Joint Projects and Other Co-operative Projects

NUCLEAR SAFETY RESEARCH

The Halden Reactor Project

The Halden Reactor Project has been in operation for 48 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. It is carried out at the Halden establishment in Norway supported by approximately 100 organisations in 20 countries.

Work in the fuel area in 2006 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 in 2006 have provided valuable insights which need to find confirmatory evidence 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 material in BWRs and PWRs continued, with the aim of characterising the effect of water chemistry and material ageing. The programme on human factors focused on tests and data analyses carried out in the Halden man-machine laboratory, encompassing 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 resulted in a technique that is being used at industrial level for assessing whether cable insulation is damaged, and in those cases to determine the extent and location of the damage.

An Enlarged Halden Programme Group Meeting (bringing together both programme representatives and participating country experts) is planned for March 2007. The main results of the joint programme will be reported on that occasion. A number of international workshops, such as those on Advanced Control Systems Designs, Irradiation-Assisted Stress Corrosion Cracking, and LOCA tests were organised in 2006, mainly with the purpose of discussing the outcomes of ongoing programme items.

The Halden Reactor Project operates by way of three-year renewable mandates. The current programme will be carried out during 2006-2008. Preparations are being made for the programme's continuation in the longer term, including renewal of the Halden reactor licence.

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 first two tests (still in the sodium loop) were carried out with high burn-up fuel having zirconium niobium cladding material. Fuel of Spanish and French origins, with ZIRLO and M5 cladding respectively, and burn-up in excess of 70 MWd/kg, was subjected to a ~100 cal/g energy injection during the transients. From the evaluation of the in-reactor signals during the tests and from the non-destructive examination it appeared that neither the M5 nor the ZIRLO fuel failed.

Appreciable progress has been made in the design of the water loop test facility and in the production of related components. Almost three years will be required to put the water loop in place. In the future, the Cabri tests will be complemented by additional RIA tests that will be performed in the NSRR reactor in Japan. These tests constitute the in-kind contribution from the Japan Atomic Energy Agency (JAEA) for its participation in the Cabri Project.

A meeting of the Cabri Technical Advisory Group was held in April 2006. A meeting of the Project Steering Committee was held in October in Spain.

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, 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 was also intended 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 to date 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 following completion of the MASCA-2 Project, but did not result in a concrete proposal. After publication of the final report early in 2007, a workshop will be held in Cadarache, France in October 2007 to discuss among project partners the application of the results obtained.

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 is planned to take place in October 2007.

A new three-year programme (MCCI-2) has been adopted by participants. Emphasis will be 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. A meeting of the project steering bodies was held in Paris in April 2006. On this occasion, the test conditions for the three-year programme were discussed. A meeting is planned for early 2007 to review the status of the first project tests.

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 AREVA NP 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.

Three tests were carried out in 2006. Their preparation and the first test outcomes 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 held in 2006. The project is set to continue until May 2007 to enable completion of the final report.

Framatome ANP GmbH, AREVA NP GmbH, Germany

PKL test facility in Erlangen, Germany.

The PRISME Project

Fire is a significant contributor to the overall core damage frequency for both new and old plant designs. The objective of the PRISME Project 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 to the following:

- What is, for a given fire scenario, the failure time for equipment situated in the nearby rooms that communicate with the fire room by the ventilation network and/or by a door (which is open before the fire or opens during the fire)?
- Is it valid to assume that no propagation occurs beyond the second room from the fire room when the rooms communicate through doors, and beyond the first room when rooms communicate only by the ventilation network?
- What are the safety consequences of the damper or door failing to close, or of an intervention delay which is too long?
- What is the best way to operate the ventilation network in order to limit pressure-driven phenomena and releases to nearby rooms? Is it the admission damper closing following fire detection? Is it the extraction damper closing when the temperature threshold of filters has been reached or when the filters are plugged?

The results obtained for the experimentally studied scenarios will be used as a basis for qualifying fire codes (either simplified zone model codes or CFD codes). After qualification, these codes could be applied for simulating other fire propagation scenarios in various room configurations with a good degree of confidence. The information will be useful for designers in order to select the best fire protection strategy. For the operators, these data could be useful for establishing the suitable driving for the plant, such as the driving of the ventilation network (closing dampers, to reduce the ventilation flow rate or to stop the ventilation) in case of a fire event.

Two meetings of the Programme Review Group and Management Board were held in April and October 2006. During the last meeting, the outcome of the first test and the progress made on the accompanying analytical exercise were discussed.

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 should have been 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.

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 until April 2007.

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). The project is focusing 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. In 2006, two tests were carried out addressing temperature stratification and high-power natural circulation. Two meetings of the project steering bodies were held.

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, through collective efforts by OECD/NEA member countries;
- 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 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 first meeting of the Management Board was held in June 2006. The Board approved the project's Terms of Reference and nominated the representatives in the two working groups on SCC and cable ageing. The first working group meetings were held in September (on cable ageing) and in October (on SCC), where the scope and organisation of the databases were discussed. The databases will be set up with the support of technical institutions as clearing houses.

The SCIP Project

The Studsvik Cladding Integrity Project (SCIP) aims to clarify mechanisms and reproduce conditions that can lead to cladding failure. To achieve this objective, it utilises the hot cell facilities and expertise available at the Swedish Studsvik establishment in order to carry out the necessary testing. The project has the following overall goals:

- 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 are also addressed. New types of fuel designs and cladding materials, as well as more demanding operational modes have been introduced or are being considered in order to enhance fuel utilisation and plant efficiency, through for instance higher operating power and higher discharge burn-up. These new fuel designs need to be verified with respect to relevant performance and safety aspects, notably resistance to corrosion and resistance to pellet-clad mechanical interaction (PCMI) under normal operation conditions and during transients. Assessments should also cover conditions such as those prevailing during fuel handling and storage.

Organisations from ten member countries participate in the project. As recommended by the CSNI, there is also comprehensive industry participation. Two meetings of the project steering bodies were held with NEA support in 2006.

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 should be completed by the spring of 2007. A follow-up to the project, called SETH-2, will be launched in 2007 and make use of the PANDA facility and the MISTRA facility of the French *Commissariat à l'énergie atomique* (CEA). The project will aim to resolve key computational issues for the simulation of thermal-hydraulic conditions in reactor containments.

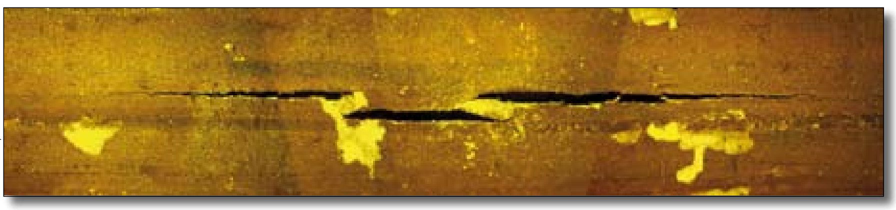
NUCLEAR SAFETY DATABASES

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

Failed, high burn-up fuel cladding after exposure to a power ramp

Studsvik, Sweden



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. Two meetings of the COMPSIS steering body were held in 2006 with NEA support.

GRS, Germany

BWR piping at the Brunsbuettel NPP in Germany after a hydrogen explosion.

The FIRE Project

The Fire Incidents Records Exchange (FIRE) Project started in 2002 and its mandate was renewed for another three-year period starting in January 2006. Eleven 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 is given to participants every year. Two meetings of the project steering body were held during 2006.

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 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 CCF events that are risk contributors, and the 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 will take 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.

Two project meetings were held in 2006. The next ICDE steering group meeting will take place in April 2007 in Sweden.

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. An updated version of the database is provided to participants every six months. Two Project Review Group meetings were held in 2006 with NEA support.

RADIOACTIVE WASTE MANAGEMENT

The Co-operative Programme on Decommissioning (CPD)

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 21 organisations actively executing or planning the decommissioning of nuclear facilities. Operating under Article 5 of the NEA Statute since its inception in 1985, 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.

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 42 decommissioning projects (26 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 collected information amongst its members for a state-of-the-art report on measuring contamination levels of materials designated to be released from regulatory control and prepared a status report of the CPD programme, describing the progress made and the main results obtained by the CPD during 1995-2005. Both reports have been issued by the RWMC Working Party on Decommissioning and Dismantling to encourage relevant NEA member organisations to consider joining the CPD.

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* joined the TDB project in 2006, bringing the number of TDB participants to 17 organisations in 12 member countries.

During 2006, work continued on the reviews of thorium, tin and iron. The thorium report is under peer review and is scheduled for publication in 2007. The tin and iron reports are scheduled for peer review during 2007. A state-of-the-art report on chemical thermodynamics of solid solutions will be published in the first part of 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. The ISOE programme is co-sponsored by the IAEA. Its membership includes 69 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 radiological protection experts from utilities and regulatory authorities. 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 480 reactor units (403 in operation and 77 in cold-shutdown or some stage of decommissioning) in 29 countries. This represents 91% of the world's operating commercial power reactors (442). 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.

Whereas the database and information exchange mechanism used initially was the floppy disk, then the CD, the data viewing and analysis component was successfully transferred to the web as part of the new ISOE Network information portal, formally launched in 2006. The ISOE Network currently has about 400 registered users from utilities and regulatory authorities who use the network to access the full range of ISOE products and share their operational radiological protection experience. In 2007, Network services will be further enhanced through the implementation of online data entry modules, and optimised functionality based on direct user feedback. The databases will continue to be maintained on CD for those with specific national requirements or without access to the web.

In 2006, 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 largely benefited from the 2006 international and regional ISOE ALARA symposia, held in Germany and Japan respectively.