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 2009, Halden began a new three-year mandate. 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 being 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 by 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 produced 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 the validity of HRA methods.

The main results of the programme were reported at two Programme Group meetings held in the Slovak Republic in May and in Norway in October. The Halden Board also met twice in 2009.

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 project for iodine experiments, hosted by Atomic Energy of Canada Limited (AECL), pools international resources to achieve a consolidated understanding of the behaviour of iodine and other fission products in post-accident nuclear reactor containment buildings. 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 2009 and were devoted to discussing the test results as well as the parameters and boundary conditions to be chosen for the remaining tests. Analytical work performed by participants enabled progress in model qualification for the iodine behaviour in the containment, and in the understanding of containment paint behaviour.

The Cabri Water Loop Project

The Cabri Water Loop Project, which began in 2000, 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 was made on the reconstruction of the reactor and the construction of the water loop test facility, with the new core envelope and the security tube of the pressurised water loop being implemented. In July 2009, a regulation hydrotest of the pressurised water loop was successfully carried out. The resumption of the tests in the framework of the Cabri Water Loop Project is expected in early 2011.

The Cabri tests are being complemented by additional reactivity-initiated 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 January 2009. A meeting of the Project Steering Committee was held in December 2009 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.

The second three-year programme (MCCI-2) started in 2006 and is to be completed in early 2010. 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 2009. On these occasions, the tests results on core-concrete interaction and the test conditions for the molten core cooling test were discussed. The last meeting is scheduled for February 2010 to review the results of the final integral test. It is planned to organise a workshop in late 2010 to draw the lessons learnt from this project.

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 are investigating safety issues relevant for current PWR plants as well as for new PWR design concepts. They are focusing 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 2009 during which the results of the first tests were presented and the test conditions for the following 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 Fire Propagation in Elementary, Multi-room Scenarios (PRISME) Project (from the French *Propagation d'un incendie pour des scénarios multi-locaux élémentaires)* began in 2006 and has 13 participating countries. The project's objective 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 2009. Two meetings of the project steering bodies were held in April and October. The PRISME integral test scenarios were fully reviewed and the experimental conditions were agreed by the project members. The six integral tests will be conducted from April to November 2010.

The ROSA Project

A first Rig-of-safety assessment (ROSA) Project was carried out from April 2005 to March 2009 to address issues in thermal-hydraulics analyses relevant to LWR safety using the ROSA large-scale test facility of the Japan Atomic Energy Agency (JAEA). In particular, it focused on the validation of simulation models and methods for complex phenomena that may occur during transients/accidents. The project was supported by safety organisations, research laboratories and industry in 14 countries, and provided an integral and separate-effect experimental database to validate the code predictive capability and accuracy of models. In particular, 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, and upper-head break and bottom break LOCA were addressed by the 12 tests carried out in this first phase. The project was successfully completed and the final report is under preparation.

A second phase of the project, called ROSA-2 and using the same large-scale test facility (LSTF), started in April 2009 with the support of 14 countries. The ROSA-2 programme is to last for three years and will consist of six tests. The subjects will be:

The ROSA large-scale test facility (LSTF).

JAEA

- intermediate break LOCAs (for risk-informed, break-size definition and verification of safety analysis codes);
- steam generator tube rupture (SGTR) and SGTR with steam line break (for improvement and new proposals regarding accident management and mitigation/emergency operation).

These tests will benefit from the utilisation of instruments newly acquired during the first ROSA Project.

The SCIP Project

The Studsvik Cladding Integrity Project (SCIP) started in July 2004 and completed its first five-year mandate in 2009, when several power ramps and a hot cell programme addressing the various failure mechanisms were executed. The nuclear fuel failure mechanisms studied in the project are:

- 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.

In December 2008, all members of the project steering bodies indicated their interest in continuing the project for another five-year period. SCIP-2 thus began in July 2009 with the participation of 13 countries (two more than in the first phase). The main objective of SCIP-2 is to generate the high-quality experimental data needed for improving the understanding of dominant failure mechanisms for water reactor fuels and to devise means for reducing fuel failures. The major focus will be on cladding failures caused by pellet-cladding mechanical interaction, especially stress corrosion and hydrogen-assisted fracture mechanisms, as well as on the propagation of cladding cracks. Improved understanding based on experiments and analyses is needed in order to reduce the occurrence, or the risk of occurrence, of fuel failures. This understanding is to be applicable to pellet-cladding interaction conditions that can arise during normal operation or anticipated transients, as well as during long-term fuel storage. The proposed programme is intended to complement other international projects in the fuel area. Extensive analyses and theoretical modelling of the fracture mechanisms are to accompany the experimental programme.

In addition to reviewing existing Studsvik ramp data, the project will study the following fuel failure mechanisms:

- pellet-cladding mechanical interaction (PCMI), the mechanical driving force for PCI and hydrogen-induced failures;
- pellet-cladding interaction (PCI), notably when cladding fails due to stress corrosion cracking;
- hydrogen-induced failures: in particular as regards zirconium alloys, classical hydride embrittlement (HE) and delayed hydrogen cracking (DHC).

Two meetings of the project steering bodies took place in 2009. In addition, a workshop was held on hydrogen-induced failures.

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 detail 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 (French *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 2009 and the results of two new tests were presented and discussed, enabling a better specification of the test configurations to come. In parallel, analytical activities were undertaken to prepare and to assess these tests.

KAERI

Testing corium melt and wter coolant interaction at the TROI facility.K

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 2009 and were devoted to presenting the new test results and to discussing the parameters and boundary conditions to be chosen for the remaining test. The project is planned to be completed during 2010.

The SFP Project

The Sandia Fuel Project (SFP) is a new NEA project supported by 13 member countries, which began in 2009. The objective of the project is to perform a highly detailed thermal-hydraulic characterisation of full-length, commercial fuel assembly mock-ups to provide data for the direct validation of severe accident codes. Code predictions based on previous results indicate that fuel assemblies can ignite and radially propagate in a complete loss-of-coolant accident. Hence, there is a need for qualified data obtained under representative fuel configurations. The experiments should focus on thermal-hydraulic and ignition phenomena in PWR 17x17 assemblies and supplement earlier results obtained for BWR assemblies. Code validations based on both the PWR and BWR experimental results will considerably enhance the code applicability to other fuel assembly designs and configurations.

The project is scheduled to last three years and to be conducted in two phases. Phase 1 will focus on axial heating and burn propagation. Phase 2 will address radial heating and burn propagation, and will include effects of fuel rod ballooning.

The first meeting of the project steering bodies was held in 2009 during which the programme of work for 2009 and 2010 was approved.

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 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 (PARs), 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 2009, two meetings of the project steering bodies were held to discuss the results of the hydrogen recombiner tests, the PAR poisoning tests and the aerosol wash-down scoping test. Hydrogen distribution tests performed in 2007 are used to support the ISP-49 benchmark exercise conducted in co-operation with the CSNI Working Group on Analysis and Management of Accidents (WGAMA). A follow-up proposal has been proposed with a three-year programme addressing dust transport in advanced gas-cooled reactors, hydrogen mitigation and iodine or aerosol behaviour in specific containment conditions.

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. Countries began submitting data in 2006, however the total number of event records in the database is still very low. One meeting of the COMPSIS steering body was held in 2009.

The FIRE Project

The Fire Incidents Records Exchange (FIRE) Project started in 2002. A third phase of the project is to start in January 2010 for an additional four years. 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 365 records, is provided to participants every year. Two meetings of the project steering body were held during 2009. A report on the collection and analysis of fire events over the period 2002-2008 was produced.

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. Additional 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 and reports are planned for 2010. Two project meetings were held in 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. One Project Review Group meeting was held in 2009.

A report was produced during 2009 describing the status of the OPDE database after six years of operation (from May 2002 to May 2008), and giving some insights based on the approximately 3 600 piping failure events in the database.

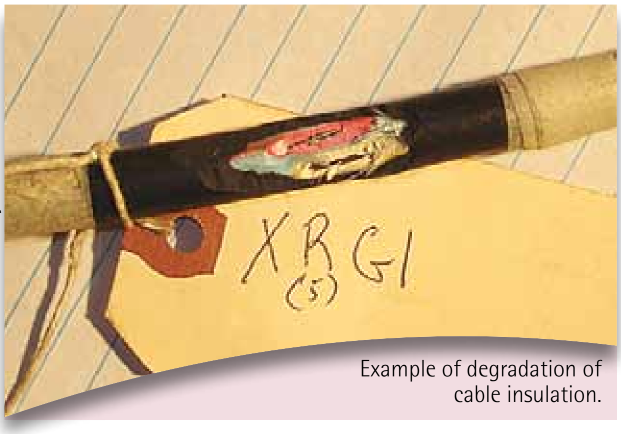
The SCAP Project

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 is currently focusing on the continuing population of the database as well as the assessment of the data. The assessment report will be issued at the end of the project and will provide the technical basis for commendable practices in support of regulatory activities in the fields of SCC and cable insulation.

The workshop on Commendable Practices for the Safe, Long-term Operation of Nuclear Reactors – OECD/NEA Stress Corrosion Cracking and Cable Ageing Project (SCAP) will take place on 25-26 May 2010 in Tokyo, Japan.



OECD Halden Reactor Project

Example of degradation of cable insulation.

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 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 51 projects under active decommissioning (31 reactors and 20 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 decontamination and dismantling of concrete structures and on remote dismantling techniques.

The Sorption-3 Project

Radionuclide sorption is one of the most important processes with regard to the prevention or retardation of radionuclide migration from a geological repository 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 the project. A guideline document will be produced on thermodynamic sorption model development and the use of such models in building a safety case.

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 project's current mandate runs from 2008 to 2012. Sixteen organisations from 14 countries are participating.

The review on chemical thermodynamic data for inorganic compounds and complexes of thorium (Volume 11) was published in early 2009. Completion and publication of the reviews of chemical thermodynamic data for inorganic compounds and complexes of iron (Fe) and tin (Sn) are scheduled for 2010. A study of inorganic species and compounds of molybdenum (Mo) and a review of ancillary data began in 2009 and will continue for four years. A complementary study of inorganic species and compounds of iron (Fe) will start in 2010.

RADIOLOGICAL PROTECTION

The ISOE System

Since its creation in 1992, the Information System on Occupational Exposure (ISOE), jointly sponsored by the IAEA, has been facilitating the exchange of data, analysis, lessons and experience in occupational radiological protection (RP) at nuclear power plants worldwide. The ISOE programme maintains the world's largest occupational exposure database and a network of utility and regulatory authority RP experts. As of December 2009, membership included 61 participating utilities in 27 countries and the regulatory authorities of 23 countries.

Four supporting ISOE Technical Centres (Europe, North America, Asia and the 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 471 reactor units in 29 countries (395 operating units and 76 under decommissioning), thus covering about 90% of the world's operating commercial power reactors. The ISOE database, publications and annual symposia along with the ISOE Network website facilitate the exchange amongst participants of operational experience and lessons learnt in the optimisation of occupational radiological protection.

In 2009, 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 also continued in 2009 through the International ALARA Symposium held in Vienna, and the regional ALARA symposia held in the United States and Japan. The ISOE Network web-based information portal (www.isoe-network.net) forged ahead as a "one-stop" website for ISOE information and experience exchange. In 2009, the website underwent a significant upgrade and the new web-based input modules for occupational exposure data collection were completed and approved for implementation. The ISOE ad hoc expert group for the revision of the International Basic Safety Standards (BSS) continued to provide, through the CRPPH, input into the BSS revision process with respect to good practice in occupational exposure.