14 member countries, was conducted from 2001 to 2007. It consisted of thermal-hydraulic experiments in support of accident management, which were carried out at facilities identified by the CSNI as those requiring international collaboration to sponsor their continued operation. The experiments carried out at the Paul Scherrer Institute (PSI) PANDA facility in Switzerland provided 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.

A follow-up to the project, called SETH-2, was launched in 2007 and will make use of the PANDA facility and the MISTRA facility of the French Commissariat à l'énergie atomique (CEA). Nine countries are participating. The project aims to resolve key computational issues for the simulation of thermal-hydraulic conditions in reactor containments and will benefit from the complementarity of the two facilities. Two meetings of the project steering bodies were held in 2008 and were devoted to presenting the first test results and to discussing the parameters and boundary conditions to be chosen for the remaining test.

The THAI Project

The Thermal-hydraulics, aerosols and iodine (THAI) Project, is supported by eight member countries and began in 2007. It consists of thermal-hydraulic experiments aiming at resolving uncertainties related to combustible hydrogen and to the behaviour of fission products, in particular iodine and aerosols. The proposed experiments are designed to fill knowledge gaps by delivering suitable data for the evaluation and simulation of the hydrogen and fission product interactions mentioned above, thus supporting the validation of accident simulation codes and models. The experiments are conducted in the THAI facility, which is operated by Becker Technologies GmbH in Germany. The Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) and AREVA NP GmbH also support the programme.

In the case of hydrogen, uncertainties mainly arise in relation to determining conditions for the occurrence of deflagration flames, and the performance of devices, such as passive autocatalytic recombiners, designed to reduce the concentration of hydrogen gas developed in a hypothetical accident. Some concern also exists regarding the applicability of several previous experiments where helium was used to simulate hydrogen. The relevance to reactor safety is connected with the destructive potential of fast deflagrations.

In the case of fission products, a number of transport processes have not yet been investigated to a level of detail sufficient to establish reliable transport models. Such processes include iodine exchange between turbulent atmospheres and walls, relocation by wash-down (washing the walls with condensate water), airborne chemical reaction of iodine with radiolytic ozone, and aerosol resuspension from a boiling sump. The control of volatile radioactive species is relevant to the potential accident source term and the radioactivity management.

In 2008, two meetings of the project steering bodies were held to discuss the results of the hydrogen combustion tests performed and the parameters for the remaining tests to be undertaken in 2009. Hydrogen distribution tests performed in 2007 were used to support a blind benchmark exercise in 2008, followed by an open one which has enabled a better understanding of the modelling of hydrogen mixing in steam environments. In 2009, a benchmark of a hydrogen combustion test will be conducted in co-operation with the CSNI Working Group on Analysis and Management of Accidents (WGAMA).

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. A new three-year mandate began in January 2008. 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 part of the project has concentrated on the development of the COMPSIS data collection guidelines, quality assurance and data exchange interface. Recently, countries have begun submitting data. Two meetings of the COMPSIS steering body were held in 2008 and a report on the first three years of achievements was issued.

The FIRE Project

The Fire Incidents Records Exchange (FIRE) Project started in 2002 and its current mandate runs until the end of 2009. Twelve countries participate. The main purpose of the project is to collect and to 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;

- record event attributes to enable quantification of fire frequencies and risk analysis.

The structure of the database is now well-defined and arrangements have been made in all participating countries to collect and to validate data. Similar to the OPDE Project, the group is reviewing and collecting past events in addition to events having taken place during the year. The quality-assurance process is in place and has proved to be efficient on the first set of data provided. An updated version of the database, which now contains more than 340 records, is provided to participants every year. Two meetings of the project steering body were held during 2008.

The ICDE Project

The International Common-cause Data Exchange (ICDE) Project collects and analyses operating data related to common-cause failures (CCF) that have the potential to affect several systems, including safety systems. The project has been in operation since 1998, and was extended with a new agreement covering the period April 2008-March 2011. 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 CCF events that are risk contributors, and member countries use the data for their national risk analyses. More activities in the area of quantification are under discussion. Reports have been produced for pumps, diesel generators, motor-operated valves, safety and relief valves, check valves, batteries, switchgear and breakers and reactor-level measurement. Data exchange for heat exchangers and control rod drive component exchange is ongoing.

Two project meetings were held in 2008. The next ICDE steering group meeting will take place in March 2009.

The OPDE Project

The Piping Failure Data Exchange (OPDE) Project started in 2002. A new three-year phase of the project was started in June 2008. Currently, 11 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 scope of the OPDE Project includes all possible events of interest with regard to piping failures in the main safety systems. It also covers 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. An updated version of the database is provided to participants every six months. Two Project Review Group meetings were held in 2008. A joint workshop on Risk-informed Piping Integrity Management was also organised between the OPDE Project and the CSNI Working Group on Integrity of Components and Structures (IAGE) to discuss the applications and uses of the OPDE database.

RADIOACTIVE WASTE MANAGEMENT

The CPD Programme

The NEA Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD) is a joint undertaking which functions within the framework of an agreement between 22 organisations actively executing or planning the decommissioning of nuclear facilities. It has operated under Article 5 of the NEA Statute since its inception in 1985, and a revised Agreement between participants will come into force on 1 January 2009 for a period of five years. The objective of the CPD is to acquire and to share information from operational experience in the decommissioning of nuclear installations that is useful for future projects.

The information exchange also ensures that best international practice is made widely available and encourages the application of safe, environmentally friendly and cost-effective methods in all decommissioning projects. It is based on biannual meetings of the Technical Advisory Group (TAG), during which the site of one of the participating projects is visited, and positive and less positive examples of decommissioning experience are openly exchanged for the benefit of all. Currently 39 projects under active decommissioning (23 reactors and 16 fuel facilities) are included in the information exchange.

Although part of the information exchanged within the CPD is confidential and restricted to programme participants, experience of general interest gained under the programme's auspices is released for broader use. In this context, the CPD is collecting and analysing its experience on remote dismantling techniques and on decontamination and dismantling of concrete structures. The resulting information will be provided to the relevant NEA committees and working groups for their review and consideration for publication as NEA reports. The CPD plans to finalise both draft reports in 2009.

The Sorption-3 Project

Radionuclide sorption is one of the most important processes with regard to the prevention or retardation of radionuclide migration to the biosphere and the overriding objective of the Sorption Project is to demonstrate the potential of thermodynamic sorption models to improve confidence in the representation of radionuclide sorption in the context of radioactive waste disposal. This objective will be met if it can be shown that the major physical-chemical mechanisms underlying the sorption of a radioelement by different types of solid materials are understood, and if it can be demonstrated that it is possible to represent the process-defining parameters with reasonable accuracy as a function of variations in relevant system parameters.

After a first phase of the Sorption Project (1997-1998) investigating the potential of thermodynamic models for improving the presentation of sorption in performance assessments for geological repositories, and a second phase (2000-2004) demonstrating the consistency and applicability of different thermodynamic models, a third phase of the Sorption Project was started in November 2007 with a mandate until April 2010. Organisations involved in geological disposal from 12 countries are participating in this project that comprises the production of a guideline document which addresses thermodynamic sorption model development and the use of such models in building a safety case. It will also include a workshop to discuss the draft document with interested parties. An important objective of the project is to facilitate communication with waste management organisations as well as regulatory authorities.

Since the start of the project a Technical Direction Team has been appointed to undertake the main drafting task, as have other experts to provide specific inputs and to undertake review tasks. By the end of 2008, significant progress had been achieved in drafting the first two chapters of the document.

The TDB-4 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 series edited by the Data Bank. Phase 3 of the TDB Project was completed at the end of January 2008.

A new fourth phase of the TDB Project (TDB-4) was started subsequently, with the remaining tasks of the third phase having been transferred to the new phase. Sixteen organisations from 14 countries are participating in this new phase.

Remaining tasks from Phase 3 include the completion and publication of the reviews of chemical thermodynamic data for inorganic compounds and complexes of iron (Fe) and tin (Sn). The review on thorium was sent to print for publication in January 2009. The fourth phase of the project will comprise complementary studies of inorganic species and compounds of iron, a review of auxiliary data, an update of the selected value database accrued during the first three phases of the project, and a review of inorganic species and compounds of molybdenum (Mo).

RADIOLOGICAL PROTECTION

The ISOE System

Since its creation in 1992, the Information System on Occupational Exposure (ISOE), jointly sponsored with the IAEA, has been facilitating the exchange of data, analysis, lessons and experience in occupational radiological protection (RP) at nuclear power plants worldwide. At the beginning of 2008, the new ISOE Terms and Conditions came into force. As of December 2008, membership in ISOE included 59 participating utilities in 26 countries and the regulatory authorities of 22 countries.

The ISOE programme maintains the world's largest occupational exposure database and a network of utility and regulatory authority RP experts. Four ISOE Technical Centres (Europe, North America, Asia and the IAEA) manage the programme's daily technical operations of analysis and exchange of information and experience. The ISOE occupational exposure database contains information on occupational exposure levels and trends at 470 reactors (396 in operation and 74 in cold-shutdown or some stage of decommissioning) in 29 countries, covering about 91% of the world's operating commercial power reactors. Since its inception, ISOE participants have used this dual system of databases and communications networks to exchange occupational exposure data and information for dose trend analyses, technique comparisons, and cost-benefit and other analyses promoting the application of the as low as reasonably achievable (ALARA) principle in local radiological protection programmes.

In 2008, 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, on improving the quality of its occupational exposure database and on migrating ISOE resources to the ISOE Network website. The four regional ISOE Technical Centres continued to support their regional members through specialised data analyses and benchmarking visits. ISOE information and experience exchange continued through the successful organisation and hosting of the 2008 international ALARA Symposium (in Japan) and regional ALARA symposia (in the United States and Finland).

The ISOE Network web-based information portal (www.isoe-network.net) continued to serve as a "one-stop" website for ISOE information and experience exchange. Development of web-based input modules for occupational exposure data collection were finalised in 2008, and will be implemented on the Network in 2009.

At its annual meeting, the ISOE Management Board approved the publication of a report on *Work Management to Optimise Occupational Radiological Protection in the Nuclear Power Industry*, which takes into account new experience and technology in occupational radiation dose reduction and 15 years of information exchange under the ISOE programme. The Management Board also approved a proposal for improving the data collection, analysis and experience exchange aspects of participating reactors undergoing decommissioning. The ISOE ad hoc expert group for the revision of the International Basic Safety Standards (BSS) provided, through the CRPPH, input into the BSS revision process with respect to good practice in occupational exposure.