

# 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.

The 2006-2007 work scope in the fuel area included important loss-of-coolant accident (LOCA) tests 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 France, Japan and the United States. The tests carried out have provided valuable insights, which have been confirmed in hot cell post-irradiation examinations. Properties of  $UO_2$ , gadolinia and MOX fuels in a variety of conditions relevant to operation and licensing were investigated. 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.

An Enlarged Halden Programme Group meeting (bringing together both programme representatives and participating country experts) was held in March. The main results of the programme were reported on that occasion.

A number of international workshops, such as those on Advanced Control Systems Designs, Irradiation Assisted Stress-cracking Corrosion and LOCA tests were organised in 2007, mainly to discuss the outcomes of ongoing programme items. Another Programme Group meeting was held in the Czech Republic in September. The Halden Board also met twice in 2007.

### 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:

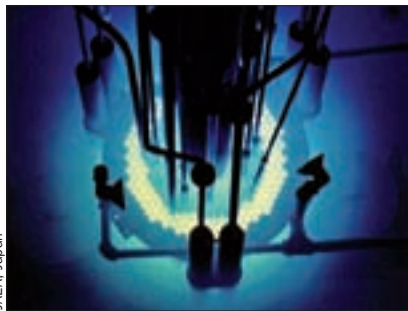
- quantification of the relative contributions of homogeneous bulk aqueous phase processes, homogeneous aqueous phase processes in paint pores and heterogeneous processes on surfaces to organic iodine formation;
- the measurement of adsorption/desorption rate constants on containment surfaces as a function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

One meeting of the project steering bodies was held in 2007 and was mainly devoted to discussing the parameters and boundary conditions to be chosen for the test matrix.

### 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. Recently, the Japan Atomic Energy Agency (JAEA) has joined the project, bringing additional high burn-up experiments carried out in the JAEA nuclear safety research pulse reactor.



Top view of the JAEA nuclear safety research reactor (NSRR).

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 occurred 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 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 January. A meeting of the Project Steering Committee was held in October in the United States.

## The MASCA-2 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, which was to last three years, was supported by organisations in 17 countries. It was based on experiments that were mainly carried out at the Kurchatov Institute in the Russian Federation, and that made use of a variety of facilities in which corium compositions prototypical of power reactors could 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 provided 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) was addressed in addition to scaling effects. The programme also sought to generate data on relevant physical properties of mixtures and alloys that are important for the development of qualified mechanistic models.

The final meeting of the project steering bodies was held in 2006 during which the results obtained and plans for the final report were reviewed. Discussions were also held to assess the possible need for a new programme at the Kurchatov Institute facilities, but did not result in a concrete proposal. The final report of the project was issued in June, and a concluding workshop was held in France in October, where the main outcomes were presented and discussed among project partners. This completes the MASCA-2 Project.

## 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 new three-year programme (MCCI-2) has been adopted by participants, which started in 2006. Emphasis is 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 2007. On these occasions, the first tests results and the test conditions for the remaining programme were discussed. The next meeting is planned for April 2008 to review new results and to specify the end of the project tests' matrix.

## The PKL Project

This project started in 2004 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.

The 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;
- loss of residual heat removal during mid-loop operation with an open reactor coolant system.

The last tests were carried out in 2006. Their outcomes were extensively discussed at the final meetings of the project steering bodies, which took place in May 2007. Although the project was officially completed in November 2007 with the publication of the final report, a proposal was distributed at the end of the year for a follow-up project to address heat transfer in steam generators and remaining issues on boron precipitation.

## 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 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 ten 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 communication 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 2007. Two meetings of the project steering bodies were held in April and October. The conditions for the entire test series were addressed in the meetings, 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 accord-

ing 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 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 in 2008. 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.

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

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. Difficulties encountered by the Operating Agent led to the postponement of the last test which has been rescheduled to take place early in 2008.

## 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 is intended 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 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 twelve tests, of which eight have been carried out so far. Four tests were performed in 2007, one on temperature stratification, one

on water hammer and two on primary cooling through depressurisation. The remaining four tests were discussed by the project steering bodies, which defined the tests' initial and boundary conditions. They will be conducted in 2008 and in the first part of 2009. Project members also discussed the issues to be addressed in a possible follow-up project. Two project meetings were held in 2007, in May in France, and in November in Japan.

## The SCAP Project

The Stress Corrosion Cracking and Cable Ageing Project (SCAP), which is supported by 14 NEA 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.



OECD Halden Reactor Project

Cable ageing specimen.

The project has been designed to last for four years and is being funded by a Japanese voluntary contribution. It is anticipated that the database definition and the collection of a representative amount of data for starting the assessment will take approximately two years. The assessment phase and the commendable practice report are expected to take one year each.

The Management Board held its second meeting in May and approved the programme of work for 2007 and 2008, as proposed by the two working groups on SCC and cables. The scopes and structures of the databases have been defined, and their formats have been finalised.

## 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 dif-

ferent 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 with NEA support in 2007.

## 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 premixing 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 a 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. The first operational meeting of this project will be held in January 2008.

### The SETH Project

The SESAR Thermal-hydraulics (SETH) Project, which is supported by 14 NEA member countries, began in 2001. 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 AREVA'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 and the Project Board therefore decided to extend the programme's duration to the end of 2006, after completion of the last three tests. The final report was completed in May 2007. A workshop was organised in June 2007, during which participants discussed the application of the results for benchmarking codes used for reactor applications.

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 2007 and were mainly devoted to discussing the parameters and boundary conditions to be chosen for the test matrix. The operating Agent has made the preparatory arrangements for performing the tests in 2008.

### 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.



Becker Technologies GmbH, Germany

Schematic view of the THAI facility.

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 re-suspension from a boiling sump. The control of volatile radioactive species is relevant to the potential accident source term and the radioactivity management.

In 2007 two meetings of the project steering bodies were held to discuss the parameters for the tests to be performed in 2008. Tests performed in 2007 are also being used to support a blind benchmark exercise, being conducted as a complementary study by a number of project participants.

## 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 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 2007 with NEA support. A new three-year phase of the project will start in January 2008.

## 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 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 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 300 records, is provided to participants every year. One meeting of the project steering body was held during 2007.

## 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 2005–March 2008. 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 and an internal seminar about the topic took place in 2007. 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. The next report to be produced will be on water level measurement.

Two project meetings were held in 2007. The next ICDE steering group meeting will take place in April 2008 in Germany. A new three-year phase is planned to follow the current one.

## The OPDE Project

The Piping Failure Data Exchange (OPDE) Project started in 2002. The first phase of the project 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 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 2007 with NEA support. One of the subjects discussed was the terms and

conditions for the third phase of the project (June 2008-May 2011).

## 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 came into force on 1 January 2004 for a period of five years. The objective of the CPD is to acquire and share information from operational experience in the decommissioning of nuclear installations that is useful for future projects. Two new organisations joined the programme during 2007: Barsebäck Kraft AB and Studsvik Nuclear AB (both located in Sweden).

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 44 decommissioning projects (28 reactors, 8 reprocessing plants and 8 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 began two studies during 2007, one on Remote Dismantling Techniques and one on Decontamination and Dismantling of Concrete Structures. The reports from the task groups undertaking these studies are scheduled to be published in 2009.

### The 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 series edited by the Data Bank. The current Phase III of the TDB Project runs until the end of January 2008. The 17 scientific institutions and technical authorities from 13 NEA member countries participating in the TDB Project decided to extend the project (TDB Phase IV) to 2012.

During 2007, a state-of-the-art report on the chemical thermodynamics of solid solutions was published. Work on the reviews of thorium, tin and iron data continued. The thorium report will be published in early 2008. The tin and

iron reports are scheduled for peer review during 2008. TDB Phase IV will begin in February 2008 and will include a review of auxiliary data, an update of the selected value database accrued during earlier phases of the project, a review of molybdenum data and a review of additional iron data.

## RADIOLOGICAL PROTECTION

### The ISOE System

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 (RP) at nuclear power plants worldwide. Sponsored jointly with the IAEA, the ISOE programme has a membership of 71 participating utilities in 29 countries, as well as the regulatory authorities of 25 countries.

The ISOE programme maintains the world's largest occupational exposure database and a network of utility and regulatory authority RP experts. Four supporting ISOE Technical Centres (Europe, North America, Asia and IAEA) manage the programme's day-to-day technical operations of analysis and exchange of information and experience. The ISOE occupational exposure database itself contains information on occupational exposure levels and trends at 481 reactor units (401 in operation and 80 in cold-shutdown or some stage of decommissioning) in 29 countries, thus covering 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 2007, 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 through specialised data analyses and benchmarking visits. ISOE information and experience exchange continued through the successful organisation and hosting of the 2007 international and regional ISOE ALARA symposia, in the United States and the Republic of Korea respectively.

The ISOE Network web-based information portal, formally launched in 2006, was considerably enhanced during 2007. The portal provides members with a "one-stop" website for ISOE information and experience exchange. In 2008, members will be able to enter their occupational exposure data into the ISOE database directly through the website.

In 2007, the ISOE Steering Group approved the programme's new Terms and Conditions for the period 2008-2011.