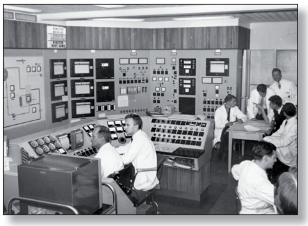
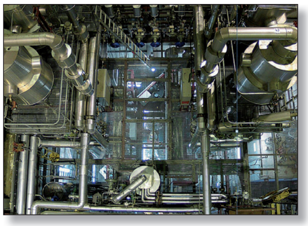
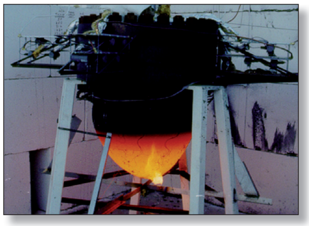
Nuclear Safety
2012

Main Benefits from 30 Years of Joint Projects in Nuclear Safety

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Main Benefits from 30 Years of Joint Projects in Nuclear Safety

NEA No. 7073

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 30 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Cover photos: Halden reactor control room in the 1960s (OECD/NEA Halden Project, with the kind permission of the Institute for Energy Technology, Norway); lower head vessel failure test (OECD/NEA OLHF Project, with the kind permission of Sandia National Laboratory, United States); PKL top view (OECD/NEA PKL Project, with the kind permission of AREVA NP, Germany).

Foreword

The objective of the OECD Nuclear Energy Agency (NEA) is to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal basis required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. The NEA standing technical committees (STCs) are actively engaged in the generation of knowledge (workshops, state-of-the-art reports, international standard problems, joint research projects, etc.) and the members of the committees and their working groups are excellent sources of tacit knowledge and communities of practice. The NEA committees have generated significant technical and scientific information that is of value to the regulators and the developers of nuclear technology.

The NEA platform provides a unique capability for conducting joint international research projects. These projects, established under the auspices of the NEA standing technical committees, bring together the world's leading experts who contribute to maintaining and improving expertise and tools in participating countries, to enhancing technical exchange among specialists, and to promoting consensus building on approaches to resolve complex safety issues. Hence, these programmes also make a very significant contribution to knowledge management.

The joint projects, primarily addressing the areas of nuclear safety and radioactive waste management, enable interested countries, on a cost-sharing basis, to pursue research or the sharing of data with respect to particular areas or issues. Since 1958, when the first project at the Halden reactor was established, more than 30 joint projects have been conducted with wide participation of many NEA member and non-member countries.

Overall, the activities of the NEA Committee on the Safety of Nuclear Installations (CSNI) and those of the joint projects conducted under NEA auspices are responding to the challenges of power uprates, higher burn-up, new fuel element designs, new cladding materials and the development of models to analyse accidents, including severe accidents. Thorough understanding of phenomena and failure mechanisms associated with accidents as well as common assessments of experimental data and computer code models strengthen the technical bases for safety decisions. Consistent with NEA practice, all new programmes also carefully consider the lessons learnt from recent operating experience, including reported events and accidents such as the recent one at the Fukushima Daiichi nuclear power plant.

The purpose of this report is to describe the achievements of the OECD/NEA joint projects on nuclear safety research that have been carried out over the past three decades, with a particular focus on thermal hydraulics, fuel behaviour and severe accidents. It shows that the resolution of specific safety issues in these areas has greatly benefited from the joint projects' activities and results. It also highlights the added value of international co-operation for maintaining unique experimental infrastructure, preserving skills and generating new knowledge.

The projects described in this report are organised as individual experimental programmes involving major facilities and thus contributing to the maintenance of indispensable safety research infrastructure and the expertise of the operating teams. Other joint projects, namely five database projects, are not covered in the main section of this report but are briefly described in Appendix 1.

Acknowledgements

The NEA is very thankful to Mr. Ashok Thadani and to Mr. Victor Teschendorff for having prepared this report. Mr. Thadani was formerly Director of the Office of Nuclear Safety Research at the US Nuclear Regulatory Commission as well as CSNI Chairman for four years. Mr. Teschendorff was formerly in charge of supervising nuclear safety research for the German Technical Safety Organisation GRS, and Chairman of the CSNI Programme Review Group for five years. The NEA wishes to acknowledge Mr. Carlo Vitanza from the Halden establishment in Norway and Mr. Miroslav Hrehor from the Nuclear Research Institute (NRI) in the Czech Republic, who supported NEA joint projects for a number of years and provided valuable advice in reviewing this report.

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1. Introduction

The Nuclear Energy Agency (NEA) is a semi-autonomous body within the Organisation for Economic Co-operation and Development (OECD). The objective of the Agency is to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal basis required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. The major NEA activities are conducted by seven standing technical committees (STCs) as well as an executive group managing the NEA Data Bank. The NEA committees are actively engaged in the generation of knowledge (workshops, state-of-the-art reports, international standard problems, joint research projects, etc.) and the members of the committees and their working groups are excellent sources of tacit knowledge and communities of practice. The NEA committees have generated significant technical and scientific information that is of value to the regulators and the developers of nuclear technology.

One of the NEA's major achievements is the knowledge generated by international joint research projects (JRPs) carried out under the auspices, and with the support, of the NEA [1]. Such projects, primarily in the areas of nuclear safety and waste management, enable interested countries, on a cost-sharing basis, to pursue research or the sharing of data with respect to particular areas or issues.

The purpose of this report is to describe the achievements of the OECD/NEA joint projects on safety research that were carried out during the past three decades with a specific focus on thermal-hydraulics, fuel behaviour and severe accidents. It shows that the resolution of specific safety issues in these areas has greatly profited from JRPs. It highlights the benefit of working together for maintaining unique experimental infrastructure, preserving skills and generating new knowledge. Those who are interested in a particular project will find in Appendix 1 detailed information about the objectives, participating countries, test facilities, experimental programmes and selected results, along with budget figures and references.

The NEA Strategic Plan for 2011-2016 [2] guides the NEA as it seeks to meet the evolving needs of member countries in the field of nuclear energy. With respect to the nuclear safety and regulation sector, the goal is to assist member countries in their efforts to ensure high standards of safety in the use of nuclear energy, by supporting the development of effective and efficient regulation and oversight of nuclear installations and activities, and by helping to maintain and advance the scientific and technological knowledge base.

Over the last 50 years significant knowledge has been accumulated in nuclear technology. However, over the past decade or more there has been an erosion of scientific resources in many countries due to reductions in budgets, ageing and retirement of many outstanding scientists, engineers and managers, loss of experimental facilities and reductions in educational programmes in nuclear engineering. The NEA has fully recognised this concern and has published many reports, technical opinion papers and collective statements on this topic (e.g. *Nuclear Education and Training: Cause for Concern?* [3], *Future Nuclear Regulatory Challenges* [4], *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk* [5, 6], etc.). With the industry push to reduce conservatisms in decisions, to extend the operational period of nuclear power plants and increased interest in building new nuclear power plants, it becomes imperative that additional mechanisms be developed to make available both the tacit and the explicit knowledge to the new generation of engineers and scientists for safety and technology decisions.

This matter has become even more critical as more countries consider the utilisation of nuclear energy. In addition, the historic earthquake followed by a historic tsunami on 11 March 2011 in Japan raises many issues relating to the treatment of rare events, combinations of events, human factors, severe accidents, severe accident management, training of personnel and potentially many other issues. It is important to obtain more complete information about the events at Fukushima Daiichi and to fully understand the safety implications before initiating any significant new actions. It is also possible that some of the earlier conclusions from past research projects may have to be revisited after the lessons from the events are fully understood.

The NEA and its committees' operating plans reflect well the need for learning from operating experience, preserving important knowledge as well as generating new knowledge to deal with safety/technology issues. Technical information in the form of scientific research, engineering analyses, regulatory reviews, operational and maintenance records, and other documents and data relating to nuclear power installations has been accumulated. Further, as noted above, considerable knowledge resides with the people involved in the development and application of the scientific/engineering information.

The NEA committees support the development of models, the issuance of state-of-the-art reports and the assessment of operational experience for safety and technology information. The NEA committees also provide an excellent forum for building consensus on complex subjects and issuing reports reflecting the collective opinion of the member countries. In cases where there are gaps in knowledge in important areas, action plans are set up by the committees to further investigate any complex matters.

The Committee on the Safety of Nuclear Installations (CSNI) organised the Seminar on Transfer of Competence, Knowledge and Experience Gained through CSNI Activities in the Field of Thermal-Hydraulics (THICKET), which is an excellent example of efforts to transfer knowledge to the younger generation of scientists and engineers. Initiated in 2004, the seminar is conducted every four years. Such a forum also allows many countries to appreciate the capabilities provided by the NEA. The NEA publishes a large number of reports that embody the Agency's activities for distribution to various interested parties.

The NEA Data Bank deserves special recognition for acting as a service centre to member countries by providing state-of-the-art computer codes and nuclear data to scientists performing calculations and simulations in different nuclear applications.

Knowledge preservation has been an issue of concern in the NEA committees. The Committee on Nuclear Regulatory Activities (CNRA) reports on future regulatory challenges [4, 7, 8, 9, 10] and the "Workshop on Assuring Nuclear Safety Competence into the 21st Century" [11] highlight the regulators' concern to preserve important knowledge and to educate young scientists and engineers. The CNRA report entitled "Committee on Nuclear Regulatory Activities: A Historical Perspective (1989-2004)" [12] is an excellent knowledge management effort that consolidates and references documents reflecting CNRA workshops and conferences from 1989 to 2004 on important safety/regulatory topics.

The CSNI report entitled *Nuclear Safety Research in OECD Countries – Major Facilities and Programmes at Risk* [5] is an excellent example of the need to preserve important experimental facilities which, if lost, would be extremely difficult to construct in future to address potentially important new safety issues. The CSNI has provided an excellent forum for conducting experiments and for model development on important safety topics. The cost-shared joint research projects have brought together the world's best experts and enabled the use of experimental facilities in the member countries to generate and share information and data. This report summarises the main results achieved and reflects the importance of joint projects for international co-operation on safety research as well as the benefit for the participating countries in terms of efficient use of resources to support enhanced understanding of safety topics, improved analytical tools and knowledge management.

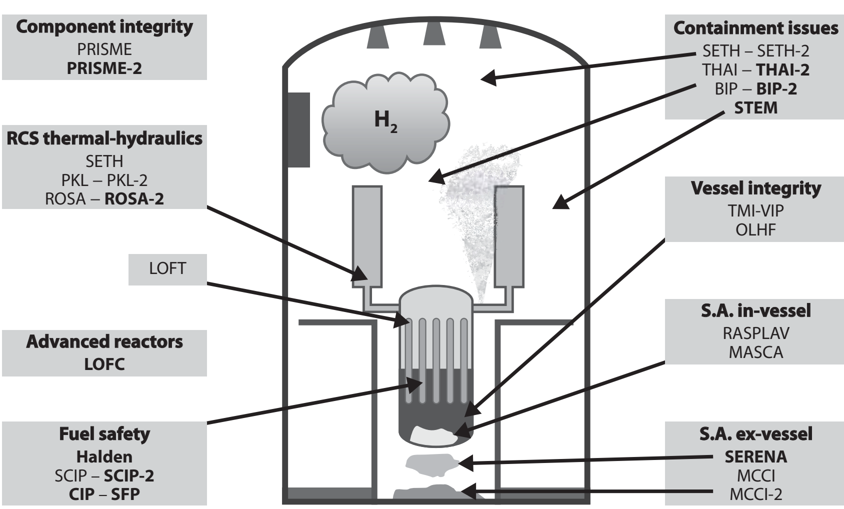
After reviewing the NEA historical background regarding the establishment of joint co-operative projects and co-operation in safety and licencing support, this report reviews the main technical issues and the corresponding CSNI activities and joint projects outcome grouped in three main domains of safety research: thermal-hydraulics, fuel behaviour and severe accidents. A final chapter "working together" describes the process for establishing a joint project and its role in building consensus.

2. NEA historical background and plans in safety and licensing support

The first outcome of the NEA was the establishment of three joint undertakings, requiring heavy funding, through intergovernmental co-operation [1]. The Halden Reactor Project was created in June 1958 by an agreement with the then Norwegian Atomic Energy Institute (now the Institute for Energy Technology) for the joint operation of an experimental boiling heavy water reactor built in Halden, Norway. This 50-year-old project evolved gradually under successive 3-year programmes to perform research and development programmes in the areas of nuclear safety, including fuel reliability, integrity of reactor internals, online computer control and monitoring and human factors. The OECD Dragon Project created in April 1959 allowed construction and operation of a 20-MW(th) experimental high temperature gas-cooled reactor in the United Kingdom. The work concentrated on testing fuels and fuel elements' behaviour under high temperature until its end in 1976. The European Company for the Chemical Processing of Irradiated Fuels: the Eurochemic Company, established in July 1959 in Belgium built and operated a plant and a laboratory for the reprocessing of spent nuclear fuel until it stopped operation in 1975.

While in the 1960s international co-operation was intended for those large research programmes later covered by national industrial programmes, it appeared in turn in the early 1980s that national safety research programmes would be difficult to be continued without external support. In this respect, upon a proposal from the United States to the NEA Steering Committee, the first NEA joint project fully dedicated to the nuclear safety research – the Loss-of-Fluid Test (LOFT) Project – was launched in spring 1983 with ten member countries. Since then until now more than 30 safety joint projects or projects follow-up were set up, with an average duration of 4 years and an average participation of a dozen of countries.

Figure 1: Overview of experimental joint projects



In 1992, a Senior Group of Experts on Safety Research (SESAR) was set up by CSNI to review research being carried out and to identify future requirements and priorities [13]. A follow-on report took their arguments further [14]. In these reports concerns were raised about the member countries' ability to maintain an adequate level of safety research. In 1995, CSNI set up another SESAR focusing on research capabilities and facilities [15]. One of the recommendations was "that CSNI take a proactive role in organising and implementing co-operative projects…". As a consequence, CSNI charged a new SESAR with the specific action to address the imminent loss of facilities and programmes at risk (SESAR/FAP). The group was asked to identify facilities of potential interest for international collaboration, to make specific recommendations regarding facilities, research programmes and joint projects. Their report was issued in 2001 [5], but the recommendations were already available in 1999. All projects initiated after 2000 were inspired by the FAP recommendations. The impact of the report was significant, both on the number and on the contents of the joint projects. Figure 2 shows the increase in the number of projects following SESAR/FAP. Other qualitative features of joint projects are presented in Appendix 2.

Figure 2: Number of ongoing projects in the time period 1982-2011

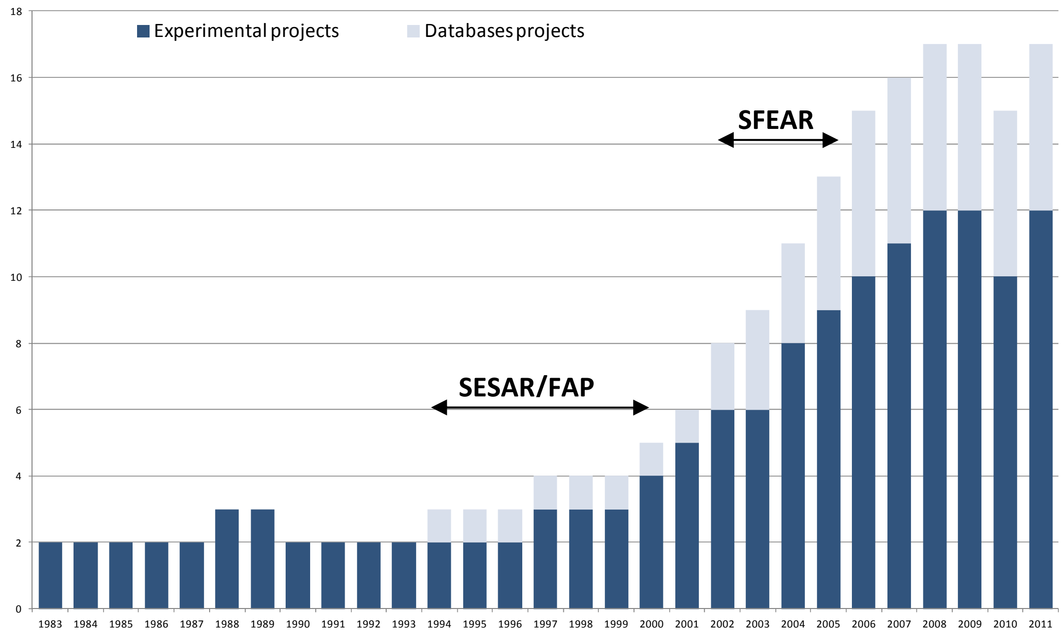


Table 1 shows the links between the recommendations and the new projects. There were very few recommendations by SESAR/FAP that have proven impossible to fulfil, notably in the seismic area.

Although five key facilities were preserved in the time period 2000-2006, other important facilities were shut down. Recognising the need to have key experimental facilities available for all types of light water reactors (LWRs) and for emerging safety questions for advanced LWRs, CSNI started a follow-up activity entitled Support Facilities for Existing and Advanced Reactors (SESAR/SFEAR). The SFEAR Group started in 2004 by identifying safety issues for all thematic areas unique to the nuclear industry; the issues were ranked for their safety relevance, the state of knowledge was evaluated, and finally, the ability of specific facilities to address the issue was assessed. The final report [16] issued in 2007 was very selective in its short-term recommendations by proposing only four facilities for joint projects. The long-term recommendations contain a list of facilities to be monitored.

Table 1: Links between SESAR/FAP recommendations and NEA joint projects started after 1999

SESAR/FAP recommendation	Resulting CSNI action
1. Maintain the PANDA, PKL and SPES facilities in the thermal-hydraulic area (facilities in near-term danger of closure)	Initiated the SETH programme utilising the PANDA and PKL facilities (no host country support for SPES)
2. Monitor and maintain key thermal-hydraulic facilities in the long term (APEX, RD-14M, ROSA/LSTF)	Facility status monitored. Initiated programme utilising the ROSA facility when it was in danger of being shutdown
3. Maintain the RASPLAV and MACE facilities in the severe accident area (these facilities were in near term danger of closure)	Initiated the MASCA programme as follow-on to RASPLAV to maintain facilities. Initiated the MCCI programme utilising the MACE facility
4. Develop centre of excellence on fuel-coolant interaction (FCI) in consideration of potential loss of the FARO and KROTOS facilities	SERENA programme FARO shut down. KROTOS kept in standby. SERENA has recommended an experimental programme in KROTOS and TROI *(became SERENA Project)*
5. Develop centre of excellence (COE) on iodine chemistry and fp behaviour	Proposal for a project currently under evaluation (AECL proposal) *(became BIP Project)*
6. Define an experiment co-operative programme based upon fire PSA needs	Start of the FIRE database and of the PRISME experimental programme
7. Experts to define fuel experiment needs, promote industry involvement	Cabri and SCIP project initiated, NSRR in Cabri, industry strongly involved

The many decades of experience from operation of nuclear facilities and results from research have led to continued improvement in safety of nuclear power plants. NEA has been extensively engaged from the early days of the development of this technology.

The NEA standing committees that focus on safety research and regulation are the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA).

The mission of the CNRA and the CSNI is to assist member countries in maintaining and further developing the knowledge, competence and infrastructure needed to support the safe operation of NPPs and fuel cycle facilities throughout their life cycle, as well as their efficient and effective regulation based upon sound technical information, shared experience and up-to-date methods.

The joint CSNI/CNRA Strategic Plan for years 2011-2016 [17] recognises the challenges that the regulators and safety research organisations will face in the next years as reflected in the statement below.

> *"This joint strategic plan for the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) takes into account the evolving status of the nuclear industry worldwide and the main challenges that will face the regulatory bodies, technical safety organisations, and the scientific community over the next six years (2011-2016). The Committees provide a unique forum for established regulatory bodies and technical safety organizations to exchange knowledge and information, and to collaborate on scientific research. The combined success achieved through the Committees enables the individual countries to be more effective and efficient in their programs".*

An important part of the mandate of the committees is to consider operating experience and assess any lessons that can be learnt so appropriate safety decisions can be taken by the national bodies. Thus, the Committees will need to carefully consider the lessons to be learnt from the significant impact of the 11 March 2011 earthquake and the tsunami at the Fukushima Daiichi nuclear power plants.

The committees recognise the evolving status of the nuclear industry worldwide and the challenges that regulators and technical safety organisations will need to address over the next five years. The following five challenges from this joint strategic plan are likely to determine the focus of CNRA and CSNI activities:

1. Adequate nuclear skills and infrastructure

There is a significant shift internationally in nuclear workforce demographics being experienced, as many staff experienced in all areas that affect the safety of installations (e.g. construction, operation, maintenance, engineering, technical safety, research and regulation) approach retirement. At the same time, there is an increased demand for these skills. The CSNI and the CNRA programme of work will focus on new means to sustain safety excellence in operating and new nuclear installations.

2. Effectiveness and efficiency of activities related to safety

To sustain high levels of nuclear safety requires continued attention to learning from experience and from others, and is especially important in a climate of rapid change and expansion in the use of nuclear energy. Additionally, regulatory bodies and their technical safety organisations need to ensure that new regulations, guidance, etc., and the revisions of existing regulations, have sound technical bases and lead to enhancements in safety. Through continued learning from each other and harmonisation of implementation strategies for key regulatory processes, improvements in regulatory effectiveness and efficiency may be achieved. Improvements in effectiveness can be gained through learning from others what has worked or not worked in response to particular issues, pooling of knowledge and experience, selection of relevant research efforts, etc; and in efficiency, through the use of collaborative efforts and cost sharing.

3. Safe operation of current nuclear installations

The safety of operating installations depends on a number of factors such as plant location and configuration, ageing of materials and components, safety culture and human performance in maintenance, engineering and operation. Regulatory oversight through the use of inspection and performance assessment ensures that safety margins necessary for adequate protection are maintained. As more installations operate with extended licenses and at increased power levels, it is critical to understand the safety implications of changes in plant configuration, operational modes, and the maintenance of ageing components. Safety can also be enhanced, through the use of operating experience, analysis, research and available tools such as probabilistic safety assessments (PSA) to gain insights that are not available from purely deterministic analyses. The committees' programme of work focuses on maintaining safety margins in light of advances in scientific and technical knowledge to ensure the implications of ageing, changes in plant configuration, and requested operating domains are understood and well managed and, on improving the use of risk-informed regulatory strategies, updated with operating experience and safety research results, to evaluate, measure, and enhance the safety performance of nuclear installations.

4. Safety in new nuclear installations

If new technology or analytical methodologies are utilised in a design, the regulatory body must ensure sufficient technical basis is demonstrated. Additionally, an assessment of current regulations and standards should be included in the review of new reactor designs technology. International experience and lessons learnt from safety reviews and construction of new installations should be considered at the national level for safety reviews. Recent new design and construction experience has already demonstrated new challenges with a global workforce and suppliers. International collaborative efforts, such as the Multinational Design Evaluation Programme (MDEP), can yield improvements in regulatory practices and increases in the knowledge and understanding of new technology. The committees support, to the extent practical, the utilisation of new or improved analytical tools in safety reviews and identify best safety practices through international collaborative efforts.

5. *Safety in advanced reactor designs*

New approaches, new concepts and new technology often present new issues for safety. The development and validation of new analytical tools and research is necessary to support the identification and resolution of new or unique safety issues based on the technology of the advanced designs. Standards and safety practices for advanced designs have the greatest potential for international harmonisation and should be pursued to the extent practical. Likewise, international collaborative projects and cost-sharing have significant potential for mutual gains. The committees, in particular, CSNI, provide a forum to discuss advanced design issues and encourage the balanced and gradual incorporation of items relevant to advanced reactor safety in the working groups' programme of work.

This report is based on the review of major joint projects in areas needing additional knowledge to better understand the phenomena involved and improve the analytical tools. In the following, the projects are described under three areas: thermal-hydraulics, fuel and severe accidents. The few projects not directly falling into one of the three areas were attributed to the area most pertinent to the phenomena investigated. Although it is recognised that the three areas are interconnected, the projects are described under these areas to emphasise the role of specific disciplines.

3. Thermal-hydraulics

3.1 Background

Safety concerns about commercial nuclear power reactors arose in the early 1970s. After dealing with reactivity control and nuclear stability, concerns focused on the question how decay heat could be removed from a water-cooled nuclear core after a loss of coolant caused by failure of the vessel or the piping. It became clear that all plants should be equipped with an emergency core cooling system (ECCS) to cope with a loss-of-coolant accident (LOCA). While the failure of the vessel was not judged to be credible, regulation prescribed that even in case of a double ended guillotine break (DEGB) of a main coolant loop, ECCS performance must ensure that the core remains amenable to cooling and limits were set to the maximum fuel clad temperature, the depth of clad oxidation and the total amount of Zirconium oxidised during the accident. By defining the large break (LB)-LOCA as a design basis accident for ECCS performance, thermal-hydraulics became the key subject of safety research and triggered the largest nuclear safety research programmes in history. Computer codes were developed to simulate the hypothetical LB-LOCA, mostly applying conservative assumptions for modelling uncertainties and plant boundary conditions, in order to provide sufficient margin to safety limits.

The Three Mile Island-2 (TMI-2) accident occurred in March 1979 and early risk studies showed that small breaks may be more important in terms of risk than the DEGB. Attention turned to small breaks (SB-LOCA), and initiating events of relatively higher probability were studied. Whereas LB-LOCA was dominated by critical discharge rate, steam binding during the reflood phase, counter-current flow limitation and post-dryout heat transfer, a variety of complex phenomena became of influence: natural circulation in single and two-phase flow, stratification in horizontal pipes, heat transfer in the steam generators under depressed water level and under the influence of nitrogen, just to mention a few. This posed a new challenge both on computer codes and the experiments for validation. The amount of required research called for sharing the burden among an international community. At the same time, regulators were looking for international consensus on the technical basis for closing classical issues like ECCS performance under all LOCA conditions. The OECD-LOFT Project and the international 2D/3D Project [18] are the best known examples of the co-operations that followed in the thermal-hydraulic area.

On the analytical side, bi-lateral and multi-lateral co-operation agreements were established for sharing computer code development and validation. OECD/NEA became a most efficient forum for this co-operation.

With many reactors in operation and growing operational experience along with more consideration of hypothetical core melt accidents, a strong demand arose for best-estimate, i.e. as realistic as possible, calculations. Optimisation of core designs and of modes of plant exploitation were more reasons to ask for the quantification of margins by using best-estimate calculations. Power uprates realised in many existing plants and planned in many others raise multiple safety questions, some of them addressing the fuel (see Chapter 4). Thermal-hydraulic codes play a key role in evaluating the remaining safety margins.

Computer codes and the validation basis had reached a level of realism and sophistication in modelling that allowed this transition from conservative to realistic modelling. The adoption of best-estimate approach in a regulatory context entailed, of course, the need for quantifying uncertainties of calculations. This additional demand gave rise to the development of several uncertainty quantification methods. The assessment of their performance and practicability was done in international co-operation from the beginning.

With fast growing computing power and development of fluid-dynamic simulation methods outside the nuclear industry, a new class of codes appeared, the computational fluid dynamic codes (CFD). These codes are characterised by resolving the computational domain by a fine 3-dimensional grid and by including turbulence modelling. These new computations, although still extremely time consuming, are making their way in the nuclear industry. They allow addressing safety relevant phenomena like thermal stratification and mixing, mixing of boric acid in complex geometries, or sub-channel flows in a fuel element in much greater detail than system codes or classical component codes [19]. Using CFD codes for safety assessment is still a challenge because the practices of validation and application from thermal-hydraulic system codes cannot be readily applied. International co-operation will be the key to a wider acceptance of CFD calculations for safety demonstrations.

3.2 Thermal-hydraulic issues

Safety research in the field of thermal-hydraulics has been for a large part dealing with the following topics.

Loss-of-coolant accidents (LOCA)

LOCA was considered the dominating design basis accident. A tremendous amount of research was devoted to the demonstration that the ECCS and the containment are properly designed to cope with the full spectrum of postulated LOCAs. A large variety of fluid dynamics and heat transfer phenomena is involved in the course of a LOCA. The development of advanced computer codes and their systematic validation have established the technical basis for closing most of the LOCA issues.

Strainer clogging

Performance of the ECCS in the recirculation mode is dependent upon the ability of the containment sump strainers to remove debris, without plugging to the point of cutting off recirculation flow. This includes consideration of solid debris as well as chemical effects which can cause gelatinous material.

Boron dilution

Events where boiling and condensation can occur in the primary cooling system have the potential to form areas where non-borated water can collect. If primary coolant pumps are started, or the restart of natural circulation occurs in combination with other unfavourable circumstances, it was postulated that there was the potential for a slug of non-borated water to enter the core and cause a criticality or even return to power. This issue involves understanding both the thermal-hydraulic response as well as the reactor kinetics, with emphasis on the 3D aspects of boron transport and mixing in the reactor pressure vessel (RPV). Although resolved in some countries (e.g. the United States) through implementation of emergency procedures this issue is still subject of further investigation in others.

Pressurised thermal shock (PTS)

PTS refers to a condition that challenges the integrity of the RPV. The integrity of this primary component in a nuclear power plant is vital which explains why the PTS events receive much attention in safety analyses, specifically in the context of long-term operation and license renewal of older plants. Three conditions are typically considered as prerequisites that lead to a PTS safety concern. These include neutron embrittlement of the RPV, some type of existing flaw, and finally a rapid drop in temperature in the system (overcooling event). PTS assessment has to rely on boundary conditions on the fluid side thus challenging thermal-hydraulic calculations.

Power oscillations

After power oscillations were observed in some boiling water reactors (BWRs), they have been extensively studied using 3D coupled thermal-hydraulic/neutronic codes. The validation of these code systems was mostly relying on reactor transients. There is a need to better understand the impact of power uprates on the potential for oscillations including anticipated transients without scram (ATWS) conditions. The impact of higher power levels on BWR power stability needs to be understood. Also, for future BWR designs, this issue may need further research to confirm predicted behaviour.

Shutdown states

Most work to date in the thermal-hydraulic area has been associated with analysing accidents initiated from full power conditions. However, during shutdown conditions, plant configurations, temperature, pressure, system operability and coolant inventories may be substantially different than they are at full power. Accordingly, accidents that occur will likely have a much different progression, timing and potential consequences than those at full power. For example, loss of residual heat removal (RHR) under shutdown conditions, i.e. during mid-loop operation, has occurred several times worldwide and still plays an important role in risk studies for PWRs. When the reactor coolant system is open at the time of RHR failure, the decay heat cannot be transferred to the steam generators and other measures become necessary in order to prevent or compensate for the loss of inventory that finally would lead to core damage. To ensure analytical capabilities are adequate, experimental confirmation of system behaviour will likely be required.

Passive systems

Some advanced designs, namely Generation III reactors, take credit of passive systems or features for emergency core cooling, decay heat removal and/or containment cooling. Although some experimental work has been done on passive systems in the past, the applicability of the results to new designs will be dependent upon the specific design and proposed operating conditions. Accordingly, the capability to experimentally investigate passive safety system performance should be maintained and used to validate analytical tools and confirm system performance, including reliability and performance under a range of conditions that could result from ageing, environmental conditions, etc.

3.3 CSNI thermal-hydraulic activities

Thermal-hydraulics has been a major discipline and field of action of CSNI since its beginning. The former Principal Working Group No. 2 (PWG-2) has successfully conducted major activities. Today, thermal-hydraulic safety topics for cooling systems and containment are covered by the mandate of the Working Group on Analysis and Management of Accidents (WGAMA). The CSNI Operating Plan [20] designates WGAMA as the leading group for the following technical goals:

- To assess advanced methods and tools used for event/accident analysis; to assess methods and tools for severe accidents and source term analysis; to quantify corresponding uncertainties; and to further improve accident management.

- To develop approaches and methods to quantify safety margins.
- To identify and resolve safety issues specific to new designs.

The activities of these working groups and specialised task groups have worked in many ways to enhance our understanding of the thermal-hydraulic phenomena and processes underlying accidents and transients in water-cooled reactors, and how to model and assess them. They have thereby contributed to the resolution of the above mentioned safety issues. The common understanding was documented in six state-of-the-art reports.

Two of the highly appreciated products from these activities deserve special mentioning: the code validation matrices and the international standard problems (ISPs). They were first created in the thermal-hydraulic area and then adopted in other areas, namely severe accidents and containment.

Code validation matrices were constructed with the aim to provide a firm database for code validation. Upon identifying the phenomena dominating different types of accidents and transients, test facilities suitable for reproducing these phenomena were selected. This was a successful attempt to collect in a systematic way the best set of available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties. Validation matrices were established for integral experiments and separate effects tests [21, 22]. They were supplemented later for VVER reactor phenomena [23]. Having reached consensus that these matrices constitute a commonly recognised set of data for code validation is a major achievement.

In order to increase confidence in the validity and accuracy of simulation tools and the competence of the users, CSNI has sponsored a considerable number of *international standard problems (ISPs)* [24]. These are comparative exercises in which predictions or recalculations of an experiment are compared among each other and above all with the results of a carefully specified experiment. Fifty ISPs were conducted from 1974 until now. Thermal-hydraulic tests were dominating the first decade, four of them based on tests in the LOFT facility. The comparison reports on each ISP yielded valuable insight in capabilities and limits of code predictions and brought out the user effect.

3.4 OECD/NEA joint projects addressing thermal-hydraulic issues

Experimental facilities and programmes have played an important role in safety research from the beginning and have contributed substantially to the resolution of various safety issues.

OECD/NEA joint projects were conducted making use of the major thermal-hydraulic experimental facilities in OECD countries [15]. These facilities were originally built as part of a national research programme to solve a specific safety relevant issue or a cluster of related safety relevant topics.

The following integral loop test facilities simulate the primary coolant system of a PWR in different scales:

- LOFT (Loss-of-Fluid Test, United States): model of a 1 000-MW PWR, volume and power scale about 1:50. The only integral test facility with a nuclear core (1/2 length), one intact loop and one half loop with the break.
- PKL (*Primärkreislauf*, Germany): model of a 1 300-MW PWR, volume and power scale 1:145, full height, four loops.
- LSTF (Large Scale Test Facility, Japan): model of a 1 100-MW PWR, volume scale 1:48, full height, two loops, full pressure.

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- PSB (Russia): model of a VVER-1000 reactor, volume and power scale 1:300, full height, four loops.

For simulating safety relevant containment functions the following facilities served for NEA projects:

- PANDA (Switzerland): design originally based on a 670-MW SBWR (GE); volume scale 1:25, full height; 6 large interconnected pressure vessels.
- MISTRA (France): insulated steel containment of 97.4 m^3. Two cells, a flat cap and a bottom fixed together with twin flanges, inner compartment with an annular ring.
- THAI (Germany): containment test facility; insulated stainless steel vessel of 60 m^3; 1.4 MPa; internal structure cylindrical with four block flanges; supply of steam, gases, aerosols and iodine.
- EREC BC V-213 (Russia): bubbler condenser test facility at Electrogorsk Research Centre for VVER-440, Type 213, reactors, scaling 1:100 for volumes and mass- and energy reservoirs.
- DIVA (France): contained multi-room configuration: three 120 m^3 rooms, one 150 m^3 corridor, one 170 m^3 room on the first floor, doors with controlled leakage, ventilation network.

The following paragraphs describe briefly the thermal-hydraulic projects and how they have contributed to the understanding of phenomena, to the development of analytical tools and to the resolution of safety issues. A summary and other pertinent information including references for each project is provided in Appendix 1.

Loss-of-Fluid Test (LOFT) Project (1983-1990)

The OECD/NEA LOFT Project was designed to use the LOFT nuclear test facility that had completed a successful national programme of nuclear and non-nuclear tests, addressing several configurations of LB- and SB-LOCAs. The unique opportunity to validate thermal-hydraulic system codes on experiments in a nuclear test facility was seized by the CSNI, resulting in four international standard problem exercises. The experimental programme under the OECD/NEA project consisted of six thermal-hydraulic and two fission product release experiments. Four tests were devoted to loss-of-feedwater and SB-LOCA with early or delayed pump trip, as well as a secondary side feed and bleed procedure. The two LB-LOCA tests with 200% breaks were specified as licensing cases for US- and UK-conditions. At the end, two tests were performed with core temperatures allowed to rise to a point of cladding rupture and release of fission gas.

The detailed experimental results provided valuable evidence on thermal-hydraulic issues and contributed unique data to the international code validation database. Especially the two tests with fission gas release have remained as a data source on severe core damage phenomena in a large fuel bundle.

SESAR Thermal-hydraulics (SETH) Project (2001-2006)

Following a recommendation from the SESAR/FAP Group addressing thermal-hydraulic issues and safeguarding of major integral test facilities for their resolution, CSNI sponsored a project called SETH (Sesar Thermal-hydraulics) involving two test facilities, PKL for coolant system issues and PANDA for containment issues.

The PANDA programme investigated basic flow structures such as jets and plumes generated by injecting steam or steam/helium mixtures in a volume initially uniformly filled with air, steam/air mixtures or steam and the resulting effects on gas mixing and stratification. These phenomena are investigated in a multi-compartment and relatively simple geometry, and at a large scale, approaching the dimensions of actual containment compartments. The experimental programme included three test series on wall plumes, free plumes and horizontal jets, as well as one specific free plume test with three-gas mixture (steam, air and helium). Particle image velocimetry (PIV) provided detailed information on the shape, the structure and velocity fluctuations of plumes and jets. A mass spectrometer with 40 sampling ports delivered detailed information on gas distribution, stratification patterns and inter-compartment gas transport. The data obtained for the relevant variables are of high resolution and sufficient accuracy to be used for verification, validation and further development of 3D computational tools and ad hoc models in lumped parameter codes, as they are used for containment safety analysis. Enhanced codes will provide a more reliable evaluation of the containment response to severe accidents and could support planning of accident management measures.

In the PKL programme, four tests were addressing the inherent boron dilution issue after SB-LOCA where the heat was removed by steam generator (SG) secondary side cool-down with 100 K/h. All integral tests were run utilising boric acid and instrumentation to measure boron concentration. Regarding the boron-free condensate that is accumulated during reflux-condenser mode cooling, it was found that the maximum size of a condensate slug is limited by the volume of the loop seal plus part of the SG outlet pipe. When natural circulation restarts, pronounced condensate slugs arrive at the reactor pressure vessel only from the loops without ECC injection.

One integral test was devoted to the loss of the residual heat removal system in a PWR under shutdown conditions with water level in the loops reduced to ¾ height and a SG in stand-by to remove decay heat. The test provided insight into the thermal-hydraulic processes leading to re-establishing the heat sink.

The PKL test programme provided a database for validation of system codes for conditions relevant for boron dilution scenarios.

SESAR Thermal-hydraulics (SETH-2) Project (2007-2011)

The SETH-2 Project was designed to bring complementary investigations to be conducted in PANDA and MISTRA facilities. A new four-year experimental programme was designed, requiring a minimum investment in the facilities. The different size and configuration of the two facilities offered the opportunity for counterpart testing and will allow conclusions on the validity and scalability of models. The main innovations expected from the tests were: firstly, an improved basic understanding of complex flow patterns induced by condensation, spray droplets, heat and momentum sources affecting the hydrogen behaviour, such as mixing and stratification in multi-compartment confinements; secondly, the creation of a database with fine spatial resolution data for validating and further improving 3D containment codes and CFD codes; and thirdly generation of specific data for advanced containment designs utilising passive systems, such as convection systems with rupture disks, containment decay heat removal coolers, needed for verification and demonstration of predictive capabilities of lumped parameter codes.

The enhanced prediction quality of the codes will contribute to further reducing the uncertainties of these new simulation methods and to increasing their credibility and acceptance, in particular with respect to the assessment and mitigation of hydrogen risks, and to assuring and further improving the safety of present and future light water reactors (LWR), in general.

PKL Project (2004-2007)

Participating member countries to SETH found it beneficial to use the PKL facility in a follow-up project, specifically for resolving remaining questions for the boron dilution issue. Three integral tests, one of them with two runs, were conducted utilising boric acid and instrumentation to detect boron concentration. Important findings were that boron concentration at RPV inlet even under adverse conditions would not fall below 700 ppm and that natural circulation would not restart simultaneously in several loops.

Other tests were devoted to the loss of the residual heat removal system in a PWR under shutdown conditions. All integral tests were run utilising boric acid and instrumentation to measure boron concentration.

Another three integral tests, some them with several runs, were performed to further investigate PWR system behaviour under shutdown conditions when the loops are filled only to a ¾-level. The tests covered both situations: a closed and a partly open primary circuit. The objective was to investigate the impact of primary and secondary side parameters on heat removal, pressure and boron dilution. Using a SG as heat sink in case of a failure of the RHRS gives a grace period before steam is discharged into the containment through the unscrewed RPV lid. The flow conditions in this low-pressure scenario are a challenge to computer codes. The PKL project established an excellent database for validation of these transients with long periods of natural circulation and flow stagnation.

The project was accompanied by analytical activities, namely three workshops with wide participation from member countries using different computer codes and comparing the results.

Concerning the post-SB-LOCA boron dilution issue, the PKL Project has completed the understanding of the physics and thermal-hydraulics of boron dilution, so that the issue can be considered, from an experimental point of view, to be closed. The database for thermal-hydraulic system codes has been substantially broadened.

Concerning the loss of RHRS during shutdown, the tests provided insight into and improved our understanding of natural circulation phenomena under conditions with closed or open circuit. The tests demonstrated the ability of a PWR to remove the residual heat by a SG on stand-by and to re-establish a heat sink in a shut-down plant during ¾-loop operation.

PKL-2 Project (2008-2011)

After the PKL project has helped to resolve pending issues, namely the boron dilution, the participating members proposed to use the facility further to support the resolution of thermal-hydraulic issues that are relevant for current PWRs as well as for new PWR designs. A follow-up project was established that focuses on complex heat transfer mechanisms in the SGs and the boron precipitation process under postulated accident situations. The PKL-2 tests investigated safety issues relevant for current PWR plants as well as for new PWR design concepts and focused on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations. The first category includes tests addressing the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water, in both vertical and horizontal steam generators. Cooldown procedures in the case where steam generators have partly dried out on the secondary side were covered as well. A further topic was the heat transfer in the steam generators under reflux condenser conditions (e.g. fast secondary side depressurisation). Fast cooldown transients (with water filled reactor coolant system) such as main steam line break, completed by tests on mixing of hot and cold water in the RPV downcomer and the lower plenum were also considered in the test programme. Further investigation addressed boron precipitation processes in the core following large break loss-of-coolant accidents. From the last two tests one is addressing upper head void behaviour during main steam line break and the other is a counterpart test of the ROSA-2 Project, both facility addressing a small break LOCA in the hot leg with failure of HPI and initiation of accident management measure (SG depressurisation). The tests on the heat transfer mechanisms in the SGs in the presence of nitrogen are complemented by tests in the PMK test facility in Hungary for horizontal steam generators. The tests on fast cooldown transients are complemented by tests at the ROCOM test facility in Germany on mixing in the RPV downcomer and the lower plenum.

A workshop held in 2010 on analytical activities aimed to bring together code users that performed calculations reproducing the PKL-2 (PKL, ROCOM, PMK) experiments to present their simulation codes results and to discuss modelling issues and problems. The activity provided an efficient way to evaluate the current code capabilities for the transients included in the project. A proposal for a follow-up programme "PKL-Phase 3" (2012-2015) was under discussion at the end of 2011 which will include PKL, ROCOM, PMK and PACTEL (Finland) facilities.

Rig of Safety Assessment (ROSA) Project (2005-2009)

SESAR/FAP report [5] had identified LSTF as an almost unique integral test facility for studying thermal-hydraulics expected during PWR transients and accidents at full pressure and in a comparatively large scale. LSTF was put on the list of facilities to be monitored and maintained. When the facility became in danger of being closed action was taken to define an experimental programme that would address topics of common interest and would help to maintain this important facility. The ROSA Project was established using the LSTF facility. The experimental programme focused in particular on the validation needs of simulation models and methods for various complex phenomena that may occur during DBA and beyond-DBA and to increase the level of detail and accuracy in the analyses of the key phenomena during transients and accidents of interest.

Under the ROSA Project, 12 LSTF experiments were performed addressing the following five topics by either separate effect test (SET) or integral-effect test (IET), to generate data for the development and validation of both the best estimate (BE) system codes and CFD codes:

- temperature stratification and coolant mixing during ECCS coolant injection;
- unstable and destructive phenomena such as water hammer;
- natural circulation under high core power conditions;
- natural circulation with superheated steam;
- primary cooling through SG secondary depressurisation;
- small break in the RPV upper head.

These subjects were defined among the project participants from 14 countries who share the need to maintain and improve the technical competence in thermal-hydraulics for nuclear reactor safety evaluations. The obtained data have been shared among the participants and utilised for the post-test analyses by using both the BE and CFD codes. For all the experiments, pre-test analyses were done to survey optimum test conditions to meet the test objectives.

Rig of Safety Assessment (ROSA-2) Project (2009-2012)

After ROSA/LSTF has turned out to be an excellent facility for studying complex thermal-hydraulic issues in a very large scaled integral test facility at full pressure, the participants proposed to set up a follow-up project. The test programme addressed safety issues and specifically:

- intermediate break LOCA, including risk-informed break size definition and verification of safety analysis codes;
- improvements and new proposals for accident management mitigation and/or emergency operation, focused on the recovery from steam generator tube rupture (SGTR).

The experimental programme is devised to allow for an open test, which is to be defined in consultation with project members and which might cover the above issues or

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another safety relevant issue. The project provided an integral and separate effect experimental database, which will be used to validate code predictive capability and accuracy of models. Phenomena coupled with the intermediate break LOCA and with SGTR are to be investigated. The experimental programme and associated analytical activities will help to create a group among OECD/NEA member countries which share the need to maintain or improve the technical competence in thermal-hydraulics for nuclear reactor safety evaluations. By the end of 2011, five out of the six tests were completed.

PSB-VVER Project (2003-2008)

The PSB-VVER Project was established to investigate in an integral test facility thermal-hydraulic conditions that are highly relevant to the code validation for PWRs in general and to VVER-1000 safety assessments in particular.

The programme consisted of five experiments in the PSB-VVER test facility. The 11% upper plenum break test was run as a counterpart to a previous test in the smaller VVER-specific integral test facility ISB and provided information on facility differences and scaling. A test simulating SG-header rupture filled a real gap in the VVER validation, i.e. the complete absence of test data for a primary to secondary leaks. This test was selected for an analytical exercise in the frame of the project. An SB-LOCA test with delayed upper plenum injection was run as a counterpart test to existing similar tests in the integral facilities LOBI, SPES, BETHSY and LSTF. Basic data for natural circulation phenomena under "clean" VVER-specific boundary conditions were provided by another test. Unfortunately, the final test on LB-LOCA was performed, using reduced initial power of the core mock-up. Repetition of this test with full initial power is out of the boundaries of OECD PSB-VVER Project, but this test data should be accessible for project participants.

The data generated in this project filled important gaps in the VVER validation matrix established under a CSNI action [23]. Those tests that were run as counterparts to similar tests in other integral facilities enhanced our understanding of the scaling issue. The related analytical exercises gave insight into the applicability of thermal-hydraulic computer codes that were originally developed and validated for LWRs of western design.

Bubbler Condenser Project (2001-2002)

The bubbler condenser of VVER-440 (type 213) NPPs is designed to reduce the pressure of the confinement in design basis accidents and has to maintain its integrity under accident conditions. Detailed analyses identified the need to improve the modelling of accidents and to extend the knowledge of integral and separated effects. Although experimental and analytical investigations had been performed before, safety authorities requested answers to remaining questions for completing the bubbler-condenser assessment. Hungarian, Czech and Slovak utilities took the initiative to perform a joint experimental programme (not OECD/NEA sponsored) to be realised in a specialised facility at EREC, located at Electrogorsk near Moscow, which had also been used for earlier bubbler condenser experimental work.

Responding to a request from Hungary, the CSNI established a Bubbler Condenser Steering Group (BC SG) with the mandate to produce convincing evidence that the VVER-440/V213 type bubbler condenser works during DBAs as designed, to help in the planning of the new EREC experiments and in the interpretation of the results and to provide well qualified experimental results serving as basis for the validation of best estimate calculation tools.

In the years 1998-2000 three experiments were performed in the facility at EREC. These tests were promoted, financed and administered by the European Commission. Qualified experimental data were produced that strengthen the basis for computer code validation.

The OECD/NEA project was set up in 2001 to reach consensus on the conclusions from these tests and the research done before. Among the detailed conclusions the most important one is that the sequences investigated in the tests do not cause any significant challenge for the VVER-440/213 type BC and localisation system. Related analytical activities, involving 3D calculations for non-uniformities of flow distributions, improved the understanding of the phenomena observed in the tests. The project has significantly contributed to the resolution of a VVER-specific safety issue.

Fire Propagation in Elementary, Multi-room Scenarios (PRISME) Project (2006-2011)

The programme provided experimental data addressing relevant fire propagation phenomena and aiming to produce significant improvements on fire modelling capabilities. In particular, the project focused on providing unique code validation data for fire and smoke propagation from the source room to neighbouring room(s) under a variety of conditions and room configuration. Propagation through a door (opening or controlled leakage), through a pipe going through the room or by flow inversion in the duct were considered. The project concentrated on well defined separate effects and in particular on the effects of propagation mode and of configuration of neighbour rooms. The interaction of various propagation modes in realistic fire scenarios was investigated in integral tests.

The objective of the separate effects tests was to characterise the propagation mechanisms involving several rooms and quantify their respective weights on the consequences of a fire for the neighbouring rooms. To study the propagation mechanisms through a door, about six tests were foreseen in DIVA, involving 2 or 3 rooms, with various ventilation rates.

The PRISME integral test phase included six tests, defined on the basis of the results of the separate effects tests and numerical simulations.

In addition to the experiments and their analysis, extensive analyses using computer codes were performed on a voluntary basis by the signatories, with a view to providing an understanding of the phenomena of interest, and to producing a consistent interpretation of the results. This project is followed by the PRISME-2 Project (2011-2016) to address remaining issues in fire propagation.

Thermal-hydraulics, Hydrogen, Aerosols, Iodine (THAI) Project (2007-2009)

The THAI Project comprised a series of experiments in the THAI facility operated by Becker Technologies, Germany. It addressed processes in the containment, specifically open questions concerning the behaviour of: a) hydrogen, b) iodine and c) aerosols, in a severe accident. Although a severe accident project (for details see Chapter 5), THAI is mentioned here because the dominating phenomena involved are mostly thermal-hydraulic: natural convection, stratification and mixing of gases and steam, evaporation and condensation, etc. The database developed in frame of this project has been and will be used to validate lumped parameter and CFD tools available and under development for containment analysis.

Loss of Forced Coolant (LOFC) Project (2011-2013)

The CSNI Task Group on Advanced Reactor Experimental Facilities (TAREF) issued a report in June 2009 suggesting to develop an international programme centred on the high temperature engineering test reactor (HTTR) operated by JAEA in Oarai. The LOFC project started in 2011 to investigate safety issues and specifically the anticipated transient without scram (ATWS) with occurrence of reactor re-criticality. The test consists of three test cases, run one through three, whose results comparison will provide the incremental performance availability within the vessel cooling system (VCS) range. The programme is devised to maximise the information deliverables for code validation for one of the most important safety aspects about reactor kinetics, core physics and thermal-hydraulics for gas-cooled reactors.

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3.5 Summary and conclusions

Thermal-hydraulic issues were dominating safety concerns and rulemaking from the early days of nuclear power plant operation. The focus shifted from large break LOCA to more frequent initiating events. The issues connected with loss-of-coolant accidents have been mostly resolved by analytical and experimental programmes, many of them in international co-operation. Validation matrices, international standard problems and results from NEA joint research projects have substantially contributed to a consensus on the technical basis for closing classical thermal-hydraulic issues. The process of resolving recent issues like boron dilution and strainer clogging has greatly profited from these projects.

The LOFT Project gave the participating countries access to a unique nuclear test facility and helped them resolving their national safety cases involving LOCAs. The SESAR Projects were successful in maintaining the test facilities PANDA, PKL and MISTRA. Mixing phenomena were investigated and data for CFD code validation were provided. The PKL Projects investigated natural circulation in a four-loop integral facility and largely supported the resolution of the boron dilution issue. The ROSA Projects investigated complex thermal-hydraulic phenomena in a large scale facility under full pressure and strengthened the database for system code validation. The PSB-VVER Project and the Bubbler Condenser Project have significantly contributed to the resolution of VVER-specific safety issues in the cooling circuit and in the confinement. The PRISME Projects investigated fire and smoke propagation in a complex arrangement of large rooms. Modelling the circulation and mixing of hot gases and their interaction with the structure will profit from the data generated in this project.

Operating experience, lifetime management and lessons learnt from events may pose new questions requiring safety research also in the thermal-hydraulic area. Issues that may arise for new reactor designs are on the agenda of the pertinent working groups of CSNI.

Computer code modelling has advanced from conservative assumptions to a best-estimate approach, complemented by methods for quantifying uncertainties. Computational fluid dynamic codes are fast entering the nuclear industry. Development and validation for two-phase flows is in progress. The growing CFD application for safety demonstration has to be accompanied by commonly accepted best practices.

Thermal-hydraulic research has greatly advanced our understanding of phenomena that dominate transients and accidents and how to model them. NEA joint projects have largely supported this achievement.

4. Fuel behaviour

4.1 Background

The largest quantity of radioactive material in the plant is contained in the fuel in the form of fission products and higher actinides. These build up over the lifetime of the fuel and, to varying degrees, have the potential to be released from the fuel in the event of fuel damage. Fuel elements not only form the reactor active core but they are found in the spent fuel pool as well where cooling has to be maintained until they are removed from the reactor site.

The fuel cladding is the barrier to prevent radioactive material from being released in case of an accident. Moreover, it ensures that the fuel maintains a coolable geometry. Preserving the cladding integrity is thus a fundamental safety concern and failure mechanisms have to be thoroughly understood.

For currently operating light water reactors, the accidents with the greatest potential for fuel damage are large reactivity insertions (RIA), loss-of-coolant accidents (LOCA) and events that cause a departure from nucleate boiling on the cladding. The behaviour of the fuel under these conditions will be affected by the fuel burn-up level, cladding condition (e.g. oxidation and embrittlement) and location in the core.

Nuclear power plants have been designed with substantial margins in operational and safety system performance. As analytical methods have been refined and existing margins to the acceptance criteria have been quantified using a best-estimate methodology, utilities were allowed to increase reactor power in their existing nuclear power plants (NPPs). In addition, improvements in plant instrumentation have permitted small power increases, taking advantage of lower uncertainties than those assumed in some regulations. These increases, referred to as power uprates, are means to improve plant economic performance. Besides the question of fuel behaviour, higher reactor power results in changes in other plant operating conditions, including increased hydraulic forces, corrosion, decay heat, and flow-induced vibration.

Load-following is a mode of operation that imposes changing conditions on the fuel especially in the vicinity of control rods. Pellet clad interaction (PCI) could impose limits on this flexibility. The ability of NPPs to react timely on varying demand in the grid is becoming more important as a growing fraction of electricity comes from sources with unsteady production (wind and solar). For economic reasons, power uprates and longer fuel cycles have been realised in many NPPs, and more of them are still expected. Higher fuel discharge burn-up levels enhance economic performance further. Besides the wish to make best use of the original enrichment there are additional considerations about transport, storage and final treatment of the fuel. The fuel strategies of utilities and vendors are influenced by the possibilities and cost of reprocessing and final disposal in the individual member countries. Limits to burn-up will be set by operational and safety considerations.

With all measures taken for higher plant performance, attention must be paid to maintaining adequate operational and safety margins.

Industry has developed advanced fuel assembly designs in order to fulfil higher demands regarding thermal-mechanical performance, and new, more corrosion resistant materials for cladding and assembly structure. Currently, fuel pellets with better performance regarding fission gas release and improved mechanical properties are about to be introduced as potential future standard material. Improved plant availability and more reliable and predictable plant operation are only possible with robust and reliable fuel. This leads to an increased significance of each individual fuel rod failure.

For advanced light water reactors, it is expected that even higher burn-up levels will be desired for economic and other reasons e.g. proliferation resistance. To support such fuel designs, higher enrichment, burnable positions and additional advances in cladding materials will likely be needed. Ensuring these new fuel designs achieve an acceptable level of safety will require testing and analysis to confirm fuel performance, validate analysis tools and establish safety limits.

4.2 Fuel issues

The safety issues associated with current LWR fuel are related to deciding where to establish safety limits based upon how fuel performance under accident conditions changes with changes in burn-up, cladding material and service condition.

High burn-up and mixed oxide (MOX) fuel

The trend to higher discharge burn-up is a challenge to fuel behaviour during transients and accidents. High burn-up with longer exploitation times in the reactor may affect cladding integrity. Increased fission gas pressure imposes higher stress on the cladding. Oxide layers on the cladding may grow and hydrogen uptake enhances embrittlement and makes the rod more likely to fail during an accident. In some fuel designs, higher enrichment entails the use of burnable poisons in the fuel. Fuel pellets tend to fragment and small fragments could even relocate when the cladding tube is ballooning. The use of mixed oxide fuel can make these effects even more pronounced.

Fuel design changes and new materials

Responding to utilities' demands, fuel vendors developed advanced designs for fuel elements with modifications on the side of the moderator and coolant, e.g. water channels for BWR assemblies to optimise power shape and turbulence enhancing spacers to provide margin for heat flux limits in PWR assemblies.

The most significant advances were made by industry on the cladding tube materials and processing during fabrication. New zirconium alloys with niobium content were introduced that are more resistant to corrosion. With the changes in material properties, the response to accidental conditions needs to be assessed. A large part of this assessment relies on correlation of data from in-pile and out-of-pile testing. It has to be ensured that this database is extended to capture the new materials and the wide range of operational and accidental conditions. Model development and validation has to keep up with the new empirical data.

Acceptance criteria for loss-of-coolant accidents (LOCA) and reactivity initiated accidents (RIA)

Water-cooled reactors are equipped with an emergency core cooling system (ECCS) to cope with this category of design basis accidents. Acceptance criteria have been established to assess the ECCS to ensure that the reactor core remains amenable to cooling throughout the accident and the structural integrity of the fuel is maintained. Demonstration that the criteria are met turned out to be a real challenge due to the particular properties of the zirconium alloys used as cladding material. The metallurgical and mechanical phenomena under LOCA conditions are complex and have been subject to extensive experimental and analytical studies. A necessary condition for updating the acceptance criteria in light of new findings from research would require a firmly established and widely recognised database.

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As fuel assembly designs and cladding materials change, MOX fuel is introduced and burn-up increases, the issue of reactivity insertion accidents needs to be revisited. Failure modes and criteria need to be investigated and updated based on experimental data, consistent with design basis and beyond design basis events.

Power oscillations

Power oscillations have been observed in some BWRs. Understanding and modelling them has been a challenge to 3D coupled thermal-hydraulic/neutronic codes. Although not primarily a fuel issue, questions were raised on fuel response to such type of transient. There is a need to better understand the impact of power uprates on the potential for oscillations including ATWS conditions. High burn-up fuel might be more sensitive to the changing power and cooling conditions.

4.3 CSNI fuel activities

The importance of fuel issues in a comprehensive approach to safety has been recognised by CSNI from the beginning. Under the previous Operating Plan (OP), fuel was one of the explicitly mentioned safety issues/topics (SITs). The new OP that came into force in 2011 [20] addresses fuel safety explicitly by defining as one of the technical goals to assess the impact of new fuel technologies on the safety of existing nuclear installations. Activities of the Working Group on Fuel Safety (WGFS) have included:

- benchmarking of calculation tools and methods, such as fuel performance codes against HALDEN and other projects' tests;
- topical meetings on the design basis accidents of most interest, knowledge assessment through state-of-the-art reports (SOARs), recently on LOCA and RIA [25, 26].

The new CSNI OP [20] has revisited and confirmed the mandate of the WGFS. The main objective is to advance the current understanding and address safety issues related to fuel safety. The group is mainly charged with assessing the technical basis for current safety criteria and their applicability to high burn-up and to new fuel designs and materials, determining needs and priorities for future research programmes with the aim of understanding and adequately modelling key phenomena and of quantifying safety margins, and with reviewing the adequacy of fuel codes and methodologies used for different core assessments as related to high burn-up fuel. The group will aim at facilitating international convergence in fuel safety issues, including experimental approaches, interpretation and use of the experimental data or of other relevant information.

4.4 OECD/NEA joint projects addressing fuel issues

Safety research on fuel behaviour under operational, transient and accident conditions always relied on experimental data because the knowledge base is mostly empirical in this area. The Halden Project was once established to investigate fuel behaviour in a dedicated research reactor in Norway and fuel issues have been an important part of all successive programmes of the Halden Project since, which are not described in this report.

Another dedicated research reactor, CABRI in France, was selected to establish a research programme on high burn-up fuel under RIA conditions. Specific aspects of cladding behaviour under specific operational and transient conditions continue to be investigated at Studsvik (Sweden).

Finally, an incident in the Hungarian Paks-2 NPP offered the opportunity to gain insight in the behaviour of a considerable number of full size fuel elements outside the reactor under insufficient cooling conditions.

The following paragraphs describe briefly the fuel projects and how they have contributed to the understanding of phenomena, to the development of analytical tools and to the resolution of safety issues. Summary and other pertinent information for each project is provided in Appendix 1.

CABRI Water Loop Project (2000-2015)

The project aims at understanding the behaviour of high burn-up fuel under reactivity initiated accident (RIA) conditions. Samples of fuel having served over several cycles in commercial power reactors are used. Inserted in the CABRI experimental reactor, they are exposed to reactivity ramps by the driver core and undergo high power pulses that challenge their integrity.

Results of two tests already performed in the sodium loop of the CABRI facility in 2002 were made available to the participants. Considering the available data and the capabilities of other facilities it appeared advisable to complement the current results by tests on very high burn-up using advanced fuels and to perform tests in fully representative PWR conditions, specifically for phenomena occurring after the initial pellet cladding mechanical interaction (PCMI). For this reason, it was decided to implement a pressurised water loop in the CABRI reactor where the new tests of the project will be performed. These tests will assess departure from nucleate boiling (DNB) and post-DNB phenomena as well as post-failure events for high burn-up and ultra high burn-up fuel and MOX fuel, specifically addressing fission gas behaviour. The findings from these in-pile experiments will allow assessing the response of high burn-up fuel to postulated RIA. These findings, together with data from other experimental facilities, could serve as a basis for revising the current acceptance criteria for RIA in several Member countries.

According to the original planning, the years 2003 and 2004 were devoted to the implementation of the water loop in the CABRI reactor after dismantling the sodium loop. Due to technical difficulties and requests for safety upgrading of the reactor the project is substantially delayed.

Studsvik Cladding Integrity Project (SCIP) (2004-2009)

The overall objective of the project was to provide experimental data that are needed for improving the understanding of the dominant failure mechanisms for water reactor fuels. The major focus was on cladding failures that are caused by pellet-cladding mechanical interaction (PCMI) especially stress corrosion cracking (SSC) 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 risk of fuel failures. This understanding is to be applicable to pellet-cladding interaction conditions that can arise during normal operation or in anticipated transients, as well as during fuel long-term storage. The programme was intended to complement other international projects in the fuel area. The project utilised the hot cell facilities and expertise available at the Swedish Studsvik establishment. Extensive analyses and theoretical modelling of the fracture mechanisms accompanied the experimental programme.

The understanding of the effect of burn-up and power ramp profile on the stresses and strains has been improved for claddings with standard UO_2 pellets. In the area of PCI, an out-of pile testing technique designed for testing of irradiated cladding has been developed. The results obtained within SCIP have improved our understanding of PCMI behaviour and PCMI driven failures, especially hydrogen induced failures in high burn-up cladding. We now have a better understanding of the combined effect of different materials and test parameters on hydrogen induced failures.

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Studsvik Cladding Integrity Project – Phase 2 (SCIP-2) (2009-2014)

The overriding objective of the continuation of the SCIP Programme (SCIP-2) is to strengthen the safety aspects of the plant operation and to contribute to more reliable fuel by further deepening the understanding of mechanisms leading to fuel failures driven by pellet-cladding mechanical interaction (PCMI), in particular failures due to stress corrosion cracking and failures caused by hydrogen-assisted fracture.

This follow-up project, on the basis of the considerable knowledge generated in the previous SCIP Programme, is to generate high quality experimental data that are needed for improving the understanding of the dominant failure mechanisms for water reactor fuels and devise means for reducing fuel failures. In SCIP-2, the scope will be to look further into modern cladding materials. Furthermore, advanced pellets with additives and larger grains will be included. The studies with new materials aim at confirming similarities with well established materials on one hand, at identifying potential differences and consequences of these differences on failure-related performance on the other hand.

Paks Fuel Project (2004-2007)

The initiator for this project was an incident in the Hungarian Paks-2 NPP in 2003, where several fuel assemblies were damaged due to insufficient coolant circulation following a special crust removal operation in a special cleaning tank connected to the spent fuel storage pool.

A joint project between the IAEA and the OECD/NEA was established in 2005 with the aim to contribute to a complete understanding of the incident and the consequences it had on the fuel. Moreover, it was considered a useful case for a comparative code exercise, in particular for models devised to predict fuel damage and potential releases under abnormal cooling conditions.

Numerical simulation of the most relevant aspects of the event and comparison of the calculation results with the available information was carried out between 2006 and 2007. A database was collected to provide input data for the code calculations. The activities comprised the following three tasks:

- Thermal-hydraulic calculations described the cooling conditions possibly established during the incident.
- Simulation of fuel behaviour describing the oxidation and degradation mechanisms of fuel assemblies.
- Estimation of the release of fission products from the failed fuel rods with comparison to available measured data.

The produced numerical results improved the understanding of the causes and mechanisms of fuel failures during the Paks-2 incident and provided new information on the behaviour of nuclear fuel under accident conditions.

Sandia Fuel Project (SFP) (2009-2012)

Fuel assemblies stored in a spent fuel pool have to be kept under continuous cooling. Studies with severe accident codes indicate that in case of a complete loss-of-coolant accident fuel assemblies can ignite and the ignition can propagate radially. Hence, there is a need of qualified data obtained in representative fuel configurations. The objective of the project to 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.

The experiments focus on thermal-hydraulic and ignition phenomena in PWR 17x17 assemblies and supplement earlier results obtained for BWR assemblies, which are made available to the SFP Project. It is believed that 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 events in Fukushima Daiichi Unit 4 have highlighted that the loss of heat removal from the spent fuel pool can lead to significant fuel damage. Full understanding of the phenomena after dryout and the timing of events are of great importance for managing such events.

4.5 Summary and conclusions

Maintaining fuel and cladding integrity during transients and accidents is a fundamental safety concern. Therefore, failure mechanisms ought to be thoroughly investigated and understood.

Power uprates, load following, longer cycles and higher discharge burn-up mean much more demanding fuel operating conditions. Care has to be taken that sufficient safety margins are maintained.

Fuel research and development is one of the areas that bring innovation to existing reactors. Another specificity is the direct impact on plant performance which entails competition between fuel vendors.

NEA joint projects have addressed quite different fuel issues of common interest, all of them of high safety significance. The CABRI reactor offers an almost unique opportunity for in-pile testing of high burn-up fuel under realistic PWR conditions. Hot cells are another important infrastructure for fuel safety research. The Studsvik Cladding Integrity Projects have profited from the facilities there and the analytical capabilities around it. The Paks incident was recognised as an opportunity to gain insight in the behaviour of a large amount of real fuel under degraded cooling conditions, to observe beyond design basis phenomena and the capabilities of computer codes to predict them. The Sandia Fuel Project acknowledges the fact that transients and accidents may challenge fuel integrity not only in the reactor core but in the spent fuel pool as well. The Fukushima event was a lesson on this issue.

In summary, the activities of the CSNI Working Group on Fuel Safety and NEA joint projects in the fuel area are responding to the challenges of power uprates, higher burn-up, new fuel element designs and new cladding materials. Thorough understanding of phenomena and failure mechanisms as well as common assessment of experimental data and computer code models strengthen the technical basis for a possible revision of acceptance criteria.

5. Severe accidents

5.1 Background

The potential risk to the public from nuclear power generation arises from low-frequency accidents that are predicted to progress to the point where fuel degradation occurs and large quantities of radioactive material may be released to the environment. These accidents, termed severe accidents, are generally considered to be events beyond the traditional design basis of the currently operating nuclear power plants. The prevention or mitigation of severe accidents is the largest contributor to reducing risk to the public from the operation of the nuclear power plants. The scenarios involve initiating events accompanied by multiple failures resulting in non-coolable geometry and significant damage to the core, and potentially leading to a release of significant amount of radioactive material to the containment. Under certain circumstances, the containment may also be postulated to fail or to be bypassed resulting in a major release to the environment. Therefore, reactor and containment designs are a vital link in prevention and mitigation of accidents. Understanding important challenges to the integrity of the reactor fuel, reactor pressure vessel and the containment requires systems analysis and models of plant response to various severe accident sequences. A realistic assessment of severe accident source term requires modelling of a wide range of phenomena associated with core melt progression, containment performance, and fission product release and transport. The amount, composition, chemistry, and timing of fission product release along with hydrogen from the fuel through the reactor coolant system (where some fission products may be retained as a result of various types of interactions) and into containment result in the source term available for release to the environment. Estimating the transport of source term requires understanding the phenomena that can challenge the integrity of the cladding, reactor pressure vessel and the containment under harsh conditions. These factors affect the onsite and offsite consequence analysis and protective actions which need to be planned for. The source term is also affected by the type of fuel and burn-up. The effectiveness of design features (e.g. depressurisation) and measures (e.g. sprays, water chemistry, filters) to attenuate source term can be determined with validated analytical tools.

Complementary use of deterministic and probabilistic analyses are necessary to fully understand the complex phenomena involved and the associated importance and uncertainties in evaluation of their impact. Many important national and international programmes have been undertaken in the field of severe accidents and their results have been shared through international networks. CSNI has played a major role in organising and administering co-operative research projects in the area of severe accidents. These projects include The Three Mile Island Pressure Vessel Investigation, Phebus-FP (performed in co-operation with EC and not under OECD but involving nearly all NEA countries), RASPLAV and the follow-on projects MASCA-1 and 2 (conducted in Russia to develop capability to assess the RPV integrity under core melt conditions), OECD-LHF (conducted in the United States to assess the mechanical behaviour of the RPV lower head under pressurised severe accident conditions), MCCI (conducted in the United States to assess ex-vessel molten core debris coolability), Behaviour of Iodine (conducted in Canada), SERENA (conducted in France and Korea following an analytical programme assessing the state of knowledge related to fuel-coolant interactions) and THAI (conducted in Germany). These programmes and the CSNI activities on accident management have contributed to knowledge about severe accident phenomena, the resolution of many questions related to severe accidents and accident management measures (features and procedures) to terminate or mitigate the accident progression. Besides allowing development of deterministic models, these efforts have also contributed to the capability to conduct probabilistic evaluations of severe accident scenarios. CSNI has published a report [27] that provides the status of the current understanding of important severe accident issues for various nuclear power plant designs. In addition, another report [28] provides an excellent summary of phenomenological uncertainties in the severe accident evaluation methodologies. However, it should be noted that the lessons to be learnt from the recent events at the Fukushima Daiichi nuclear power plants (e.g. in-vessel retention, accident management) would need to be studied carefully to either confirm earlier conclusions and/or develop any new safety research projects.

5.2 Severe accident issues

A fundamental safety objective in any NPP programme is to protect people and the environment from harmful effects of radiation such that the individuals living or working near nuclear power plants should be able to go about their daily lives without special concerns by virtue of their proximity to the plant(s). Traditional NPP safety requirements are based on deterministic considerations that employ a defence-in-depth safety philosophy. This approach to safety can also be complemented by careful consideration of insights from probabilistic safety assessments through the application of a risk-informed process. The integration of deterministic and probabilistic factors leads to a more robust design and operational practices and thus helps ensure that the risk to individuals (and the environment) is acceptably low.

Broadly, the IAEA and member states have prepared documents that require that strategies be developed to prevent accidents, terminate progression of core damage, maintain containment integrity as long as possible, minimise release of radioactive material and achieve long-term stable state. It is also proposed that Level 1 and Level 2 PSA can be useful in identifying severe accident challenges that should be addressed through accident management programme. Severe accident management includes design features to address important challenges and phenomena, instrumentation, procedures and training, including command and control issues.

The risk to the members of the public is largely controlled by severe accidents where the core may have melted and containment function may also have been compromised. In order to develop design features and accident management guidelines to deal with such challenges, it is critical that the progression of such accidents and the associated impact on systems and structures be understood. Achieving this understanding has required extensive experimental programmes and development of validated codes for various analyses. Three decades of extensive severe accident research has identified the following key severe accident issues:

- in-vessel melt progression;
- hydrogen generation and control;
- core-debris concrete interaction – ability to cool core debris;
- high pressure core melt ejection;
- steam explosions;
- containment performance under severe accident conditions;
- equipment survivability;
- accident management.

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In-vessel melt progression

The amount, composition, rate and timing are important to determine the effectiveness of accident management measures and, the ability of the reactor pressure vessel to maintain its integrity. Further, extensive studies have been conducted to determine if the reactor vessel cooled from outside would retain molten core debris within the vessel. With the reactor vessel intact and molten core debris retained in the lower head, phenomena such as ex-vessel steam explosion and core-concrete interactions, which occur as a result of core debris relocation to the reactor cavity, could be prevented. This severe accident management strategy of in-vessel melt retention has been approved for the Loviisa plant in Finland and has been incorporated in the designs of AP 600 and AP 1000 passive plants. Reactor vessel integrity is assumed if the reactor coolant system is depressurised and the compartment containing the reactor vessel adequately flooded.

In-vessel steam explosion

During the initial stages of progression of severe accidents, molten debris from the damaged core would relocate to the lower plenum of the reactor pressure vessel. If a sufficient amount of water remained in the lower plenum, the molten core material falling into the water could generate steam and if severe enough, an explosion. This explosion has the potential to challenge the reactor vessel and containment integrity. It should be pointed out that higher burn-up levels and MOX fuel could change the dynamics of melt progression and fission product release.

Containment performance under severe accident conditions

The goal is to ensure that the containment structure has a high probability of withstanding the loads associated with severe accident phenomena, and that the potential for significant radioactive releases from containment is small. The containment should be assumed to have failed if any of the following conditions occur:

- containment structural failure;
- the containment is bypassed;
- the containment fails to isolate;
- the containment seal materials fail as a result of over-temperature;
- the molten core debris melts through the concrete basemat into the subsoil.

Ex-vessel steam explosion

Reactor vessel failure at high or low pressure coincident with water present within the reactor cavity may lead to interactions between fuel and coolant with a potential for rapid steam generation and, depending upon the amount, rate, fragmentation and mixing of the molten material possible steam explosions. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporisation and acceleration of surrounding water creating substantial pressure and impact loads.

Hydrogen generation and control

The issue regarding hydrogen generation centres on the rate and quantity of hydrogen production and the associated hydrogen-steam mass and energy release rates into the containment during both in-vessel and ex-vessel phases of severe accidents. These parameters strongly influence the flammability of the containment atmosphere and the magnitude, timing, and location of potential hydrogen combustion. Hydrogen combustion in the containment could produce pressure and thermal loads that may threaten the integrity of the containment boundary. There are uncertainties in the phenomenological knowledge of hydrogen generation and combustion. Severe accident research results are used to design mitigation features.

Core debris-concrete interaction – ability to cool core debris

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensable gases, which could contribute to eventual overpressure failure of the containment. Downward erosion of the basemat concrete may also lead to basemat penetration with the potential for ground water contamination and subsequent discharge of radionuclides to the surface environment. Also, thermal attack by molten corium on retaining sidewalls could produce structural failure within the containment causing damage to vital systems and perhaps to failure of containment boundary. Therefore, the issues of importance are a) reactor cavity floor space to enhance debris spreading, b) means to flood the reactor cavity to assist in the cooling process and c) protection for the containment liner and other concrete structural members within the reactor cavity.

High pressure core melt

A high pressure core melt ejection (HPM) could lead to containment failure with radioactive releases to the environment. HPM is the ejection of core debris and hydrogen from the reactor vessel at high pressure. This would cause fragmentation and dispersal of core debris and hydrogen within the containment atmosphere, termed direct containment heating (DCH), that has the potential for causing early containment failure. Failure could occur due to the heat-up and pressurisation of the containment due to hydrogen combustion and core debris. In addition, HPM can lead to direct attack on the containment shell. However, results of more recent research indicate that the likelihood of this failure mode is low.

Another potential result of high pressure melt conditions can be a thermally induced failure of steam generator tubes and associated containment bypass. The creep rupture failure mode of the steam generator tubes depends on several factors including the thermal-hydraulic conditions at various locations in the primary system which determine the temperature and pressure to which the steam generator tubes are subjected as the accident progresses. Presence of defects in the steam generator tubes will increase the likelihood of this failure mode.

The research results have identified the importance of having capability to be able to depressurise the primary system under the severe accident conditions.

Equipment survivability

The survivability of certain equipment, both electrical and mechanical, is needed to prevent and mitigate the consequences of severe accidents. Severe accidents can be divided in two classes, in-vessel and ex-vessel severe accidents. During the in-vessel events, the core is losing its coolability, leading to at least a partial fuel melt. During the ex-vessel events, a reactor vessel failure is assumed, leading to a relocation of molten corium to the containment. Such postulated severe accidents result in environmental conditions that are generally more limiting than those from design-basis events. A review of the credible severe accident scenarios will identify all equipment, both electrical and mechanical, and instrumentation that can support implementation of severe accident management strategies.

Accident management

Accident management (AM) encompasses those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimise offsite releases. In effect, AM extends defence-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe accident regimes, and by making full use of plant equipment and operator actions to terminate severe accidents and limit offsite releases. Although new designs of nuclear power plants have enhanced capabilities for the prevention and mitigation of severe accidents, AM will remain an important element of defence-in-depth for all nuclear power plants. The new designs of nuclear power plants will tend to relieve the operators of the need for rapid decisions and permit a greater reliance on support from outside resources. The overall responsibility for AM, including development, implementation, and maintenance of the accident management plan, lies with the nuclear utility, because the utility bears the ultimate responsibility for the safety of the plant and for establishing and maintaining an emergency response organisation capable of effectively responding to potential accident situations.

5.3 International severe accident research and the NEA role

The international nuclear community has been conducting research on severe accidents for the existing reactors for decades and has used the results to inform and improve nuclear power plant safety, ability to conduct risk assessments and to improve regulations that in part stem from consideration of severe accidents. In the specific area of severe accident risk, much of the early research was conducted by national organisations. As the safety research budgets have contracted and some experimental facilities have been shutdown and others in danger of being shutdown, the NEA platform has allowed an effective way of getting information to fill the remaining gaps in knowledge.

Experience has shown that these NEA activities entail substantial analytical activity accompanying the execution of the experimental programme. This activity is centred on code assessment and validation and where suitable, on model development. Code benchmarking or analytical exercises consisting of both pre-test and post-test calculations are organised among the project participants. This analytical activity has proven to be a very effective manner to maintain or develop relevant technical expertise. The analytical tools have been used to refine estimates of power plant risk, such as from previously postulated early containment failure mechanisms (large-scale steam explosions that fail containment, direct containment heating, and, for large-dry containments, hydrogen combustion) and to understand the effectiveness of mitigation measures, as appropriate.

Moreover, research results are being used to estimate the source term and the offsite consequences of selected severe reactor accidents. Severe accident research has also been used to support development of risk-informed regulations. The international community has benefited from the CSNI severe accident research projects to resolve complex issues and to maintain critical expertise to deal with the remaining concerns with severe accidents. In this regard, in 2002, the Committee on the Safety of Nuclear Installations (CSNI) established a senior group of experts on nuclear safety research (SESAR) to assess the need for and strategy of maintaining key research facilities [6]. In 2007, OECD published the outcome of the SESAR review in *Nuclear Safety Research in OECD Countries: Supporting Facilities for Existing and Advanced Reactors (SFEAR)* [16].

The following paragraphs describe briefly the severe accident projects and how they have contributed to the understanding of phenomena, to the development of analytical tools and to the resolution of safety issues. Summary and other pertinent information for each project is provided in Appendix 1.

5.4 OECD/NEA joint projects addressing severe accident issues

Three Mile Island Reactor Pressure Vessel Investigation OECD Project (1988-1993)

This programme was designed to evaluate the potential modes of failure and the margin to failure of the TMI-2 reactor vessel during the TMI-2 accident. The conditions and properties of material extracted from the lower head of the TMI-2 reactor pressure vessel were investigated to determine the temperature conditions and the extent of the damage by chemical and thermal attack on the lower head, as well as the margin of structural integrity of the vessel during the accident. This led to the understanding of the TMI-2 accident scenario and the lower head conditions. This project showed that global creep failure of the reactor vessel could occur under conditions of high vessel temperature and pressure. These findings have supported accident management strategies.

RASPLAV Project (Phase 1: 1994-1996 – Phase 2: 1997-2000)

Little was known about the complex interactions that take place during a core meltdown, so one of the RASPLAV project's primary goals was to develop an understanding of this process. The information gathered during the large scale tests at the Kurchatov Institute has allowed scientists to develop models of a core meltdown. These models can be used in the design of new reactors and in refining the accident procedures for existing ones.

Two aspects of the issue were considered. First, for existing reactors, where external cooling may not be practicable, the process and time sequence before melt-through were studied. This was to help develop management strategies for severe accidents. Secondly, for future and some existing reactor designs, the project aimed to determine the heat transfer conditions under which cavity flooding can be a viable accident management option.

The project was run in two successive phases. The RASPLAV Phase-2 Project investigated the progression of a severe accident and in particular the thermal loading imposed by a corium pool on the lower head of a light water reactor (LWR) vessel. It followed an earlier Phase-1 Project dedicated mainly to the build-up of the experimental and analytical infrastructure.

Material Scaling (MASCA) Project (2001-2003)

The MASCA Project was a follow-up of the RASPLAV Project and investigated in-vessel phenomena during a severe accident. Maintaining MASCA was one of the SESAR/FAP recommendations [5]. In particular, MASCA addressed the influence of the chemical composition of the molten corium on the heat transfer to the pressure vessel environment. The project addressed this by investigating stratification phenomena of the molten pool and the partitioning of fission products (FP) within the different layers of the melt. The tests aimed to resolve remaining uncertainties about the heat load on the reactor vessel and thus the possibility of retaining the melt in the vessel. The experiments were carried out with corium compositions prototypical of power reactors which use iron and steel materials. The MASCA experimental goal was achieved through corium tests of different scale and was complemented by pre- and post-test analyses and development of computational models. Additional measurements of thermo-physical properties of the melts, such as density, thermal conductivity and liquidus-solidus temperatures considerably expanded the material properties data obtained during the RASPLAV Project.

Material Scaling-Phase 2 (MASCA-2) Project (2004-2006)

The MASCA-2 Project was an extension of the MASCA Project. It was based on experiments that were mainly carried out at the Kurchatov Institute and that make use of a variety of facilities in which the corium compositions prototypical of power reactors could be tested. The tests aimed to provide experimental information on the phase equilibrium for different corium mixture compositions that can occur in water reactors. In order to enhance the application of MASCA results for reactor cases, the influence of an oxidising atmosphere and the impact of non-uniform temperatures (presence of crusts or solid debris) was addressed. This 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.

OECD Sandia Lower Head Failure (OLHF) Project (1998-2002)

Although the Three Mile Island vessel did not fail, code analyses conducted in the course of an OECD/NEA TMI-II Vessel Investigation Project (VIP) predicted creep rupture in the prevailing conditions. This implies that the existing state-of-the art modelling of the lower head failure was not mature because it did not take full account of the effect of the thermal loading. These methodologies have been further developed since the TMI-VIP project to analyse existing and next generation reactors from the perspective of accident assessment, management, and mitigation. In order to improve and validate structural analysis codes, there was a need for experimental data on lower head deformation and failure phenomena.

During a severe accident, the lower head of the reactor pressure vessel (RPV) can be subjected to significant thermal and pressure loads. It is possible that the lower head will fail, releasing large amounts of molten corium into containment. The Three Mile Island accident involved the melting of about 20 tonnes of corium, which collapsed into the lower head of the RPV. Despite the presence of water, the lower head reached temperatures of ~1 300 K for 30 minutes in an area with an equivalent diameter of 1 m. During this period the reactor cooling system was at 10 MPa. The Sandia National Laboratory conducted eight USNRC-sponsored tests on lower head failure with prototypic material and geometry. The objective of this OECD project was to extend the USNRC SNL/LHF programme to address such issues as the timing and size of lower head failure under conditions of low reactor coolant system pressure and large differential temperatures across the lower head wall.

Melt Coolability and Concrete Interaction (MCCI) Project (Phase 1: 2002-2005 – Phase 2: 2006-2010)

In a core melt accident, if the molten core is not retained in-vessel despite severe accident mitigation actions, the core debris will relocate to the reactor cavity region and interact with the structural concrete, potentially resulting in basemat failure through erosion or overpressurisation. This would result in the release of fission products into the environment. Although this is a late release event, the radiological consequences could be substantial enough to warrant an effective mitigation strategy for preventing such a release. The severe accident management guidance (SAMG) for operating light water reactor plants includes, as one of several strategies, flooding the reactor cavity in the event of an ex-vessel core melt release.

The OECD-sponsored Melt Coolability and Concrete Interaction (MCCI) Programme is conducting reactor material experiments and associated analysis with the objectives of resolving the ex-vessel debris coolability issue, and to address remaining uncertainties related to long-term two-dimensional core-concrete interactions under both wet and dry cavity conditions. Achievement of these two objectives will demonstrate the efficacy of severe accident management guidelines for existing plants, and provide the technical basis for better containment designs for future plants. In addition to identifying the desired concrete material for basemat designs, these tests have provided a broad database for the development and validation of models and codes that are used to extrapolate to plant scale. The first phase of the programme (MCCI-1) was completed in 2005 and showed that cooling of the melt is reduced at increasing concrete content. The effect of concrete type was also addressed as well as derivation of material properties such as porosity and permeability. A subsequent three-year programme (MCCI-2) has been completed. The scope of this programme was to conduct separate effects and integral tests to investigate 1) interplay of different cooling mechanisms, and to provide data for model development and code assessment, 2) investigate new design features to enhance coolability, 3) generate two-dimensional core-concrete interaction data and 4) validate severe accident codes. The programme has achieved its objectives of improved severe accident management guidelines for existing plants as well as better containment designs for future plants. Some uncertainties remain if significant melt metal fraction is present that may result in a stratified pool configuration.

Behaviour of Iodine Project (BIP) (2007-2011)

This research was performed to quantify various processes on surfaces leading to gaseous iodine formation; measure adsorption/desorption rate constants on containment surfaces as a function of temperature, relative humidity, and carrier-gas composition in humid environments; and measure absorption of iodine on sump materials.

The NEA Behaviour of Iodine Project (BIP) has been created to provide separate effects and modelling studies of iodine behaviour in a nuclear reactor containment building following a severe accident. This joint project complemented other national and international experimental programmes which are also studying this phenomenon. As part of the project the results of three radioiodine test facility (RTF) experiments were provided by Atomic Energy of Canada Limited (AECL). Project participants seek to combine international resources to produce a consolidated understanding of the behaviour of iodine and other fission products in this scenario. The results provided by this project will be useful for regulators and operators in the context of managing post-accident situations in the containment building. A follow-up three-year project (BIP-2) is addressing the remaining issue as regards iodine interaction with panting and organic iodine formation (2011-2014).

Steam Explosion Resolution for Nuclear Applications (SERENA) Project (2008-2012)

The NEA co-ordinated Programme on Steam Explosion Resolution for Nuclear Applications (SERENA) was established to assess the capabilities of the current generation of fuel-coolant interaction (FCI) computer codes to predict steam explosion-induced loads in reactor situations. One of the main findings of this programme was that in-vessel FCI would not challenge the integrity of the nuclear reactor containment but that this scenario could not be ruled out for ex-vessel FCI. This is an ongoing project that includes experiments with advanced instrumentation to examine a large spectrum of ex-vessel melt compositions and conditions that will expand code capabilities for applications to nuclear power plants. In parallel, analytical work is carried out to bring the code capabilities to allow reactor case analysis.

Thermal-hydraulics, Hydrogen, Aerosols, Iodine (THAI) Project (2007-2009)

The objectives of the THAI project were to carry out a series of experiments in the THAI facility operated by Becker Technologies, Germany, in order to address open questions concerning the behaviour of: a) hydrogen, b) iodine and c) aerosols, in a severe accident situation. The understanding of the respective processes is essential for evaluating the challenge posed on containment integrity (hydrogen) and for evaluating the amount of airborne radioactivity during accidents with core damage (iodine and aerosols). The dominating processes are thermal-hydraulic; therefore, the THAI project is mentioned in Chapter 3 as well. The programme generated valuable data for evaluating the spatial distribution of hydrogen in the containment, its effective removal by means of equipment such as passive autocatalytic recombiners (PAR), and slow hydrogen combustion. Concerning fission products the programme focused on iodine and aerosol interaction (PAR) with a quick look on wash down from walls through condensate film flow and impact of the iodine/ozone/aerosol interaction on iodine volatility. An extensive analytical effort accompanied the experimental programme, mainly consisting of code calculations for pre-test assessments, result evaluations and extrapolation to reactor situations. The database developed in frame of this project has been and will be used to validate lumped parameter and CFD tools available and under development for containment analysis. The investigations have filled gaps to a large extent.

Thermal-hydraulics, Hydrogen, Aerosols, Iodine-Phase 2 (THAI-2) Project (2011-2014)

The remaining open questions regarding hydrogen combustion under spray and low oxygen PAR behaviour are the topics to be addressed in the follow-up Project THAI-2 (2011-2014) started in summer 2011. Concerning fission products the programme will focus on iodine release from a flashing jet and gaseous iodine deposition on aerosols. This project will also address atmospheric flows and subsequently graphite dust transport in a generic multi-compartment geometry of a graphite moderated reactor.

Source Term Evaluation and Mitigation (STEM) Project (2011-2015)

Past, recent and ongoing R&D programmes (e.g. Phebus FP, ISTP, ARTIST, OECD/BIP and THAI Projects…) are mainly focused on the reduction of uncertainties on the evaluation of the potential source term to the environment in case of a LWR severe accident. However, it has been recognised that known phenomenological uncertainties remain for some complex phenomena with a significant impact on source term. Therefore a new OECD/NEA Project named STEM (Source Term Evaluation and Mitigation), operated by IRSN in its facilities in Cadarache, France, was initiated in 2011 to improve the general evaluation of the source term. Reduction of known phenomenological uncertainties on certain phenomena is expected providing better information and tools to emergency teams in order to help them making a more robust diagnosis and prognosis of the progression of an accident and a better evaluation of potential release of radioactive materials. It is also expected helping the investigation phenomena involved in possible complementary mitigation measures, natural or engineered, so as to minimise releases to the environment.

5.5 Summary and conclusions

Severe accidents have the potential to cause large releases of radioactive material and thus pose risk to the public health and safety and damage the environment. The processes involved in the progression of severe accidents are very complex requiring experimental data to support development of models to determine design and procedural requirements to prevent and/or mitigate the consequences of such accidents.

Severe accident experimental research, as typified by the OECD/NEA sponsored work discussed above, is expensive because the facilities must operate under challenging conditions. In addition, facilities that use radioactive materials are especially expensive because of material control requirements. The expense has fostered the international co-operation because each country with interests in the area has recognised that it cannot afford to provide the necessary funding solely from its own budget.

It is common practice that analytical activities dealing with data prediction and interpretation, model development and computer code validation are performed by some or all of project participants in parallel with those of the project. These analyses constitute a very valuable complement and an additional benefit of the NEA safety projects. The OECD/NEA platform brings together the world's leading experts who contribute to maintaining and improving expertise and tools in OECD member countries, to enhancing technical exchange among specialists, and to promoting consensus building on approaches to resolve complex severe accident safety issues. Thus, these programmes make a very significant contribution to knowledge management.

The proceedings from the Workshop on Evaluation of Uncertainties in Relation to Severe Accidents and Level-2 Probabilistic Safety Analysis [28] have identified the remaining uncertainties in the severe accident phenomena and with increased interest in power uprates, higher burn-up levels and license extensions, OECD/NEA can continue to provide the platform for international co-operation in addressing these remaining issues for existing reactors. For the new future designs, the model for generating scientific base may require the industry to work more closely up front to agree on the required research. In view of the international nature of the new designs, it is important that NEA be engaged early so full benefit can be attained through international co-operation and the use of expertise available within CSNI in particular.

MAIN BENEFITS FROM 30 YEARS OF JOINT PROJECTS IN NUCLEAR SAFETY, ISBN 978-92-64-99171-2, © OECD 2012

6. Working together

Joint projects have become a core activity in NEA's overall effort to foster co-operation among member countries for the benefit of safety. The success of the joint projects can be attributed to the added value that results from the various aspects of working together on a joint project.

6.1 Sharing experimental infrastructure

Experimental facilities and programmes have played an important role in safety research from the beginning and have contributed substantially to the resolution of various safety issues. Most of the joint projects consist of an experimental programme in an existing facility. These facilities were originally built to solve a specific safety relevant issue or a cluster of related safety relevant topics, e.g.:

- LOFT for behaviour of a nuclear core under LOCA and beyond DBA conditions;
- PKL for combined ECC injection vs. cold leg injection, etc.

Constructing the facility and running the first experimental campaign was typically done in a national research programme, sometimes involving industry or international partners, e.g. PANDA for testing passive components for advanced BWRs.

The motivation for a facility operator to offer a programme in a running facility to the international community is primarily cost-sharing for follow-up programmes. It helps the operating country to maintain the facility for future investigations of anticipated safety issues.

The most obvious benefit for both the country offering a programme and the countries joining is cost saving by putting resources together. Not all countries can afford large scale experiments and prefer to participate in a common programme. The countries with a large research infrastructure are careful to avoid duplication of effort but look for sharing the load with partners.

Countries joining a joint project profit from receiving data that support their independent safety assessment without having the burden of financing the construction and maintenance of an own facility. Flexibility of the programme of investigation is an attractive feature. Participating countries are not confronted with a fixed programme offered by the operating agent but are explicitly encouraged to modify and enrich the programme with the aim to cover the issue in the best possible scope. This influence is not limited to the initiation phase but persists during the course of the project because first results may suggest changes for the remaining duration. In some projects, the programme foresees deliberately one or two tests at the end that are to be defined by the participants while the project is running.

Infrastructure includes not only the facility itself but the experienced team operating the facility, preparing and executing the tests, installing and calibrating the instrumentation, and managing the acquisition, storage and qualification of data. Experimental know-how is an important part of safety research infrastructure as a whole and equally important as code development and assessment.

6.2 Procedure for establishing and conducting a project

The Halden Project had a pilot function for establishing and managing a successful international project in rector safety. With the number of projects that followed, good practice was established. Following a recommendation from the CSNI Bureau, a document on procedures was issued in 2002 defining general guidelines for initiating, financing and managing projects in the future [29]. Leaving a large degree of flexibility to the individual project, these guidelines have turned out helpful for any country proposing a project and have led to a standard form of agreement.

The initiative for a new project is normally taken by an organisation operating a facility or managing a national research project. This organisation will in most cases become the operating agent (OA) if the proposal leads to a joint project. The member country then approaches CSNI with a first proposal. If endorsed by CSNI, an expert meeting is called for to collect technical comments and to have an impression on member countries interest. The procedure foresees a review of the proposal by the pertinent CSNI working group and, if requested by CSNI, also by CSNI's Programme Review Group. The next step is draft agreement to be worked out by the Secretariat with a detailed technical annex provided by the lead organisation. Member countries are then officially solicited to express their wish to participate.

Financing is established by sharing the cost among participants with the host country typically bearing 50% of the actual programme cost. It is a critical step to bring together enough support from member countries to launch a successful programme. In view of budget constraints in all countries convincing arguments are needed that the objectives of the proposed programme are addressing a common safety issue that is better investigated together and that the test facility involved is the most appropriate infrastructure also in view of emerging other issues in the future. The participating countries have comprehensive access to all results. Their commitment is clearly limited in terms of duration and financial contribution.

While NEA plays an essential role in the initiation phase, the success of the project is entirely in the hands of the participants once the project is running. NEA keeps helping the project throughout its lifetime as a facilitator giving administrative and technical support. Responsibility, however, is with the participants, not with the OA alone.

Progress of the project is frequently monitored by a management board (MB) that may take decisions of necessary adaptation of the research programme and the allocation of funds. The MB is supported by a Programme Review Group (PRG) giving technical advice. These PRGs have turned out to be an effective means of communication with the OA, giving suggestions for the running programmes and the ways of presenting the results in the best useable format for the member countries.

Analytical activities accompanying the projects have become good practice. Participating countries have access to the data from the project and may interpret and use them to their discretion. There is an added value, however, in the common interpretation and application of the experimental findings. The OA, although not formally obliged, should provide a first interpretation of the observed phenomena and the recorded data. Since many projects state the creation of an extended database for code validation as one of their major objectives, pre-test and post-test calculations with the codes normally applied for safety analysis should be a logical consequence of participating in a project. Doing this in a timely and joint activity is of benefit to all participants.

6.3 Identifying priorities

Systematic efforts were taken periodically by CSNI in the past to review the existing infrastructure for safety research in member countries and to match them with the needs.

Senior Groups of Experts on Safety Research (SESAR) were charged with identifying research priorities and issuing recommendations for CSNI activities (see Chapter 2). The latest report in this series was compiled by SESAR/SFEAR [16]. Based on identifying and ranking of topics in different safety relevant areas and referencing important experimental facilities able to address one or more of the high ranked topics the group issued short-term recommendations on facilities that should start a programme in short term. Moreover, the report recommended several other important facilities that should be monitored in medium term.

Since then, monitoring of experimental infrastructure is left to the working groups. Efforts are underway to increase their involvement in the initiation of projects and in the evaluation of their results. A step in this direction was taken in several projects where the participants made selected data available for a wider community for the purpose of an analytical exercise in form of a benchmark or an ISP. Moreover, CSNI strives to "acquire ownership" of projects by publishing the final reports of recent projects as CSNI reports and encourages immediate release of partial project data in support to new ISP.

For the future, project proposals and the definition of experimental programmes should explicitly reference the technical goals that have been identified in the CSNI or CNRA operating plans in response to the main challenges of the joint strategic plan [20].

6.4 Building consensus

Experimental programmes are usually established to provide data for resolving a specific safety issue, e.g. PKL for boron dilution and SCIP for fuel cladding behaviour.

In other cases the investigations should provide a general database to assess simulation capabilities, specifically for the new CFD codes, e.g. PANDA for 3D-gas flows in large connected volumes and PRISME for propagation of smoke and fire in buildings.

Sometimes, the experiments do not just confirm existing assumptions and add to the validation database but lead to the discovery of new phenomena. As an example, RASPLAV found the influence of iron on the convection of molten corium in the vessel lower head.

A common database is of high value. It has turned out that co-operation can even go a step further. The common recognition that the data are relevant for the scope of phenomena and safety topic under consideration is an additional value and a necessary step for building consensus.

In order to reach consensus among all member countries on a specific issue addressed by a joint project the results should be available to all of them. The agreement of each joint project states that the results are proprietary to the participating countries. However, according to the general OECD rule of releasing reports, they become publicly available to all member countries at the latest three years after the end of each project.

Analytical activities for interpretation of the experimental results and code validation exercises keep participants together and lead to a further level of agreement on results and of common technical understanding. Finally, the findings from a joint project might be reflected in a state-of-the-art report covering a wider area of safety topics or even form the basis of a technical opinion paper.

6.5 Recognising the benefit of joint projects

Joint projects have generated safety relevant programmes that would have never happened if the individual countries were left alone with maintaining and operating a large facility.

In terms of resolving a safety issue “for good”, joint projects are overcharged. For most safety issues, experimental research is only part of the solution, often an essential part, but seldom the final word.

The timescale of joint projects including preparation, contract building, programme execution, data treatment, and reporting takes typically several years. Therefore, joint projects are mostly useful for dealing with mid-term or long-term issues.

A new generation of experts has taken over. New countries come into the nuclear arena. Joint projects have a large capability for know-how transfer and for building expert capacity.

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7. Summary and conclusions

The OECD Nuclear Energy Agency (NEA), through its committees, has provided an excellent platform for the member countries to discuss common issues of safety concern, been a part of centres of excellence in many complex areas, developed joint projects to gather critical information through experimentation and analysis, provided state-of-the-art reports and thus contributed to knowledge generation, retention and transfer. Design, construction and operation of experimental facilities is very expensive and the NEA platform provides an opportunity for many countries to share the costs associated with generating data and information to resolve safety issues.

The NEA platform brings together the world's leading experts who contribute to maintaining and improving expertise and tools in NEA member countries and beyond, to enhancing technical exchange among specialists, and to promoting consensus building on approaches to resolve complex safety issues. Thus, these programmes also make a very significant contribution to knowledge management. Many important joint experimental programmes to address safety issues have been conducted and others are ongoing under NEA auspices.

It is a common practice among some or all joint project participants to conduct analytical activities dealing with data prediction and interpretation, model development and computer code validation in parallel with the conduct of the project. These analyses constitute a very valuable complement and an additional benefit of the NEA joint projects.

Thermal-hydraulic issues dominated safety concerns and rulemaking from the early days of nuclear power plant operation. The focus shifted from large break LOCA to more frequent initiating events. Classic issues connected with loss-of-coolant accidents have been resolved by analytical and experimental programmes, many of them in international co-operation. Validation matrices, international standard problems and results from NEA joint projects have largely contributed to a consensus on the technical basis for closing typical thermal-hydraulic issues. Operating experience, lifetime management and lessons learnt from events may pose new questions requiring safety research also in the thermal-hydraulic area. Issues that may arise for new reactor designs are on the agenda of the pertinent CSNI working groups.

Computer code modelling has advanced from conservative assumptions to a best-estimate approach, complemented by methods for quantifying uncertainties. CFD codes are fast entering the nuclear industry. Development and validation for two-phase flows is in progress. The growing CFD application for safety demonstration has to be accompanied by commonly accepted best practices.

Maintaining fuel and cladding integrity during transients and accidents is a fundamental safety concern. Therefore, failure mechanisms ought to be thoroughly investigated and understood. Power uprates, load following, longer cycles and higher discharge burn-up mean much more demanding fuel operating conditions. Care has to be taken so that sufficient safety margins are maintained.

NEA joint projects have addressed various fuel issues of common interest, all of them of high safety significance. The CABRI reactor offers an almost unique opportunity for in-pile testing of high burn-up fuel under realistic PWR conditions. Hot cells are another important infrastructure for fuel safety research. The Studsvik Cladding Integrity Project has benefited from the facilities there and the associated analytical capabilities. The Paks incident was recognised as an opportunity to gain insight into the behaviour of a large amount of real fuel under degraded cooling conditions, to observe beyond design basis phenomena and the capabilities of computer codes to predict them. The ongoing Sandia Fuel Project acknowledges the fact that transients and accidents may challenge fuel integrity not only in the reactor core but in the spent fuel pool as well.

Severe accidents have the potential to cause large releases of radioactive material and thus pose a risk to public health and safety as well as the environment. The processes involved in the progression of severe accidents are very complex, requiring experimental data to support the development of models to determine design and procedural requirements to prevent and/or mitigate the consequences of such accidents.

Severe accident experimental research, as typified by NEA-sponsored work, is expensive because the facilities must operate under challenging conditions. In addition, facilities that use radioactive materials are especially expensive because of material control requirements. The high expense of such facilities has fostered international co-operation because each country with interests in the area has recognised that it cannot afford to provide the necessary funding solely from its own budget. Many elements of severe accident safety issues have been addressed and useful data gathered to support model development and to provide additional important information for member countries to develop their accident management strategies.

The proceedings of the 2005 CSNI workshop on severe accidents [28] have identified the remaining uncertainties in severe accident phenomena, and with increased interest in power uprates, higher burn-up levels and license extensions, the NEA can continue to provide a platform for international co-operation on addressing these remaining issues for existing reactors. For new future designs, the model for generating the scientific base may require the safety authorities and industry to work more closely up front to define the required research. In view of the international nature of the new designs, it is important that the NEA be engaged early so that full benefit can be achieved through international co-operation and the use of expertise available within the CSNI in particular.

In summary, CSNI working group activities and NEA joint projects are responding to the challenges of power uprates, higher burn-up, new fuel element designs, new cladding materials and the development of models to analyse accidents, including severe accidents. Thorough understanding of phenomena and failure mechanisms associated with accidents as well as common assessment of experimental data and computer code models strengthen the technical bases for safety decisions.

It is important that, consistent with NEA practice, all new programmes should carefully consider the lessons learnt from recent operating experience including reported events and accidents such as the one at Fukushima.

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Appendix 1:
Short description of OECD/NEA nuclear safety joint projects
(status in December 2011)

Thermal-hydraulics projects

TH-1: LOFT, Loss-of-Fluid Test Project (1983-1989)

TH-2: BUBCON, Bubbler Condenser Project (2001-2002)

TH-3: SETH, SESAR Thermal-hydraulics Project (2001-2006)

TH-4: SETH-2, SETH Phase 2 Project (2007-2010)

TH-5: PSB-VVER Project (2003-2008)

TH-6: PKL, *Primärkreislauf* Project (2004-2007)

TH-7: PKL-2, PKL Phase 2 Project (2008-2011)

TH-8: ROSA, Rig of Safety Assessment Project (2005-2009)

TH-9: ROSA-2, ROSA Phase 2 Project (2009-2012)

TH-10: PRISME, Fire Propagation in Elementary, Multi-room Scenarios Project (2006-2011)

TH-11: PRISME-2 Project (2011-2016)

TH-12: LOFC, Loss of Forced Coolant Project (2011-2013)

Fuel behaviour projects

FU-1: HALDEN, Halden Reactor Project (1958-present)

FU-2: CIP, Cabri Water Loop International Project (2000-2015)

FU-3: PAKS, OECD-IAEA Paks Fuel Project (2004-2007)

FU-4: SCIP, Studsvik Cladding Integrity Project (2004-2009)

FU-5: SCIP-2, SCIP Phase 2 (2009-2014)

Severe accident projects

SA-1: TMI-VIP, TMI-2 Vessel Investigation Project (1988-1993)

SA-2: RASPLAV Project (1994-2000)

SA-3: OLHF, OECD Sandia Lower Head Failure Project (1998-2002)

SA-4: MASCA, Material Scaling Project (2000-2003)

SA-5: MASCA-2, MASCA Phase 2 Project (2003-2007)

SA-6: MCCI, Melt Coolability and Concrete Interaction Project (2002-2005)

SA-7: MCCI-2, MCCI Phase 2 Project (2006-2010)

SA-8: BIP, Behaviour of Iodine Project (2007-2011)

SA-9: BIP-2, Behaviour of Iodine-Phase 2 Project (2011-2014)

SA-10: SERENA, Steam Explosion Resolution for Nuclear Applications Project (2007-2012)

SA-11: THAI, Thermal-hydraulics, Hydrogen, Aerosols, Iodine Project (2007-2009)

SA-12: THAI-2, Thermal-hydraulics, Hydrogen, Aerosols, Iodine-Phase 2 Project (2011-2014)

SA-13: SFP, Sandia Fuel Project (2008-2012)

SA-14: STEM, Source Term Evaluation and Mitigation Project (2011-2015)

Other joint projects (systems and event record databases)

SY-1: SCORPIO Project (1996-1998)

SY-2: PLASMA Project (1998-2000)

EDB-1: International Common-cause Failure Data Exchange (ICDE) Project (1994-2014)

EDB-2: Piping Failure Data Exchange (OPDE) Project (2002-2011)

EDB-3: Fire Incidents Records Exchange (FIRE) Project (2002-2014)

EDB-4: Computer-based Systems Important to Safety (COMPSIS) Project (2006-2011)

EDB-5: Stress Corrosion Cracking and Cable Ageing Project (SCAP) (2006-2010)

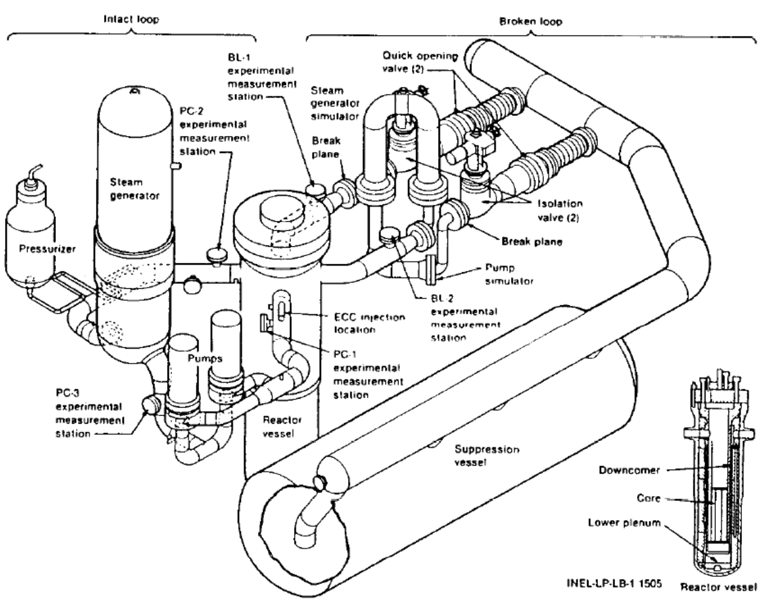
EDB-6: Component Operational Experience, Degradation and Ageing Programme (CODAP) (2011-2014)

EDB-7: Cable Ageing Data and Knowledge (CADAK) Project (2011-2014)

TH-1: Loss-of-Fluid Test (LOFT) Project

The Loss-of-Fluid Test (LOFT) Research Programme was originally set up by the US Nuclear Regulatory Commission. Part of the programme was later broadened into an international collaboration project under the aegis of the OECD Nuclear Energy Agency (NEA). This initial programme addressed several configurations of loss-of-coolant accident (LOCA) with large break tests and intermediate break tests carried out between 1978 and 1982. A new programme, subject of the OECD/NEA LOFT Project, was designed to use the LOFT experimental nuclear test facility at the Idaho National Engineering Laboratory (INEL) in a programme of safety experiments. The initial proposal was developed from an initiative of the United States Department of Energy, which also provided continuing management support. The project successfully combined the abilities and objectives of an international team with those of the reactor operation and analysis staff at INEL to provide a significant addition both to the international database of large scale experimental data on reactor safety and to the analysis and understanding of the test results.

The LOFT facility, Idaho, United States



Source: OECD/NEA LOFT Project.

The experimental programme of the OECD/NEA LOFT Project comprised eight experiments, six thermal-hydraulic experiments, and two fission product release experiments as listed hereafter.

- LP-FW-1 – Loss-of-feedwater, primary feed and bleed recovery procedure (20/02/1983).
- LP-SB-1 – Hot leg SB LOCA, early pump trip (23/06/1983).
- LP-SB-2 – Hot leg SB LOCA, delayed pump trip (14/07/1983).
- LP-SB-3 – Cold leg SB LOCA, core uncovery, secondary feed and bleed recovery procedure, accumulator injection at low-pressure differential (05/03/1984).
- LP-02-6 – 200% large-break LOCA, US licensing case (03/10/1983).

- LP-LB- 1 – 200% large-break LOCA UK, licensing case (03/02/1984).
- LP-FP-1 – Gap fission product release, large-break LOCA, German licensing case (19/12/1984).
- LP-FI-2 – Fission product release at high fuel temperatures (above 2 100 K), V-sequence (03/07/1985).

Project members undoubtedly found it valuable to obtain access to the advanced technology being developed within the project by seconding staff from whose expertise the project itself also benefited. The community of interest that this and the common programme of computer analysis also encouraged was an important contribution to the success of the project. In summary, the project has achieved the following:

1. It successfully ran an international project in which both managerial decisions and the detailed planning of the programme were organised by the collective decisions of members. The lessons learnt from this aspect of the project should be of permanent value for future international initiatives.
2. The detailed experimental results provided valuable new evidence on thermal-hydraulic issues and an important international database for computer code verification.
3. It provided a valuable forum for the exchange of specialist views and for computer code comparisons.
4. The two tests, LP-FP-1 and LP-FP-2, extended the use of LOFT to provide data on fission product release and transport from failed fuel. LP-FP-2 is also a major data source on severe core damage phenomena in a large fuel bundle and work on the assessment of data from this test can be expected to continue over a number of years.
5. Effective measures were taken to make the data of long-term value by archiving in the NEA Data Bank, by making the data available as part of the NEA Committee on the Safety of Nuclear Installations (CSNI) Code Validation Matrix, and by linking further work based on the LOFT data with current international programmes such as the CSNI specialist committees, the ICAP Programme, and the USNRC Severe Fuel Damage Programme.
6. There is general agreement that there are problems in retaining facilities and expertise in a number of areas of reactor safety and that the facilities offered by LOFT are irreplaceable. The OECD/NEA LOFT programme was able to successfully make use of LOFT but was not able to provide a route for its further retention.

Final summary report

"An Account of the OECD LOFT Project", J. Fell, S.M. Modro, OECD LOFT-T-3907, May 1990.

Participants

Austria, Finland, Germany, Italy, Japan, Spain, Sweden, Switzerland, United Kingdom, United States.

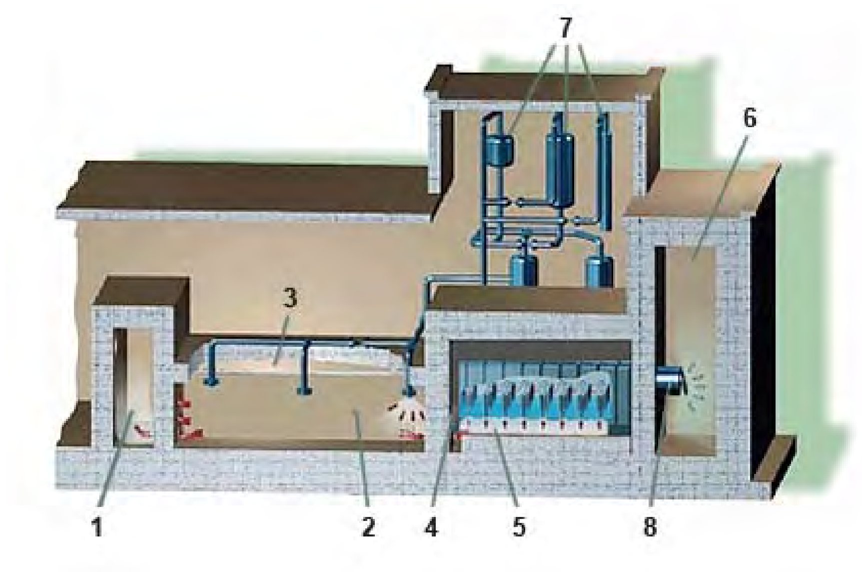
Project period: January 1983-December 1989.

Budget: USD ~100.0 million.

TH-2: Bubble Condenser (BUBCON) Project

When the Bubbler Condenser Project (BUBCON) started, there were 14 nuclear power plants of the VVER-440/213 type operating in the Czech Republic, Hungary, the Slovak Republic, Russia and Ukraine. VVER-440/213 pressurised water reactors have a pressure suppression containment structure called a "bubbler condenser" (BC) tower which can reduce the design pressure of the entire confinement volume following a design-basis accident (DBA), such as a loss-of-coolant accident (LOCA). The bubbler condenser pressure suppression system reduces the LOCA confinement pressure by expanding the released steam to a large volume and allowing it to condense in a water pool. The thermodynamic principle of the bubbler condenser function is identical to the function of the containments in western boiling water reactors with pressure suppression systems, but the boundary conditions (construction and dimensions) are different.

The bubble condenser concept



Source: OECD/NEA BUBCON Project.

This project was set up to provide answers to remaining questions about the BC tower's performance under accident conditions by means of experiments carried out at the Electrogorsk Research Centre (EREC) in Russia. Experts from the Czech Republic, France, Germany, Hungary, the Slovak Republic and the United States participated in the project; the European Union also took part. Czech, Hungarian and Slovak utilities provided the financing for the test programme. Project meetings held in 2001 and 2002 addressed the current status of research on the subject, agreed on three experiments to be carried out under the project's auspices and oversaw the co-ordination of these experiments with other initiatives, notably those taken by the European Union.

The results have been analysed and an activity report for the project is available. Based on the experimental and analytical evidences of the newly performed investigations and in-depth discussions, the OECD Bubbler-condenser Steering Group has concluded that the related loads during DBAs do not represent a challenge to the bubbler condenser's confinement integrity. Regarding more specific aspect the group has stated the following:

A1. The verification of the blow-down mass and energy rates (MER) producing loads to the BC was performed in the frame of the present project for both the previous LBLOCA tests and for the recent tests. Results confirmed conservative mass and energy estimations. It was shown that the injected MER were higher in the tests than the scaled MER (NPP/100) values. These findings confirmed the conservative nature of the approach. Conclusions were that the related loads do not represent a challenge to containment integrity. A parallel assessment of this issue with respect to LBLOCA tests is going on in the European PHARE project PR/TS/17.

A2. The first part of the question concerning conservatism (initial conditions, scenarios, test conditions, different break locations, etc.) can be answered positively. The adequacy was addressed by the scaling of the facility and possible distortions were compensated by different measures (e.g. installation of additional insulation).

A3. Non-uniformities of flow rates and water temperatures have been observed in the experiments. An appropriate understanding of the non-uniformities was obtained by detailed (3D) code calculations. The reasons and the nature of the distributions have been satisfactorily explained by code calculations.

Final report

"Answers to Remaining Questions on Bubbler Condenser — Activity Report of the OECD/NEA Bubbler Condenser Steering Group", Report NEA/CSNI/R(2003)12, April 2003.

Participants

Czech Republic, France, Germany, Hungary, Russian Federation, Slovak Republic, Ukraine, United States.

Project period: 2001-2002.

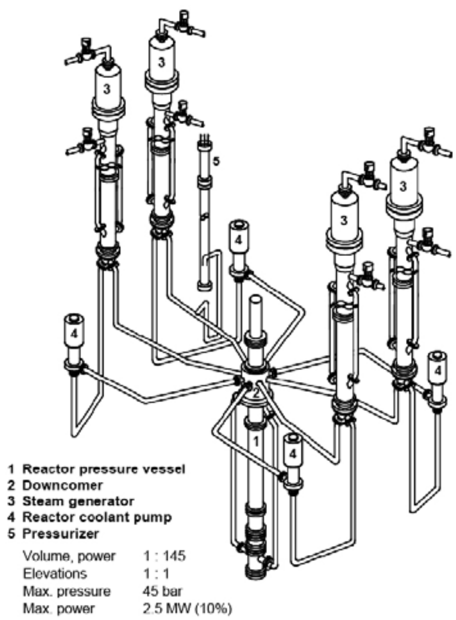
Budget: Voluntary contributions.

TH-3: SESAR Thermal-hydraulics (SETH) Project

The NEA Committee on the Safety of Nuclear Installations (CSNI) has been promoting international initiatives and collaboration to establish convincing experimental safety programmes around specific facilities. As a first step in this direction, the PKL (pressurised water reactor) loop in Germany and the PANDA in Switzerland were requested to develop proposals consistent with the priorities indicated in the CSNI Senior Group of Experts on Safety Research (SESAR) report "Nuclear Safety Research in OECD Countries: Summary Report of Major Facilities and Programmes at Risk". Their proposals, after discussions with member country experts, formed the basis for the SETH Project. The SESAR Thermal-hydraulics (SETH) Project covered two aspects of accident management:

- Countermeasures for two types of pressurised water reactor (PWR) accidents – carried out at AREVA NP's *Primär Kreislauf* (PKL) establishment in Erlangen, Germany.
- Gas flow distributions relevant to in-reactor containments (with focus on simulated hydrogen distribution) – carried out at the (PSI) PANDA establishment near Zurich, Switzerland.

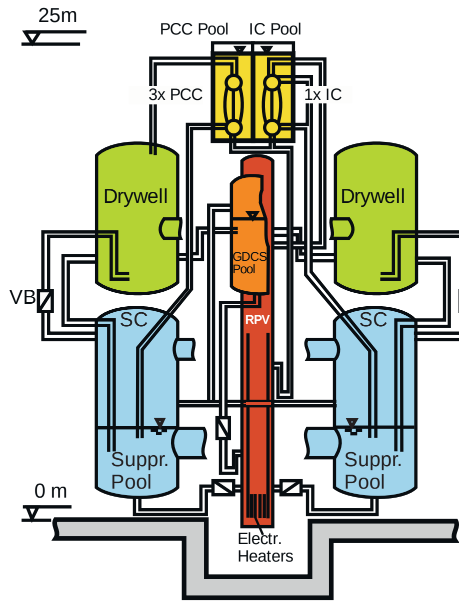
The PKL loop, Erlangen, Germany



Source: OECD/NEA SETH Project.

The PKL tests investigated two PWR safety issues: boron dilution in loss-of-coolant accidents and boron dilution during mid-loop operation (shutdown conditions). The first category of tests aimed to verify if the necessary conditions can arise for core reactivity insertion due to boron dilution during a small-break loss-of-coolant-accident with a natural circulation restart. Results received indicate that while boron dilution can occur under specific circumstances, safety is not impaired. The second test series assessed primary circuit accident management operations to prevent boron dilution as a consequence of loss of heat removal in mid-loop operation conditions. A final report on the results of the PKL tests was issued in 2004. In the PKL Programme, it was found that the maximum size of a condensate slug is limited by the volume of the loop seal plus part of the SG outlet pipe. When natural circulation restarts, pronounced condensate slugs arrive at the reactor pressure vessel only from the loops without ECC injection. One test was devoted to the loss of the residual heat removal system under shutdown conditions with water level in the loops reduced to ¾ height and a SG in stand-by to remove decay heat. The test provided insight into the thermal-hydraulic processes leading to re-establishing the heat sink. The PKL Test Programme provided a database for validation of system codes for conditions relevant for boron dilution scenarios.

The PANDA facility, Villigen, Switzerland



Source: OECD/NEA SESAR Project.

The (PSI) PANDA experiments aimed to provide data on containment three-dimensional gas flow and distribution issues that are important for code prediction capability improvements, accident management and the design of mitigating measures. These experiments were conducted on a large scale in multi-compartment geometries in order to provide data suitable for the improvement and validation of safety analysis codes. After an extensive preparation phase, the experimental series started in 2004 and continued in 2005. Due to the complexity of the PANDA experiments, some delays were encountered. The project board therefore decided to extend the programme's time frame to the end of 2006. The data obtained in the PANDA part of the project are of high resolution and sufficient accuracy to be used for verification, validation and further development of 3D computational tools and ad hoc models in lumped parameter codes, as they are used for containment safety analysis. Enhanced codes will provide a more reliable evaluation of the containment response to severe accidents and could support planning of accident management measures.

Final reports

"OECD/SETH Project – Final Report of the PKL Experimental Programme", T. Mull, B. Schoen, K. Umminger, FANP – NGTT1/04/en/04, December 2004.

"OECD/SETH Project – PANDA Experiments: Large-scale Experimental Investigation of Gas Mixing and Stratification in LWR Containment – Final Report", F. de Cachard, D. Paladino, R. Zboray, M. Andreani, PSI, April 2007.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, Republic of Korea, Spain, Sweden, Switzerland, Turkey, United Kingdom, United States.

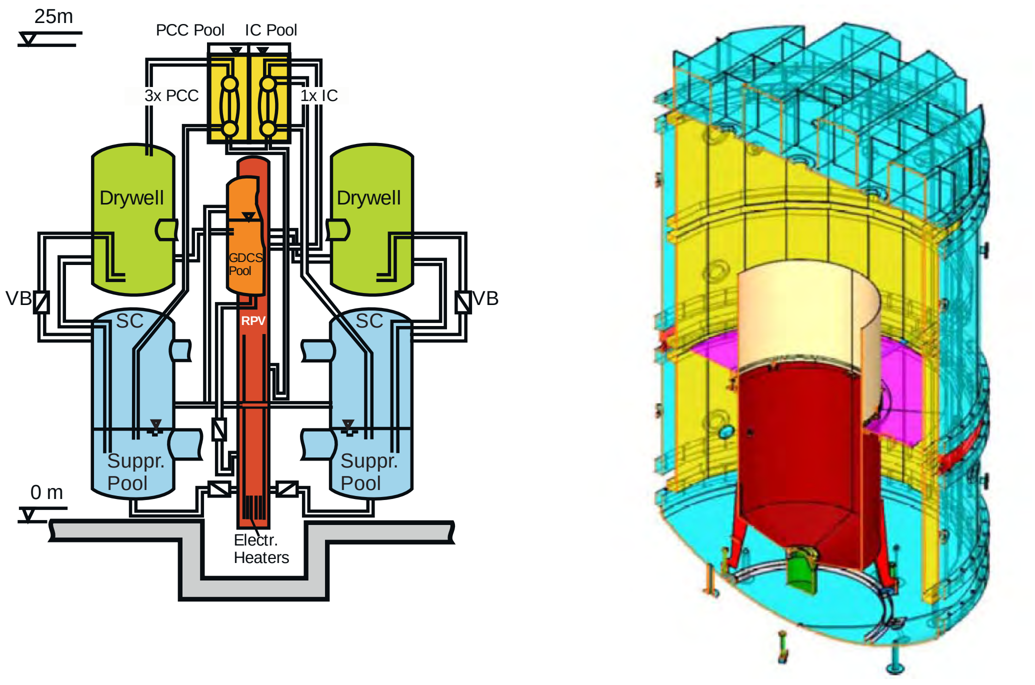
Project period: SETH/PKL: April 2001-June 2005 – SETH/PANDA: April 2001-December 2006.

Budget: USD 4.7 million.

TH-4: SESAR Thermal-hydraulics-Phase 2 (SETH-2) Project

The SESAR Thermal-hydraulics (SETH) Project began in 2001. 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. A follow-up project, SETH-2, has now been launched with the aim of resolving key computational issues for the simulation of thermal-hydraulic conditions in reactor containments. To this end, the SETH-2 Project will make use of the Paul Scherrer Institute (PSI) PANDA and Commissariat à l'énergie atomique (CEA) MISTRA facilities.

The PANDA facility, Villigen, Switzerland/the MISTRA facility, Saclay, France



Source: OECD/NEA SETH-2 Project.

The aim of the project was to generate high quality experimental data that will be used for improving the modelling and validation of computational fluid dynamics (CFD) and lumped parameter (LP) computer codes designed to predict post-accident containment thermal-hydraulic conditions. The unique and complementary instrumentation used in the PANDA and MISTRA facilities and the different size and configuration of the two installations will contribute to address a variety of measured parameters, configurations and scale, thus enhancing the value of the data for code applications. Relevant containment phenomena and separate effects will be studied in the programme, including the effects of jets, natural convection, the effect of containment coolers, the effect of sprays, etc. The experiments will not only support CFD and LP code validation for current operating reactors but also simulate conditions specific to advanced reactor designs.

Al test were completed by November 2010. In parallel, analytical activities were conducted by participants for the benchmarking and qualification of their computer codes. A concluding seminar was convened in September 2011.

Final report

The final report of SETH-2 Project.

Participants

Czech Republic (NRI), Finland (VTT), France (CEA, EDF, IRSN), Germany (GRS), Japan (JNES), Republic of Korea (KAERI), Sweden (SSM), Institute, Switzerland (PSI), Slovenia (JSI).

Project period: 2007-2010.

Budget: USD 3.2 million.

TH-5: PSB-VVER Project

The PSB-VVER Project was designed to provide the experimental data needed to allow full validation of the computer codes used in the thermal-hydraulic analysis of VVER-1000 reactors. At the time, there were 19 VVER-1000 reactors in operation and several more under construction, mostly in central and eastern Europe. These Russian-designed reactors have design characteristics that are similar to western-designed pressurised water reactors (PWRs). There are, however, important design differences which necessitate specific experimental facilities and data to ensure that the computer codes are validated and that the results of the thermal-hydraulic analyses are of acceptable quality for this type of reactor. To this end, experiments were carried out in a large-scale, thermal-hydraulics facility located at the Electrogorsk Research and Engineering Centre in Russia.

The PSB facility, Electrogorsk, Russian Federation

Source: OECD/NEA PSB-VVER Project.

Earlier NEA work established a "code validation matrix" for both LWRs and VVERs. This matrix is essentially a set of phenomena and experimental data that is used to validate thermal-hydraulic computer codes. This led to the general conclusion that the VVER-1000 matrix was not complete. Consequently, it was decided to develop an OECD project to obtain the required experimental data not covered by the matrix. The project's objective was to provide the unique experimental data needed for the validation of thermal-hydraulic codes and to support refinements to the safety assessment tools for VVER-1000 reactors.

The intended PSB-VVER experiments constituted a relevant extension of the existing code validation database. The data will be of primary relevance to the VVER-1000 system, but they are also of interest to other types of pressurised water reactors. The intended scope of the project work consisted of five PSB-VVER experiments in the following areas:

- scaling effects;
- natural circulation;

- small cold leg break loss-of-coolant accidents;
- primary to secondary leak;
- 100% double-ended cold leg break.

Extensive pre- and post-test analyses are to accompany the experimental programme throughout the entire experimental series. The possibility of setting up one or more standard problems – either limited to project participants or with broader involvement – will be considered. Three project tests have been successfully carried out and reported upon so far. The test matrix for the remaining part of the programme was discussed and revised by members. The fourth test investigated accidental conditions involving a primary to secondary leak (steam generator header rupture) and was carried out in the first half of 2005. A blind test exercise where the fourth test's outcomes were to be predicted by calculations before its execution was also organised. The final test simulating thermal-hydraulic conditions arising after a large-break LOCA in a VVER-1000 reactor, was supposed to be the first one run under these very demanding conditions. A preliminary final test (10% power) was run in January 2008; several attempts for running the final test (full power) were unsuccessful and it was decided to close the project at the end of 2008 with the agreement that the data of this final test would be provided to the project participants if it could be completed.

Final report

"Final Report of the OECD PSB-VVER Project – PSB-32", O. Melikhov, I. Elkin, S. Nikonov, A. Basov, A. Kapustin, Electrogorsk, Russia, October 2008.

Participants

Czech Republic, Finland, France, Germany, Italy, Russian Federation, United States.

Project period: February 2003-December 2008.

Budget: USD 1.25 million.

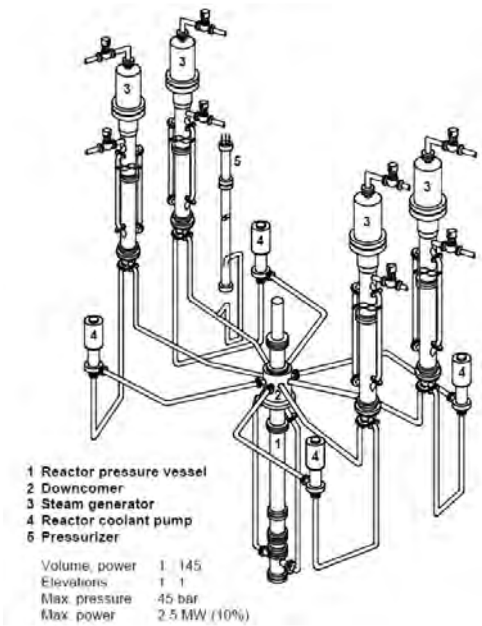
TH-6: *Primärkreislauf* (PKL) Project

The PKL Test Programme investigated pressurised water reactor (PWR) safety issues and specifically:

- boron dilution accidents; and
- loss of residual heat removal in mid-loop operation (during shutdown conditions).

These issues were investigated by means of thermal-hydraulic experiments that were conducted at the *Primärkreislauf-Versuchsanlage* (primary coolant loop test facility) PKL. This facility is owned and operated by AREVA NP and is situated in Erlangen, Germany.

The PKL facility, Erlangen, Germany



Source: OECD/NEA PLK Project.

In the first category of tests – which dealt with inherent boron dilution during a small-break loss-of-coolant accident – the key issue was whether or not there are plant thermal-hydraulic conditions that can produce reactivity insertion into the core, and thus a potential accident situation. The experiments aimed to reproduce different plant and system configurations, especially those that are believed to be potentially most serious, thus encompassing an envelope of conservative cases. Where appropriate, measures that can remedy the boron dilution issue were also investigated.

The second category of tests was to assess accident management operations of the primary circuit after a loss of residual heat removal capability during mid-loop operation. Tests with closed reactor circulation system explored safe procedures for re-establishing heat removal and avoiding significant reactivity insertion into the core. Tests with open circulation systems aimed to explore measures for preventing fuel uncoverage and to maximise safety margins provided by different ways of coolant injection.

The test matrix focused inter alia on the following items that are currently receiving great attention within the international reactor safety community:

- boron dilution events after small-break loss-of-coolant-accidents;
- loss of residual heat removal during mid-loop operation in closed reactor coolant systems, potentially also leading to boron dilution;

- loss of residual heat removal during mid-loop operation with open reactor coolant systems.

AREVA NP has for a number of years conducted valuable experiments on reactor thermal-hydraulics in the PKL facility, including earlier experiments carried out in the framework of the SETH project. Two tests were carried out in 2004 and two in 2005. A workshop covering an analytical exercise with code predictions related to the PKL tests was also conducted in 2005. The tests carried out in 2005 focused on the loss of residual heat removal in three quarter loop operation with the reactor cooling system closed and on the effect of pressuriser temperature on RCS inventory distribution. In particular, two tests were carried out to address the following scenarios:

- Run A: loss of RHR with RCS filled, initial level mid loop, pressuriser cold;
- Run B: loss of RHR with RCS filled, initial level mid loop, pressuriser hot.

The three final tests of the current programme were carried out and reported upon in 2006; the project ended with the final meeting in May 2007. A follow-up to the project, PKL-2, was conducted during the period 2008-2011.

The PKL Project has completed the understanding of the physics and thermal-hydraulics of post-SB-LOCA boron dilution, so that the issue can be considered, from an experimental point of view, to be closed.

Concerning the loss of RHRS during shutdown, the tests provided insight into and improved our understanding of natural circulation phenomena under conditions with closed or open circuit. The tests demonstrated the ability of a PWR to remove the residual heat by a SG on stand-by and to re-establish a heat sink in a shut-down plant during ¾-loop operation.

Final report

“Final Report of the OECD-PKL Project”, T. Mull, K. Umminger, A. Bucalossi, F. D'Auria, P. Monnier, I. Toth and W. Schwarz, AREVA-NP NTCTP-G/2007/en/0009, November 2007.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

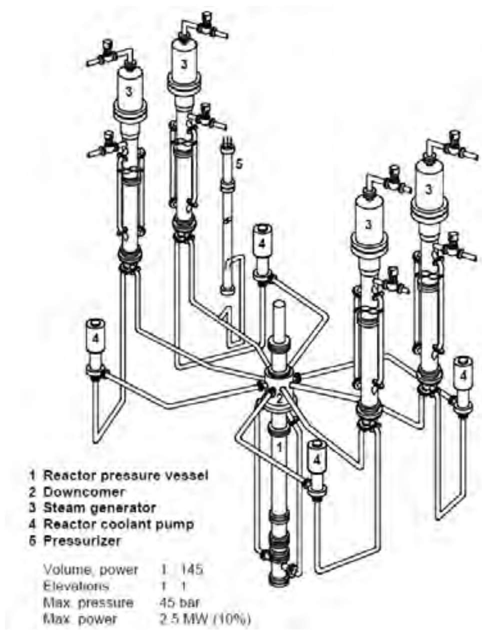
Project period: January 2004 to May 2007.

Budget: USD 1.2 million/year.

TH-7: *Primärkreislauf*-Phase 2 (PKL-2) Project

AREVA NP has for a number of years conducted valuable experiments on reactor thermal-hydraulics in the PKL facility, including earlier experiments carried out in the framework of the SETH project (2001-2003) and of the PKL-1 Project (2004-2007). The PKL-2 test programme is investigating safety issues relevant for current pressurised water reactor (PWR) plants as well as for new PWR design concepts and are focused on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations. These issues are investigated by means of thermal-hydraulic experiments conducted at the *Primärkreislauf-Versuchsanlage* (primary coolant loop test facility) PKL. This facility is owned and operated by AREVA NP and is situated in Erlangen, Germany.

The PKL facility, Erlangen, Germany



Source: OECD/NEA PLK-2 Project.

The first category included tests addressing the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water, in both vertical and horizontal steam generators. Cooldown procedures in the case where steam generators have partly dried out on the secondary side, were also covered. A further topic addressed the heat transfer in the steam generators under reflux condenser conditions (e.g. fast secondary side depressurisation). Fast cooldown transients (with water filled reactor coolant system) such as main steam line break, completed by tests on mixing of hot and cold water in the RPV downcomer and the lower plenum were also considered in the test programme. Further investigation addressed boron precipitation processes in the core following large break loss-of-coolant accidents. The last two tests were decided by the programme partners following the results of preceding experiments. The final programme included eight integral experiments at the PKL test facility covering the following topics.

- Systematic investigation of the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water (two tests). (The tests on the heat transfer mechanisms in the SGs in the presence of nitrogen are complemented by tests in the PMK test facility in Hungary for horizontal steam generators.)

- Heat transfer in steam generators that have dried out on the secondary side.
- Fast secondary side depressurisation (reflux condenser conditions) with additional system failures.
- Fast cooldown transients such as main steam line break. [The tests on fast cooldown transients are completed by tests at the ROCOM test facility (in Germany) on mixing in the RPV downcomer and the lower plenum.]
- Boron precipitation following large break loss-of-coolant accidents.
- RCS cooldown with void formation in RPV upper head.
- Counterpart Test with ROSA/LSTF Project on small break LOCA with accident management procedures.

Final report

Scheduled for 2012.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Netherlands, Spain, Sweden, Switzerland, United Kingdom, United States.

Project period

April 2008 to December 2011. A proposal is under discussion for a follow-up phase PKL-Phase 3 (2012-2015) which will include PKL, ROCOM, PMK and PACTEL (Finland) facilities.

Budget: ~EUR 1 million/year, with German parties covering 50% of this cost.

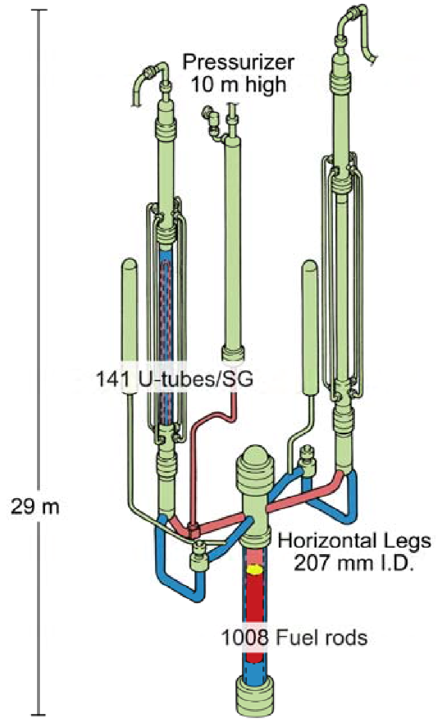
TH-8: Rig of Safety Assessment (ROSA) Project

The ROSA Project aimed at resolving issues in thermal-hydraulics analyses relevant to light water reactor (LWR) safety using the Japanese ROSA/LSTF facility. In particular, it focused on the validation of simulation models and methods for complex phenomena that may occur during design basis events (DBE) and beyond-DBE transients. The key objectives of the ROSA Project were to:

- Provide an integral and separate-effect experimental database to validate code predictive capability and accuracy of experimental thermal-hydraulic models. In particular, phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows are to be studied.
- Clarify the predictability of codes currently used for thermal-hydraulic safety analyses as well as other advanced codes presently under development, thus creating a group among the NEA member countries who share the need to maintain or improve the technical competence in thermal-hydraulics for nuclear reactor safety evaluations. The experimental programme is defined to provide a valuable and broadly usable database to achieve these objectives.

When evaluating the safety of light water reactors, computer codes are used to simulate their behaviour during design basis events and beyond-DBE transients. This involves complex multi-dimensional single-phase and two-phase flow conditions, which can include non-condensable gas in many cases. Although current thermal-hydraulic safety analysis codes have a very high predictive capability (especially for one-dimensional phenomena such as flows in piping at high flow rates) there is a need for experimental work and code development and validation for these complex flow conditions. Given the increased use of best-estimate analysis methods in licensing, which is replacing traditional conservative approaches, the validation and quantification of uncertainties in the simulation models and methods is required.

The LSTF facility, Tokai-Mura, Japan



Source: OECD/NEA ROSA Project.

Many experimental facilities have contributed to the thermal-hydraulic databases available today. However, most of current data are insufficient for future codes that incorporate multi-dimensional simulation capabilities, mainly because the spatial resolution of measurement is not enough to assess the simulation models and methods. The ROSA project addressed these issues.

The project consisted 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 later by participants [one on pressure vessel upper-head break loss-of-coolant accident (LOCA) and another on pressure vessel bottom break LOCA, combined with accident management measures with symptom-oriented operator actions].

The programme comprised a total of 12 tests. The content of each test was discussed by the project steering bodies, which defined the test initial and boundary conditions. They were conducted from 2006 until the beginning of 2009. Project members also discussed the issues to be addressed in a follow-up of the project, ROSA-2, which was started in April 2009.

For all the experiments, pre-test analyses were done to survey optimum test conditions to meet the test objectives. The obtained data have been shared among the participants and utilised for the post-test analyses by using both the BE and CFD codes.

Final report

"Final Integration Report of the OECD/NEA ROSA Project (2005-2009) – JAEA", September 2010.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

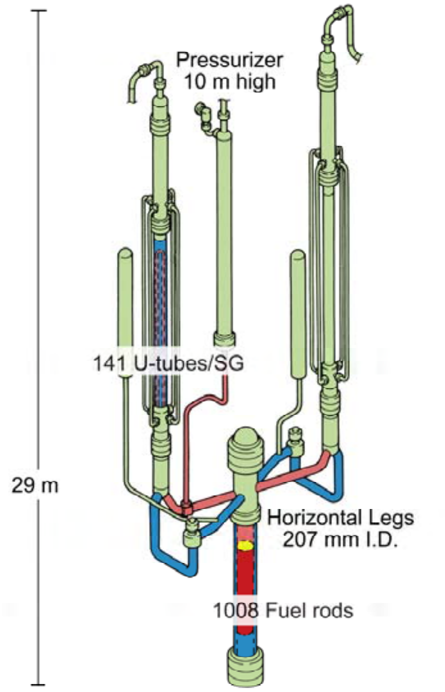
Project period: April 2005-March 2009.

Budget: USD 1 million/year.

TH-9: Rig of Safety Assessment-Phase 2 (ROSA-2) Project

The ROSA-2 Project aims to resolve key light water reactor (LWR) thermal-hydraulics safety issues highlighted from the first phase of the project, by using the ROSA/large-scale testing facility (LSTF) at the Japan Atomic Energy Agency.

The LSTF facility, Tokai-Mura, Japan



Source: OECD/NEA ROSA-2 Project.

In particular, the ROSA-2 Project is focused on the validation of simulation models and methods for the following complex phenomena of high-safety relevance for thermal-hydraulic transients in design basis events (DBE) and beyond-DBE:

- Generate system-integral and separate-effect experimental database to validate predictive capability and accuracy of computer codes and models. Thermal-hydraulic phenomena coupled with multi-dimensional flows that may include mixing, stratification, counter-current flows, parallel-channel flows and oscillatory flows will be the main focus of the investigations.
- Facilitate assessment of codes currently in use for thermal-hydraulic safety analyses as well as advanced codes presently under development including three-dimensional computational fluid dynamics (CFD) codes through active involvement of the project partners who will maintain and improve the technical competence in thermal-hydraulics for nuclear reactor safety (NRS) evaluations.

The experimental programme is intended to provide a valuable and broadly usable database to achieve the above cited objectives. The OECD/NEA ROSA-2 Project consists of six LSTF experiments that mainly include two groups of experiments which are intermediate break loss-of-coolant (LOCA) accidents and steam generator tube rupture (SGTR) accidents. One of the six experiments may be directed to a different target following a discussion with the project partners. In particular it was agreed to have a counterpart test with PKL Project on small break LOCA with accident management procedures.

The ROSA-2 Project was scheduled to run from 01 April 2009 to 31 March 2012; however after the 11 March 2011 earthquake duration was extended until 31 October 2012.

Final report

Scheduled for 2013.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

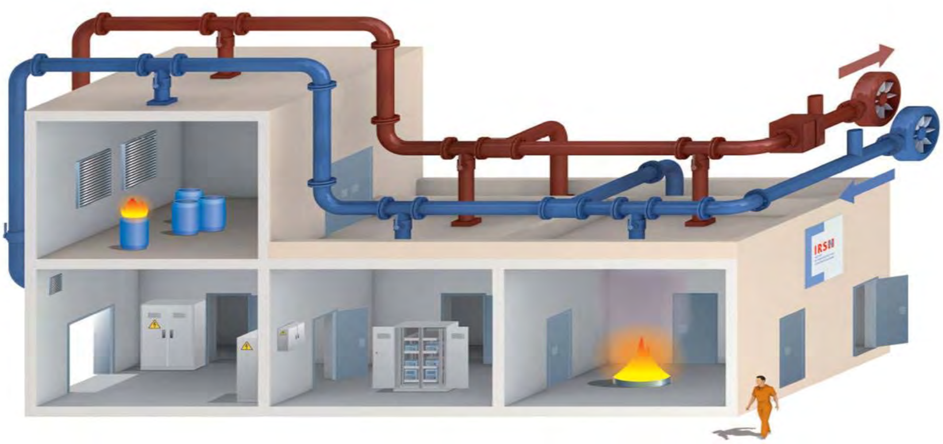
Project period: April 2009-October 2012.

Budget: USD ~1 million/year.

TH-10: Fire Propagation in Elementary, Multi-room Scenarios (PRISME) Project

The PRISME Project consisted of a series of fire and smoke propagation tests in a dedicated facility at the French *Institut de radioprotection et de sûreté nucléaire (IRSN*) centre at Cadarache. The acronym PRISME comes from the French phrase *Propagation d'un incendie pour des scénarios multi-locaux élémentaires*, which in English can be translated as "Fire propagation in elementary multi-room scenarios". The related facility, named DIVA, is used to investigate room-to-room heat and smoke propagation, the effect of network ventilation and the resulting thermal stresses to sensitive safety equipment of such room configurations. It was also planned to use data from the project to study multi-room fires and for validating fire computer codes.

The DIVA facility, Cadarache, France



Source: OECD/NEA PRISME Project.

Several propagation modes were studied: through a door; along a ventilation duct that crosses the room containing the fire and that ventilates an adjacent room; along a ventilation duct when flow is reversed within; and through leakages between several rooms.

The project aims to provide such critical information as the time that elapses before target equipment malfunctions and to qualify computer codes modelling heat and smoke propagation phenomena. The objective is to answer questions concerning smoke and heat propagation inside an installation, by means of experiments tailored for code validation purposes. In particular, the project aims to provide answers to the following questions:

- 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 will be used as a basis for qualifying fire codes (either simplified zone model codes or computerised fluid dynamics codes used in the fire safety analysis of nuclear installations and plants). 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, this data could be useful for establishing the suitable operation of the plant, such as the operation of the ventilation network (e.g. closing dampers to reduce the ventilation flow rate or to stop the ventilation) in case of a fire.

The conditions for the entire test series were addressed in the meetings, including ways to support the experimental projects with analyses and code assessments. These tests have also involved some facility modifications to meet specific members' requirements. The PRISME integral tests were carried out as per the schedule (these involve fire scenarios with the possibility to integrate specific devices and conditions).

Final report

Scheduled for end 2011.

Participants

Belgium, Canada, Finland, France, Germany, Japan, Netherlands, Republic of Korea, Spain, Sweden, United Kingdom, United States.

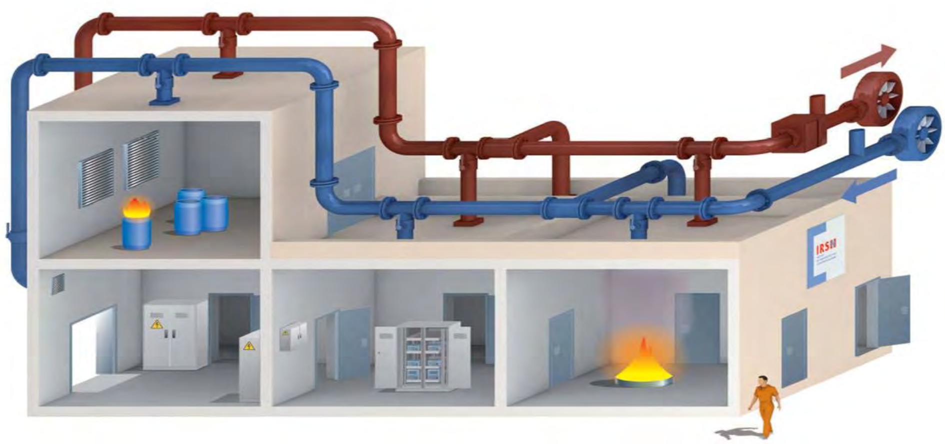
Project period: January 2006-June 2011.

Budget: EUR 7 million.

TH-11: Fire Propagation in Elementary, Multi-room Scenarios-Phase 2 (PRISME-2) Project

The PRISME-2 Project (2011-2016) is a follow-up of the PRISME Project and addresses some of the outstanding safety issues particularly with respect to the appreciable uncertainties that currently exist in risk assessments for nuclear power plants due to gaps in knowledge and modelling capabilities on fire growth and propagation, on fire extinction phenomena, on the prediction of damage to equipment and on the treatment of plant and operator response to a fire event.

The DIVA facility, Cadarache, France



Source: OECD/NEA PRISME-2 Project.

The PRISME-2 technical programme consists of the following test campaigns:

- experiments related to smoke and hot gas propagation through a horizontal opening between two superimposed compartments;
- fire spreading scenarios on real fire sources such as cable trays and electrical cabinets and fire propagation from one fire source to another;
- fire extinction studies of the performance of various extinguishing systems; and
- an open test campaign to build-upon and further investigate some of the phenomena from the earlier campaigns.

Participants

Belgium, Canada, Finland, France, Germany, Japan, Spain, Sweden.

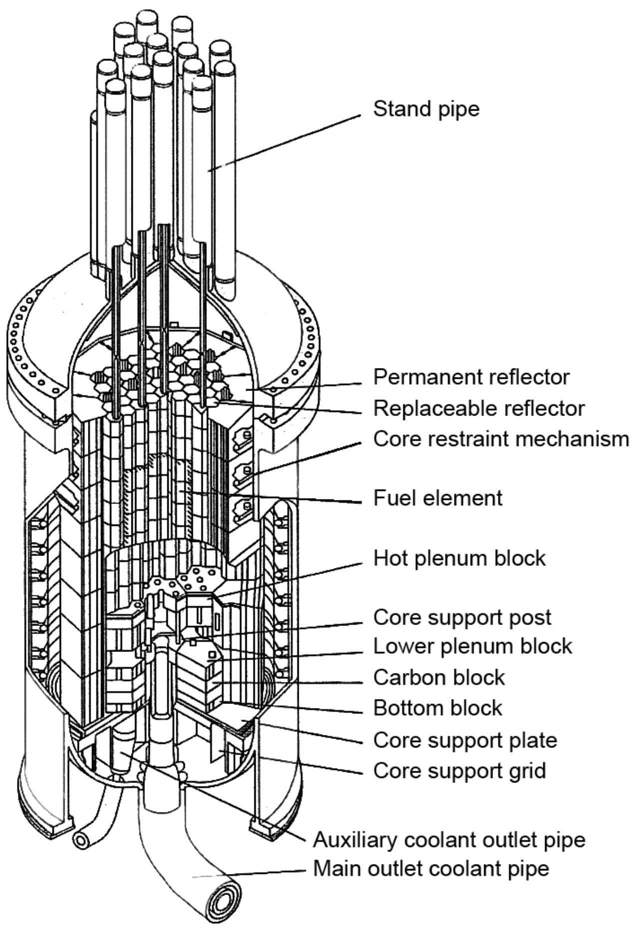
Project period: July 2011-June 2016.

Budget: EUR 7 million.

TH-12: Loss of Forced Coolant (LOFC) Project

The Loss of Forced Coolant (LOFC) Project started in April 2011 following a recommendation from the CSNI Task Group on Advanced Reactor Experimental Facilities (TAREF) for gas-cooled reactor safety studies issued in 2009. The LOFC experiments planned to study effects of reactor cavity cooling system (RCCS) performance reduction are highly relevant for safety assessments of advanced reactors such as high temperature reactor. Experiments are to be run by the Japan Atomic Energy agency (JAEA) in its high temperature engineering test reactor (HTTR) in Oarai, Japan.

The HTTR reactor, Oarai, Japan



Source: OECD/NEA LOFC Project.

The objectives of the proposed project are to conduct integrated large scale test of LOFC in the Japan Atomic Energy Agency (JAEA) HTTR reactor, to examine HTGR safety characteristics in support of regulatory activities, and to provide data useful for code validation and improvement of simulation accuracy. The objective of the experimental programme is to:

- provide experimental data to clarify the anticipated transient without scram (ATWS) in case of LOFC with occurrence of reactor re-criticality;
- provide experimental data for validation for one of the most important safety aspects about reactor kinetics, core physics and thermal-hydraulics; and

- provide experimental data to verify the capabilities of the codes regarding the simulation of phenomena coupled between reactor core physics and thermal-hydraulics.

These goals will be achieved by using the HTTR to perform three test cases, run one through three, whose results comparison will provide the incremental performance availability within the VCS range. The LOFC test is initiated by tripping all three helium gas circulators (HGCs) of the HTTR while deactivating all reactor reactivity control to disallow reactor scram due to abnormal reduction of primary coolant flow rate. The test falls into ATWS with occurrence of reactor re-criticality. The test will be conducted with and without active function of the VCS.

Participants

Czech Republic, France, Germany, Hungary, Republic of Korea, Japan, United States.

Project period: 31 March 2011-31 March 2013.

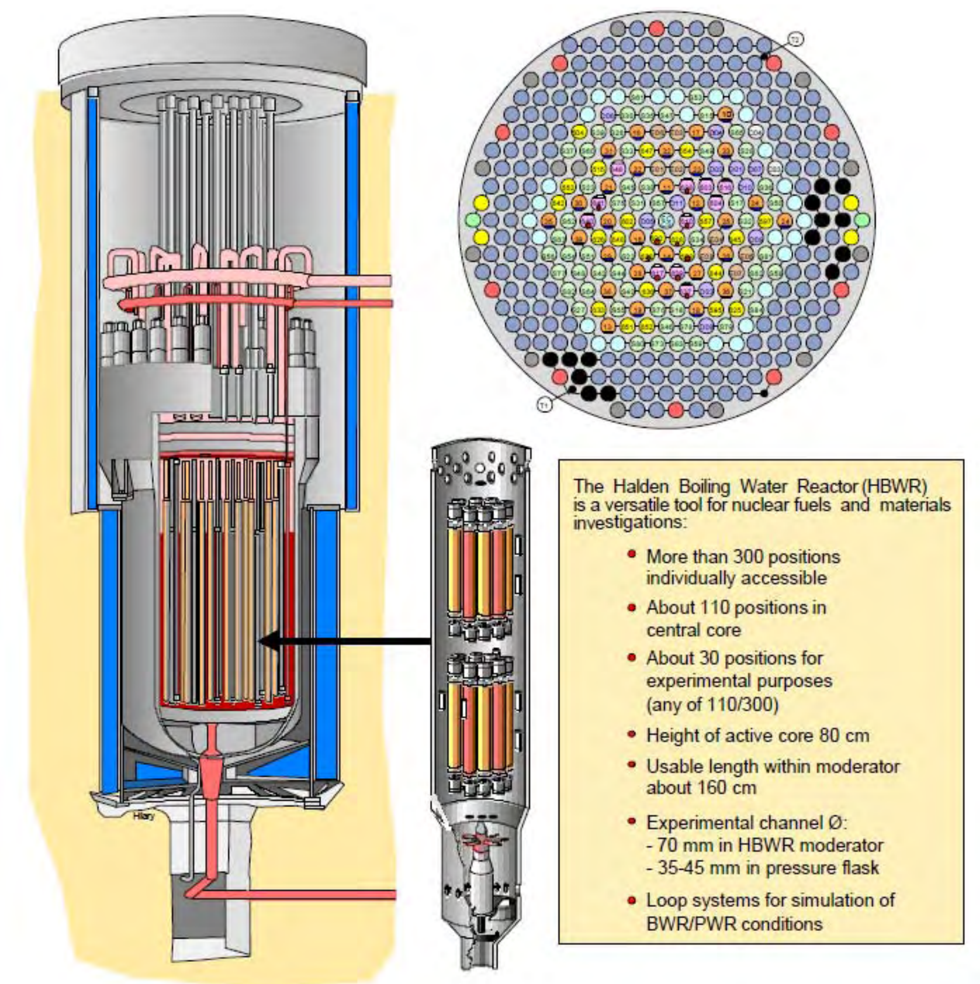
Budget: EUR 3.0 million.

FU-1: Halden Reactor Project

In 1956, the Council of the OEEC (now OECD) decided to investigate the possibility of joint research projects in the area of nuclear power. In the following year, the Norwegian Institutt for Atomenergi (IFA, today IFE) proposed the use of the Halden reactor for a joint research project spanning over a three year period. In June 1958, one year before the reactor began operation, the agreement that established the Halden Project as an international research project was signed. The Halden Project was a reality and this first 3-year agreement between Norway and 11 other countries became a leading example for international atomic research co-operation.

The original aim of the Halden Project was to pursue reactor technology research associated with the boiling heavy water reactor with a view to power production. The focus areas of research have of course varied over the years. However, throughout its 50-year history, the project has been working at the forefront of research within the areas of reactor fuel and materials testing, and the development of computer based systems for surveillance and control. Both fields are crucial to reactor safety.

The Halden reactor, Norway



Source: OECD/NEA HALDEN Project.

The Halden Project research programme is supported by organisations in 18 countries and is currently formulated based on the input and guidance from the Halden Programme Group, the Halden Board of Management and the member organisations. This input is expressed in a series of white papers on specific issues, in a priority-setting process for the different programme items as defined by the project participants, and in discussions within the Halden Programme Group and with each member organisations. The long-term directions are defined by the Halden Board, account taken for the continuity required to follow up on programme elements already addressed in earlier periods. The availability of suitable facilities also plays an important role in formulating realistic and affordable plans.

Fuels and Materials Programme: Safe and reliable utilisation of nuclear energy requires development and verification work in a variety of areas, since questions continue to arise as result of demands on plant performance and regulatory practice. The Halden Fuels Programme is intended to provide basic data on how the fuel performs in commercial reactors both at normal operation and in transient conditions. This regards both new fuels designed to meet the challenges posed by extended fuel utilisation and existing standard fuels which will be found in reactors cores many years ahead. Emphasis remains on fuel properties after prolonged service time in-reactor, but for new fuels the entire burn-up range will be addressed including demanding heat generation conditions. The Halden Materials Programme addresses ageing issues such as irradiation induced material changes and corrosion processes which can lead to progressive degradation of in-reactor components.

Man-Technology-Organisation Programme: People constitute a main element of the safety of nuclear power plants. The human factors research at the Halden Reactor Project in the Man Technology Organisation (MTO) area supports both design and safety assessments of new solutions for existing plants, upgrade and new builds. Modernisation of NPP control rooms is ongoing in many countries, moving from panel-based control rooms into hybrid, combining computerised and traditional technologies, or even to fully computerised solutions. Similarly, control rooms of current new builds consist mainly of digital solutions. The NEA/CSNI identifies human factors digital I&C as research topics for new nuclear plant technology. The proposed Halden Project programme has addresses these topics in its recent programme, and will continue to address it in the future. The Halden Project performs research on control centres, innovative HSIs, design methods, training, outage and future operational concepts to meet demands of the nuclear industry. It serves modernisation projects and new builds with technical basis for guidelines, and ideas for new and innovative solutions, as well as regulators' need for knowledge on evaluation of modern digital control centres. Human reliability analysis (HRA) is a significant issue in probabilistic risk/safety assessment (PRA/PSA) for nuclear power plants. NEA and CSNI have recently emphasised work on international studies to investigate human performance in a variety of hypothetical scenarios. These items constitute an important part of the Halden MTO Programme.

Participants

Belgium, Czech Republic, Denmark, Finland, France, Germany, Hungary, Japan, Kazakhstan, Republic of Korea, Norway, Slovak Republic, Spain, Sweden, Switzerland, United Kingdom, United States.

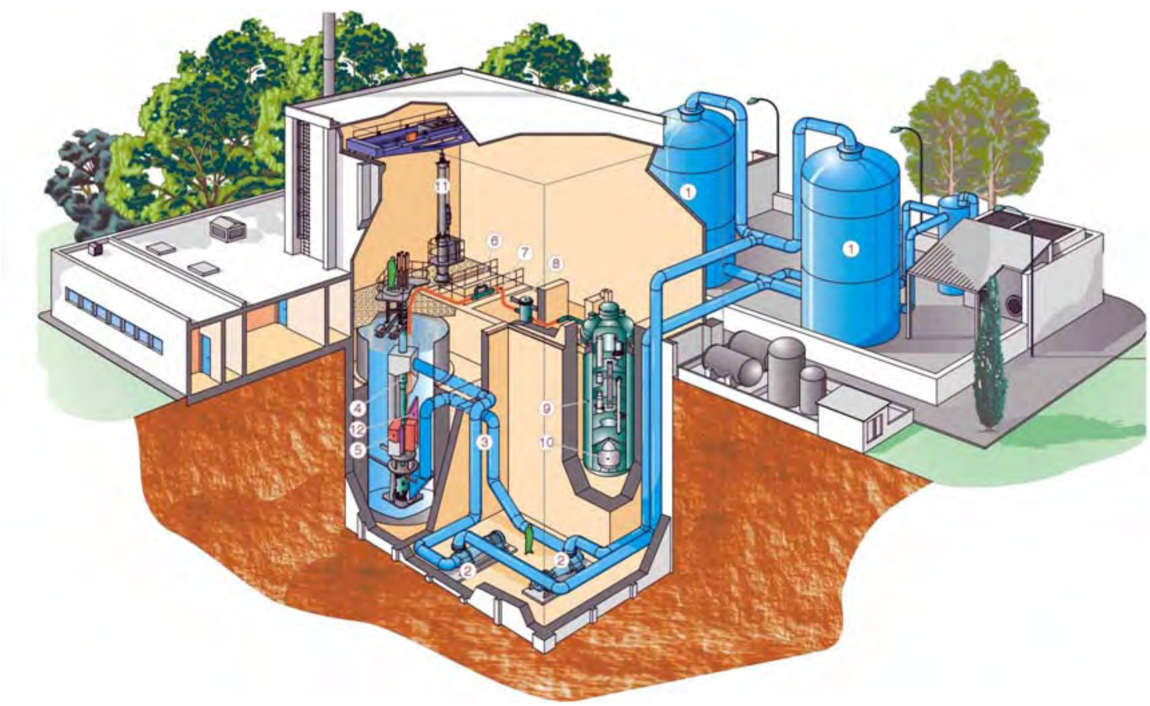
Programme period: Started in 1958. Current period runs until December 2014.

Budget: EUR ~15 million/year.

FU-2: Cabri Water Loop International Project

The Cabri Water Loop International 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. These are referred to as reactivity-induced accidents (RIA). The project stems from previous Japanese and French tests conducted on high burn-up pressurised water reactor (PWR) fuel that in a few instances exhibited failure at relatively moderate energy deposition levels. The Cabri Project aims to extend the database for high burn-up fuel performance in RIA conditions and more importantly, to perform relevant tests under coolant conditions representative of PWRs.

The CABRI reactor, Cadarache, France



Source: OECD/NEA CABRI Project.

The project began in 2000 and will run until 2015. The experimental work will be 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 characterisation or post-irradiation examinations).

Twelve tests are proposed as a basis for co-operation with IRSN partners in the Cabri Water Loop Project: these experiments include R&D tests, combined with suitable tests to validate the extrapolation to a broad spectrum of reactor cases. The experimental programme is being reviewed by the project steering bodies and consists (in principle) of the following series of tests:

- Series S0. Two tests in the sodium loop aimed at providing data at high burn-up. *All subsequent tests will be performed in the water loop*.
- Series S1. Two tests to provide comparison with S0 tests (and pulse width effect).
- Series S2. Two tests on behaviour of very high burn-up fuel.
- Series S3. Two tests on RIA phenomena and modelling.

- Series S4. Two tests on MOX fuel behaviour.
- Series S5. Two complementary tests to be specified.

Two tests have been carried out so far using fuel with very high burn-up and modern cladding materials. Subjected to energy injection beyond what is expected for power reactor cases, the two test fuel rods did not fail during the tests carried out in 2003. Planned future tests will aim to develop a consistent set of objectives and suitable fuel specimens. Post-irradiation examinations of the two tests that have been carried out so far were undertaken in 2004. They involved destructive examinations and investigated in particular the effect of hydrogen on cladding properties. The planning of future tests continues, with the aim of developing a consistent set of objectives and identifying suitable fuel specimens. Considerable progress was made on the refurbishment of the Cabri test facility and the preparation of the water loop installation, a process estimated to end in 2011. The Cabri tests are complemented by additional RIA tests carried out in the Japanese Nuclear Safety Research Reactor (NSRR).

Reports

A number of reports available to participants.

Participants

Czech Republic, Finland, France, Germany, Japan, Republic of Korea, Slovak Republic, Spain, Sweden, Switzerland, United Kingdom, United States.

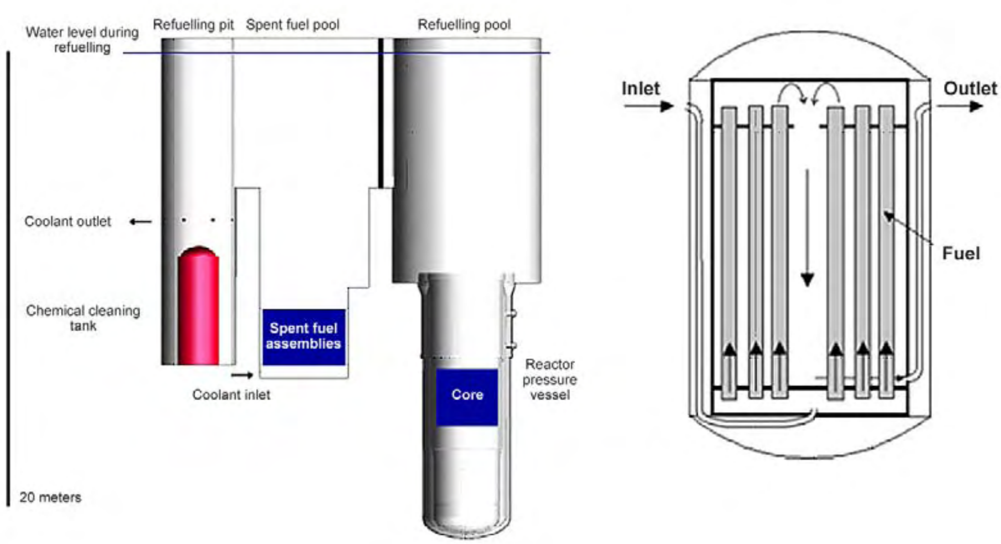
Project period: 2000-2015.

Budget: EUR 75 million.

FU-3: OECD/NEA-IAEA Paks Fuel Project

During a fuel crud removal operation on the Paks-2 unit on 10 April 2003 several fuel assemblies were severely damaged. The assemblies were being cleaned in a special tank of the Paks nuclear power plant, Hungary under deep water level in a service pit connected to the spent fuel storage pool. The first sign of fuel failures was the detection of some fission gases released from the cleaning tank. Later visual inspection revealed that most of the 30 fuel assemblies suffered heavy oxidation and fragmentation. The first evaluation of the event showed that the severe fuel damage happened due to inadequate cooling.

The Paks NPP fuel cleaning tank, Hungary



Source: OECD/NEA IAEA Paks Project.

The Paks-2 event was discussed in various committees of the OECD Nuclear Energy Agency (OECD/NEA) and of the International Atomic Energy Agency (IAEA). Recommendations were made to undertake actions to improve the understanding of the incident sequence and of the consequence this had on the fuel.

It was considered that the Paks-2 event may constitute a useful case for a comparative exercise on safety codes, in particular for models devised to predict fuel damage and potential releases under abnormal cooling conditions and the analyses on the Paks-2 event provided information which is relevant for in-reactor and spent fuel storage safety evaluations.

The OECD-IAEA Paks Fuel Project was established in 2005 as a joint project between the IAEA and the OECD/NEA. The IAEA provided financial support to the operating agent [Hungarian Academy of Sciences KFKI Atomic Energy Research Institute (AEKI)] and reviewed the progress of the project within the framework of the TC Project, RER9076 “Strengthening Safety and Reliability of Fuel and Material in Nuclear Power Plants”.

A common database was established upon collecting the necessary information for the simulation of the Paks-2 event. Based on that, numerical analyses were carried out with sophisticated models and codes that are in use for safety assessment of nuclear power plants. The simulations covered thermal-hydraulics, fuel behaviour and activity release aspects. The results were compared to available measurements from the event. The OECD-IAEA Paks Fuel Project improved our current knowledge on fuel behaviour

 MAIN BENEFITS FROM 30 YEARS OF JOINT PROJECTS IN NUCLEAR SAFETY, ISBN 978-92-64-99171-2, © OECD 2012

under accident conditions and recommended some further actions for research in this area.

Final report

Final report of the OECD-IAEA Paks Fuel Project, NEA/CSNI/R(2008)2, September 2008.

Participants

Belgium, Finland, France, Germany, Hungary, Russian Federation, Slovak Republic, United States.

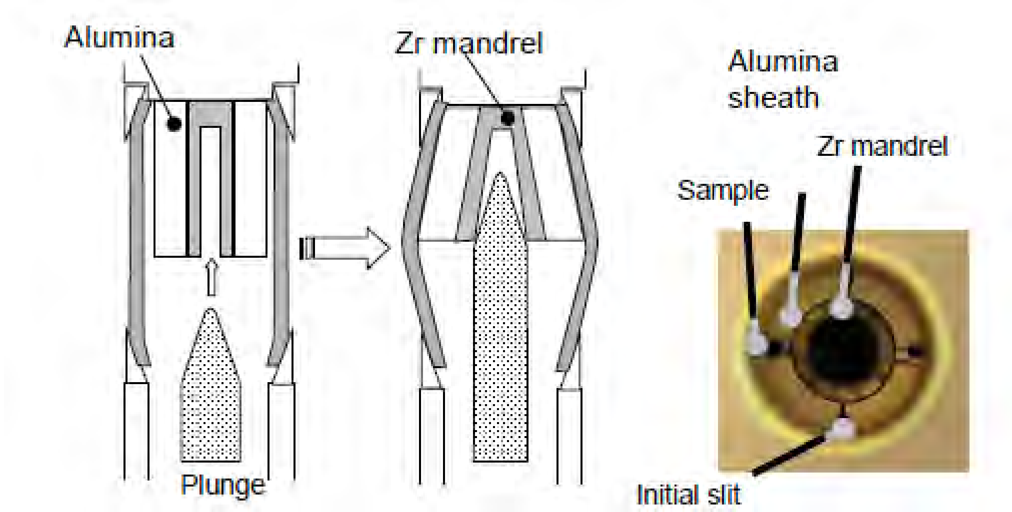
Project period: November 2004-November 2007.

Budget: Sponsored by the IAEA.

FU-4: Studsvik Cladding Integrity (SCIP) Project

New nuclear 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 operating conditions and in transients. Assessments should also cover conditions such as those prevailing in fuel handling and storage.

Cladding test at Studsvik, Sweden



Source: OECD/NEA SCIP Project.

In order to achieve this goal, a fundamental understanding of the dominant cladding failure mechanisms is needed. The OECD/NEA Studsvik Cladding Integrity Project (SCIP) addressed this need. The overriding objective of this project was to obtain an improved fundamental understanding of the dominant failure mechanisms for light water reactor (LWR) fuel cladding under pellet-clad mechanical interaction (PCMI) loading that can arise during normal operation or anticipated transients. The focus was on the following failure mechanisms, which were studied within the corresponding project tasks:

- PCMI driving force;
- pellet-clad interaction (PCI): stress corrosion cracking (SCC) 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.

For each of these failure mechanisms, the project objectives can be broken down into the following types of investigation:

- quantifying the key parameters and their influences through separate effects studies;
- identifying the key physical processes and phenomena that are involved in each failure mechanism and theoretical modelling of the fracture mechanisms;

- developing and refining testing techniques for the experimental verification of the fuel behaviour with respect to these failure mechanisms.

The project helped to sustain the existing information legacy and facilitate the transfer of knowledge by means of seminars and workshops. The project also had the following general objectives:

- to improve the general understanding of cladding integrity at high burn-up;
- to study both boiling water reactor and pressurised water reactor/VVER fuel cladding integrity;
- to complement two large international projects (CABRI and ALPS), which focus on fuel behaviour in design basis accidents (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). The results can consequently be used in solving a wider spectrum of problems and can 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 was the integrity of LWR cladding during in-reactor service, several closely related areas were addressed, which can be relevant to water reactors in general. As already mentioned, the cladding behaviour of discharged fuel during handling, transportation and storage shares some features in common with the cladding's integrity whilst in the reactor.

From the Swedish side, the project was supported by the utilities, the Swedish Nuclear Inspectorate and Westinghouse Atom.

The project has focused on the execution of several power ramps and in defining a hot cell programme with focus on the various failure mechanisms, which were studied within the corresponding project tasks. These are as follows:

1. pellet-clad interaction (PCI): stress corrosion cracking (SCC) 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;
2. hydride embrittlement: time independent fracture of existing hydrides;
3. delayed hydride cracking (DHC): time dependent crack initiation and propagation through fracture of hydrides that can form ahead of the crack tip.

Critical parameters for crack initiation and propagation by hydrogen embrittlement (HE), delayed hydride cracking (DHC) and shear instability have been identified and to some extent quantified. Approximate stress and strain levels at which those types of failures are induced during a power ramp have been established but further testing and analysis are needed to verify the results, and to further narrow down the critical stress and strain range. Overall, the results obtained within SCIP so far, have improved our understanding of PCMI behaviour and PCMI driven failures, especially hydrogen induced failures in high burn-up cladding. We now have a better understanding of the combined effect of different materials and test parameters on hydrogen induced failures. A follow-up phase of the project (SCIP-2) was decided for the period July 2009-June 2014.

Final report

"Studsvik Cladding Integrity Programme Project, Comprehensive Technical Report", Studsvik, June 2011.

Participants

Czech Republic, Finland, France, Germany, Japan, Republic of Korea, Spain, Sweden, United Kingdom, United States.

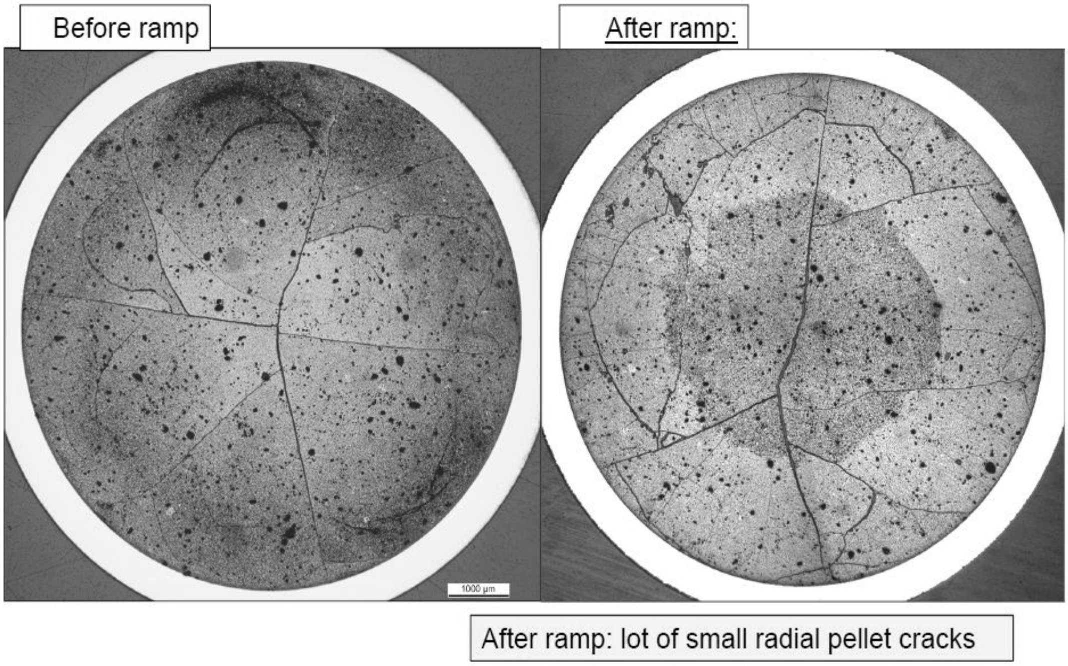
Project period: July 2004 to June 2009.

Budget: SEK 12 million (i.e. about USD 1.8 million)/year. Swedish parties covered 50% of this cost.

FU-5: Studsvik Cladding Integrity-Phase 2 (SCIP-2) Project

Phase I of the Studsvik Cladding Integrity Project (SCIP) started in July 2004 and was completed in June 2009. It utilised 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. A second phase of this project (SCIP-2) started in July 2009 and will build on the considerable knowledge generated in the previous SCIP programme.

Fuel power ramp tests at Studsvik, Sweden



Source: OECD/NEA SCIP-2 Project.

The goal of the project is to generate high quality experimental data to improve the understanding of the dominant failure mechanisms for water reactor fuels and to devise means for reducing fuel failures. The major focus is on cladding failures that are caused by pellet-cladding mechanical interaction, especially stress corrosion and hydrogen-assisted fracture mechanisms, as well as on the propagation of cladding cracks. 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 be applied to different cases. It also aims to achieve experimental efficiency through the judicious use of a combination of experimental and theoretical techniques and approaches.

The nuclear fuel failure mechanisms studied in SCIP-2 are as follows:

- review of existing Studsvik ramp data: to make available as much of the existing ramp data as possible from ramps performed in Studsvik under various programmes;
- pellet-cladding mechanical interaction (PCMI): the mechanical driving force for PCI and hydrogen-induced failures;
- pellet-cladding interaction (PCI): fuel rod failures where the cladding fails by stress corrosion cracking;

- hydrogen induced failures: two hydrogen-induced failure mechanisms observed in zirconium alloys [classical hydride embrittlement (HE) and delayed hydrogen cracking (DHC)].

Final report

Scheduled for 2015.

Participants

Czech Republic, Finland, France, Germany, Japan, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

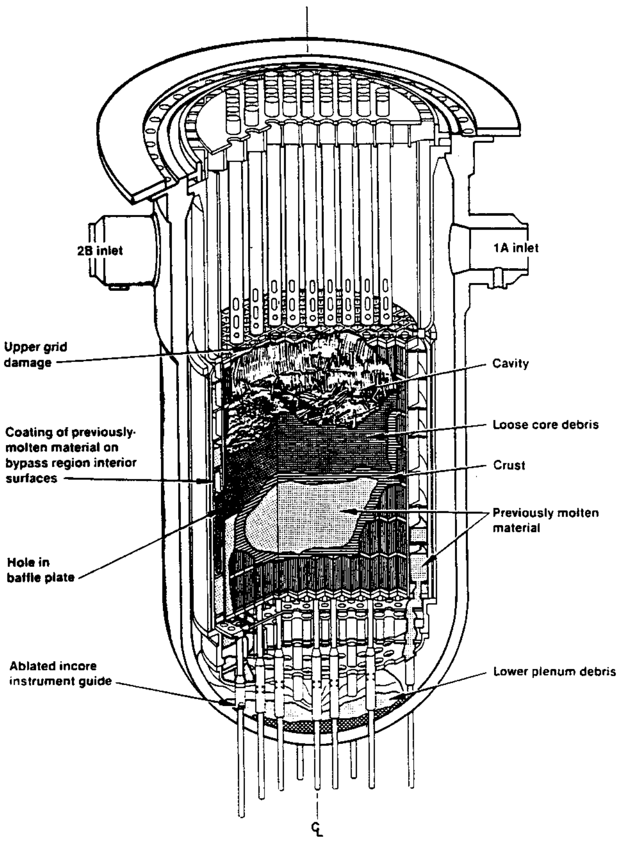
Project period: July 2009-June 2014.

Budget: SEK 15 million (about EUR 1.5 million)/year. Swedish parties cover 50% of this cost.

SA-1: Three Mile Island Vessel Investigation Project (TMI-VIP)

A Three Mile Island Unit 2 (TMI-2) Accident Evaluation Programme was originally set up by the US Department of Energy, then part of the programme was later broadened into an international collaboration project under the aegis of the OECD Nuclear Energy Agency (NEA). The initial programme was concerned primarily with the core damage progression analysis, metallographic studies of core debris samples and structural materials, as well as the mechanisms controlling fission product behaviour during the accident. However it became apparent after the programme had been established that the TMI-2 accident, which occurred 28 March 1979, had progressed further than believed. Large quantities of molten core material had relocated from the core to the lower plenum of the reactor pressure vessel, and thermal damage had occurred to instrument structures in the lower head region; hence the US Nuclear Regulatory Commission (USNRC) asked NEA to set up a second international collaborative project, the Three Mile Island Vessel Investigation Project (TMI-VIP) to examine additional aspect of the question.

View of TMI-2 core debris, United States



Source: OECD/NEA TMI-VIP Project.

The programme, subject of the OECD/TMI-VIP Project, was designed to evaluate the potential modes of failure and the margin to failure of the TMI-2 reactor vessel during the TMI-2 accident. The conditions and properties of material extracted from the lower head of the TMI-2 reactor pressure vessel were investigated to determine the temperature conditions and the extent of the damage by chemical and thermal attack on the lower head, as well as the margin of structural integrity of the vessel during the accident.

This project enabled progress in three directions: a better view and understanding of the TMI-2 accident scenario over the time (it took 8-10 years to discover the potential for lower head failure), the technical conclusions of the TMI-2 VIP (no evidence of significant bottom vessel creep and additional cooling area provided with debris bed formation) and the broad significance of these findings for accident management (importance of maintaining cooling water and importance of limiting vessel pressure). This project showed that global creep failure of the reactor vessel could occur under conditions of high vessel temperature and pressure. Therefore, accident management procedures should recognise: 1) the importance of cooling water not only for the reactor core, but also for limiting the reactor vessel wall temperature; and 2) the need for controlling pressure to avoid vessel failure. This project also noted the need for additional research to improve the likelihood of maintaining the reactor vessel integrity under conditions of a severe accident.

Final report

"TMI-2 Examination Results from the OECD/CSNI Programme – Vol. 1 and Vol. 2 – INEL", April 1992, also issued as NEA/CSNI/R(91)9.

Participants

Belgium, Finland, France, Germany, Italy, Japan, Spain, Sweden, Switzerland, United Kingdom, United States.

Budget: USD 9.0 million.

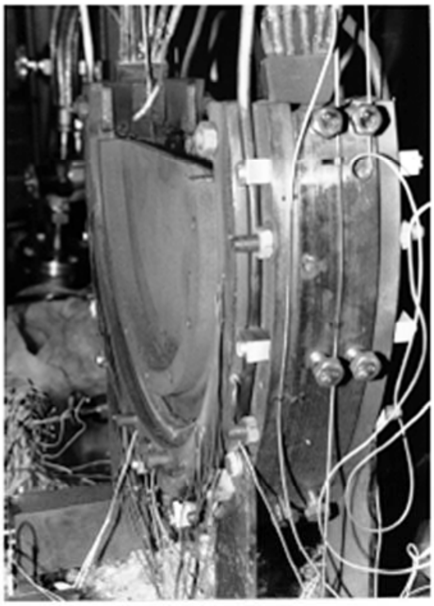
Project period: 1st January 1988-31st March 1993.

SA-2: RASPLAV Project

The RASPLAV Project aimed to refine accident management strategies during a reactor core meltdown; it was completed in June 2000. Little is known about the complex interactions that take place during a core meltdown, so one of the RASPLAV project's primary goals was to develop an understanding of this process. The information gathered during tests at the Kurchatov Institute has allowed scientists to develop models of a core meltdown. These models can be used in the design of new reactors and in refining the accident procedures for existing ones. Two aspects of the issue were considered. First, for existing reactors, where external cooling may not be practicable, the process and time sequence before melt-through were studied. This was to help develop management strategies for severe accidents. Secondly, for future and some existing reactor designs, the project aimed to determine the heat transfer conditions under which cavity flooding can be a viable accident management option.

The RASPLAV facility, Kurchatov Institute, Russian Federation

View from graphite heater



View from test wall

Source: OECD/NEA RASPLAV Project.

The project was run in two successive phases. The RASPLAV Phase-1 Project was dedicated mainly to the build-up of the experimental and analytical infrastructure. The RASPLAV Phase-2 Project investigated the progression of a severe accident and in particular the thermal loading imposed by a corium pool on the lower head of a light water reactor (LWR) vessel.

The project objectives were to obtain relevant data on the physical and thermal behaviour of the corium in large scale tests, to derive thermal-physical property data for various molten core materials, and to investigate the effects of stratification of molten materials. The programme of work involved the use of the large facilities available at the Kurtchatov Institute in Russia. Four large scale tests were carried out and were complemented by a series of smaller scale experiments, all involving the use of materials representative of power reactor cores. Experiments with these test materials in molten condition required temperatures of approximately 3 000 °C – a very challenging task, especially for large scale tests. The analytical work was done at the Russian Academy of Science's Institute of Nuclear Safety (IBRAE).

The project was successfully concluded in 2000. The results are set out in the project final report which was presented and reviewed in November 2000 at the last RASPLAV seminar, organised in Munich, Germany. The project data became open in 2004.

These experiments provided the necessary data for code validation. However, some important issues remained open regarding conditions which may lead to stratification and the associated partitioning of fission products and decay heat. These issues were considered under the follow-up programme MASCA.

Final report

"RASPLAV Final Report – Behaviour of the Corium Molten Pool under External Cooling", Kurchatov Institute, July 2000, also available as NEA/CSNI/R(2000)25.

Participants

Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, Netherlands, Republic of Korea, Russian Federation, Spain, Sweden, Switzerland, United Kingdom, United States.

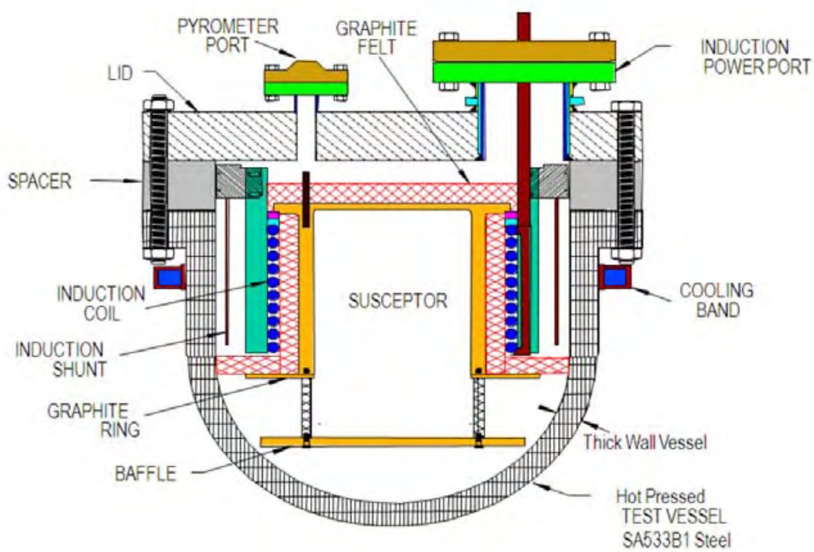
Programme period: June 1994-June 2000.

Budget: USD 11.9 million.

SA-3: OECD Sandia Lower Head Failure (OLHF) Project

During a severe accident, the lower head of the reactor pressure vessel (RPV) can be subjected to significant thermal and pressure loads. It is possible that the lower head will fail, releasing large amounts of molten corium into containment. The Three Mile Island accident involved the melting of about 20 tonnes of corium, which collapsed into the lower head of the RPV. Despite the presence of water, the lower head reached temperatures of ~1 300 K for 30 minutes in an area with an equivalent diameter of 1 m. During this period the reactor cooling system was at 10 MPa. Although the Three Mile Island vessel did not fail, code analyses conducted in the course of an OECD/NEA TMI-II Vessel Investigation Project (VIP) predicted creep rupture in the prevailing conditions. This implies that the then state-of-the-art modelling of the lower head failure was not mature because it did not take full account of the effect of the thermal loading. These methodologies have been further developed since the TMI-VIP Project to analyse existing and next generation reactors from the perspective of accident assessment, management and mitigation. In order to improve and validate structural analysis codes, there was a need for experimental data on lower head deformation and failure phenomena.

The OLHF facility, Sandia, United States



Source: OECD/NEA OLHF Project.

The objective of the OLHF Project was to investigate the timing and size of lower head failure under conditions of low reactor coolant system pressure and large differential temperatures across the lower head wall. This objective was achieved through a series of experiments at the Sandia National Laboratory, United States completed in June 2002.

The Sandia National Laboratory has formerly completed eight USNRC-sponsored tests on lower head failure (LHF). These tests were specifically designed to address lower head failure issues with prototypic material and geometry. The OECD project extended the USNRC SNL/LHF programme to address issues such as lower RCS pressures (representative of depressurised or partially depressurised conditions) and pressure transients. These tests also represented an improvement over previous tests by simulating a large temperature gradient across RPV of lower head wall. The temperature gradients addressed in these tests are representative of conditions without ex-vessel cooling.

The key observations from the integral experiments were that:

- large temperature differential leads to failure at higher inside wall temperature;
- failures are typically localised at locations where the ratio of membrane stress to yield stress was highest;
- the "critical effective strain" for a uniformly heated vessel without penetrations was found to be ~30%. Penetration failure occurs at much lower effective strain.

The tensile and creep properties were measured for temperatures up to ~1 300 K.

This project has produced a unique and well-qualified data set for code validation. These experiments and associated analyses have laid a sound foundation for predictive modelling of lower head failure.

Final report

"OECD Lower Head Failure Project Final Report", Sandia National Laboratories, December 2002 – NEA/CSNI/R (2002)27.

Participants

Belgium, Czech Republic, Finland, France, Germany, Spain, Sweden, United States.

Programme period: September 1998-June 2002.

Budget: USD 1.9 million.

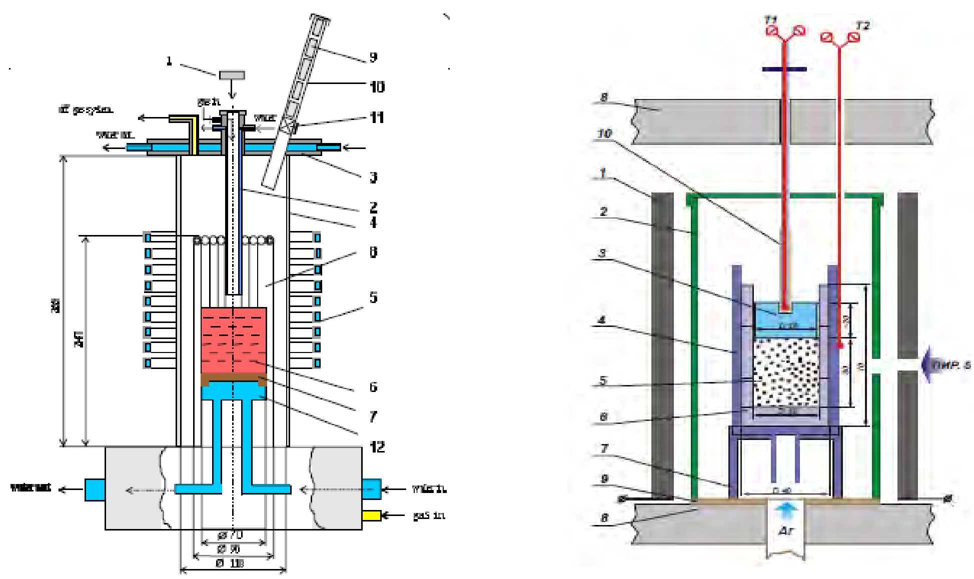
SA-4: Material Scaling (MASCA) Project

The MASCA Project was a follow-up to the RASPLAV Project and investigated in-vessel phenomena during a severe accident. In particular, it addressed the influence of the chemical composition of the molten corium on the heat transfer to the pressure vessel environment. The project addressed this by investigating stratification phenomena of the molten pool and the partitioning of fission products (FP) within the different layers of the melt. The project was scheduled to be completed in July 2003, but it was continued until 2006 under the MASCA-2 Project, given the experimental needs that still existed and the quality of the experimental work done up to that point. The tests aimed to resolve remaining uncertainties about the heat load on the reactor vessel and thus the possibility of retaining the melt in the vessel. These uncertainties are mainly associated with scaling effects and coupling between the thermal-hydraulic and chemical behaviour of the melt. Supporting experiments and analyses – in addition to helping understand key in-vessel phenomena – facilitated a consistent interpretation of the results. The experiments were carried out with corium compositions prototypical of power reactors which use iron and steel materials.

The MASCA experimental goal was achieved through corium tests of different scale and was complemented by pre- and post-test analyses and development of computational models. Additional measurements of thermo-physical properties of the melts such as density, thermal conductivity and liquidus-solidus temperatures considerably expanded the material properties data obtained during the RASPLAV Project. The major goals of the MASCA Project were to:

- investigate the influence of chemical behaviour on heat transfer in stratified molten pools of prototypical compositions;
- investigate FP behaviour in a molten pool and in particular:
  - partitioning of FP between layers in case of stratification;
  - partitioning of FP between phases during melting and solidification;
- distribution of FP simulants in the melts;
- expand the material properties database;
- develop computer models describing the relevant phenomena.

The RASPLAV facility, Kurchatov Institute, Russian Federation



Source: OECD/NEA MASCA Project.

Separate effects were studied in series of small- and mid-scale experiments. Corium tests were performed in the RASPLAV, TULPAN, TF, STF, KORPUS and TIGEL facilities. Salt tests were performed in the present RASPLAV-salt facility.

Experiments with the molten steel and corium revealed important peculiarities of interactions. Interactions between molten steel and corium results in mass exchange between phases when some amounts of metallic uranium and zirconium initially located in the suboxidised corium pass into metallic phase, leading to the change of density and sinking of the metal layer. In the temperature range between corium solidus and liquidus temperatures molten steel sufficiently overheated above the melting point (200K) easily penetrated the corium debris. However, steel zirconium metal melt does not leak through the debris due to formation of zirconium carbides and oxycarbides. Partitioning of fission products revealed that metallic fission products are concentrated in the metal phase while oxide fission products concentrated in the oxide phase. The partitioning likely depends on temperature, however insufficient tests were conducted to establish dependence of fission products partitioning as a function of temperature.

Final report

"Main Results of the First Phase of MASCA Project, Integrated Report", Kurchatov Institute, November 2004, also available as NEA/CSNI/R(2004)23.

Participants

Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, Netherlands, Republic of Korea, Russia, Spain, Sweden, Switzerland, United Kingdom, United States.

Project period: July 2000-July 2003 (extended until 2006 as the MASCA-2 project).

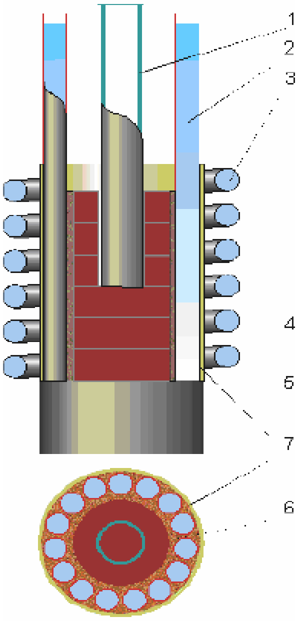
Budget: USD 3 million.

SA-5: Material Scaling-Phase 2 (MASCA-2) Project

The MASCA-2 Project was an extension of the MASCA Project. It was based on experiments that were mainly carried out at the Kurchatov Institute and that make use of a variety of facilities in which the corium compositions prototypical of power reactors could be tested.

Kurchatov Institute facilities, Russian Federation

RASPLAV-3 view



STF facility

Source: OECD/NEA MASCA Project.

The tests aimed to provide experimental information on the phase equilibrium for different corium mixture compositions that can occur in water reactors. In order to enhance the application of MASCA results for reactor cases, the influence of an oxidising atmosphere and the impact of non-uniform temperatures (presence of crusts or solid debris) was addressed. 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 first meeting of the MASCA-2 Project was held in October 2003 in Madrid, Spain. The refurbishment of several facilities was started in order to accommodate the specific requirements of the new test conditions. A MASCA seminar was held in June 2004 to review the project results to date. The project integration report is now available to participants and a MASCA-2 seminar was held on 11-12 October 2007 in Cadarache, France.

The integrated report providing main results of the MASCA 1 and 2 projects was issued in June, 2007. Studies of the influence of control materials were conducted and a special series of experiments dealt with the oxide and metal phases interactions in the oxidising atmosphere. The experiments under this project also provided additional data on physicochemical properties. Among those were measurements of C-32 corium properties: spreading temperature, kinematic viscosity in the temperature range between 2 380 and 2 600° C, and density in a liquid phase. Measurements of viscosity, thermal conductivity, and density of the liquid metallic mixtures were also performed. Moreover, electric conductivity and heat conductivity of metallic body were measured at low temperature.

Final report:

"Main Results of the MASCA-1 and 2 Project, Integrated application Report", D.F. Tsurikov, V.F. Strizhov, S.V. Bechta, V.N. Zagriazkin, N.P. Kiselev, Kurchatov Institute, June 2007, also available as NEA/CSNI/R(2007)15.

Participants

Belgium, Canada, Finland, France, Germany, Hungary, Japan, Republic of Korea, Russian Federation, Spain, Sweden, Switzerland, United States.

Project period: July 2003-June 2007.

Budget: USD 1 million/year.

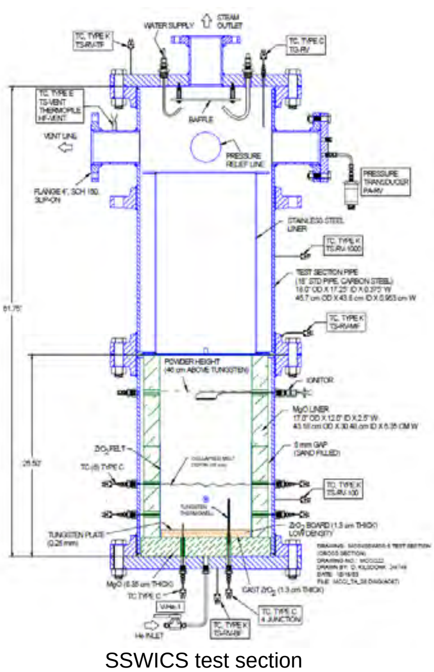
SA-6: Melt Coolability and Concrete Interaction (MCCI) Project

In a core melt accident, if the molten core is not retained in-vessel despite severe accident mitigation actions, the core debris will relocate to the reactor cavity region and interact with the structural concrete – potentially resulting in basemat failure through erosion or overpressurisation. This would result in the release of fission products into the environment. Although this is a late release event, the radiological consequences could be substantial enough to warrant an effective mitigation strategy for preventing such a release. The severe accident management guidance (SAMG) for operating light water reactor plants includes, as one of several strategies, flooding the reactor cavity in the event of an ex-vessel core melt release. The Melt Coolability and Concrete Interaction (MCCI) Project aimed to provide experimental data on these severe accident phenomena and to resolve two important accident management issues:

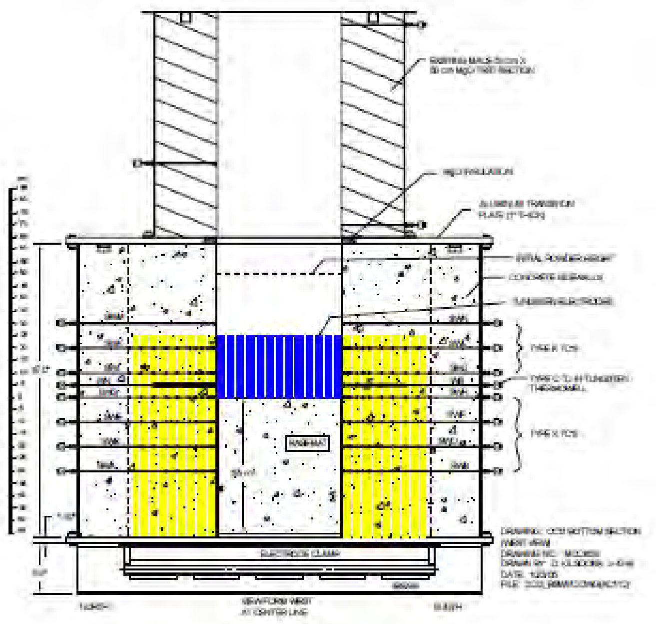
- verify that molten debris that has spread on the base of the containment can be stabilised and cooled by water flooding from the top;
- assess the two-dimensional, long-term interaction of the molten mass with the concrete structure of the containment, as the kinetics of such interaction is essential for assessing the consequences of a severe accident.

To achieve these basic objectives, supporting experiments and analyses were performed at Argonne National Laboratory (ANL), with a view to providing an understanding the phenomena, and to produce a consistent interpretation of the results relevant to accident management.

Argonne National Laboratory facilities, United States



SSWICS test section



CCI test section

Source: OECD/NEA MCCI Project.

An internationally-sponsored programme had been carried out beforehand at ANL to address the corium coolability issue. The MCCI Project aimed to complete this earlier research and achieve the following technical objectives:

- resolve the ex-vessel debris coolability issue through a redirected programme which focuses on providing both confirmatory evidence and test data for the coolability mechanisms identified in previous ANL integral effect tests;
- address remaining uncertainties related to the long-term, two-dimensional, melt-concrete interaction under dry cavity conditions.

Achieving these two programme objectives will lead to improved severe accident management guidelines for existing plants as well as better containment designs for future plants.

The first MCCI experiments focused on water ingress mechanisms, as these are thought to be the most effective ones for cooling the melt. These experiments have demonstrated how cooling of the melt by water is affected by the concrete-melt composition and that cooling of the melt by water is reduced at increasing concrete content, i.e. cooling by water flooding is more effective in the early phase of the melt-concrete interaction. The effect of concrete type, such as siliceous and limestone types (used respectively in Europe and the United States), has also been addressed. Material properties such as porosity and permeability have been derived from these tests.

A first melt-concrete interaction test with siliceous concrete in 2003 produced unexpected results (a strong asymmetry in concrete ablation), although the associated analytical exercise proved very valuable in helping to understand code capabilities and shortcomings. A second test was carried out in 2004 at 30% lower power than the first on limestone concrete (instead of the siliceous concrete used in the first test). The strength of the solid upper crust, a parameter that is of great interest for modelling and understanding MCCI at plant scale, was also determined during these experiments. A third test with siliceous concrete was successfully carried out in 2005, yielding excellent data on axial and radial concrete ablation.

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 took place on 10-11 October 2007 in Cadarache.

The project for follow-up with a second phase: MCCI-2 project run from 2006 to 2010.

Final report

"OECD MCCI Project Final Report", M.T. Farmer, S. Lomperski, D.J. Kilsdonk, and R.W. Aeschlimann, MCCI-2005-TR06, ANL, February 2006.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Japan, Norway, Republic of Korea, Spain, Sweden, Switzerland, United States.

Project period: January 2002 to December 2005.

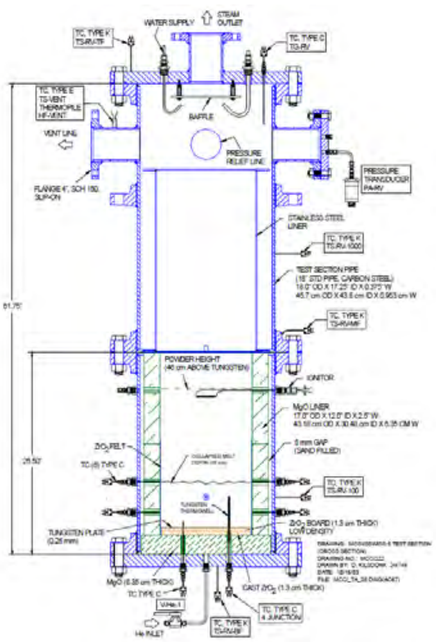
Budget: USD 1.2 million/year.

Project management: US Nuclear Regulatory Commission (USNRC).

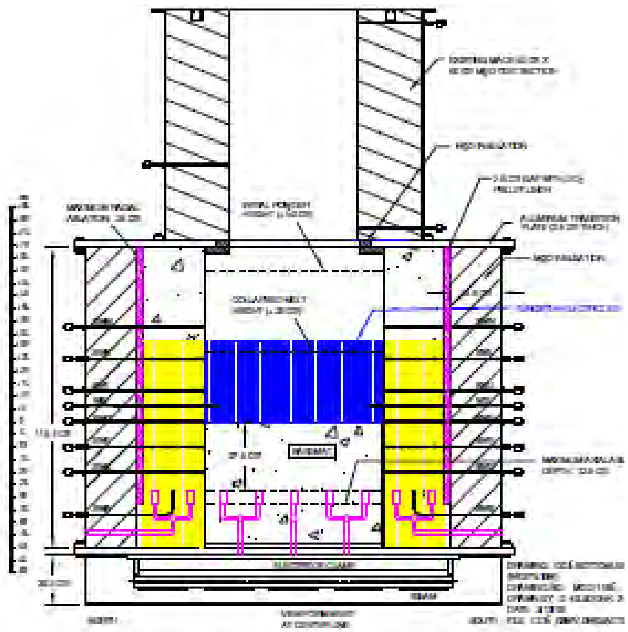
SA-7: Melt Coolability and Concrete Interaction-Phase 2 (MCCI-2) Project

After successful completion of the MCCI Project at Argonne National Laboratory (ANL), a second project using the same ANL facilities was set up.

Argonne National Laboratory facilities, United States



SSWICS test section



CCI with flooding test section

Source: OECD/NEA MCCI-2 Project.

The MCCI-2 Programme was carried out from 2006 to early 2010 to help bridge data gaps not fully covered in MCCI-1. Testing falls into four categories:

- combined effect tests to investigate the interplay of different cooling mechanisms and to provide data for model development and code assessment;
- tests to investigate the effectiveness of new design features that enhance debris coolability;
- tests to generate additional 2-D core-concrete interaction data for model development and code validation;
- integral test at larger scale to confirm synergistic effect of different cooling mechanisms and to provide data for validation of severe accident codes.

Aside from these tests, a supporting analysis task was carried out to further develop/validate debris coolability models that form the basis for extrapolating the experiment findings to plant conditions. In total, ten tests were conducted in this programme (all successful).

Four category 1 tests were performed using the small scale water ingression and crust strength (SSWICS) apparatus. Tests were conducted to provide additional crust strength data to confirm the concept of a floating crust boundary condition at plant scale and to investigate the effect of gas sparging on water ingression cooling of corium. Crust strength tests (2) showed that the strength of un-sectioned crust samples was consistent with that of the sectioned specimens tested in MCCI-1. Gas sparging tests (2) showed that the presence of sparging significantly increases the cooling rate of a solidifying corium pool over that observed when sparging is absent.

Category 2 tests to examine the effectiveness of design features for augmenting coolability, i.e. melt stabilisationconcepts, were of two cooling types:

- water-cooled basemat test (WCB-1) addressed cooling with external water-cooled surface(s);
- SSWICS-12 and -13 tests addressed water ingress into the melt volume by fragmentation.

Category 3 tests were to provide additional 2-D core-concrete interaction data. CCI tests in both the MCCI and French VULCANO facilities have shown a marked dependence of cavity erosion behaviour on concrete type. Tests with limestone/common sand (LCS) concrete generally exhibit a radial/axial power split of ~1; conversely, siliceous tests exhibit splits that are significantly greater than 1. CCI-4 conducted with LCS concrete, but with increased metal content (structural + cladding) to evaluate effect on cavity erosion behaviour. CCI-5 conduced with siliceous concrete, but the apparatus was modified to increase lateral scale to diminish wall effects to the greatest extent possible. Test aspect ratio (cavity width/melt depth) increased from 1 to 3.7.

Category 4 was an integral test to validate severe accident codes. The large scale CCI-6 test was conducted with early flooding to focus on debris coolability. Key features were:

- 70 cm x 70 cm cross section, 28 cm melt depth: 900 kg (63/25/6/6 wt% UO2/ZrO2/Cr/concrete);
- siliceous concrete;
- cavity flooded ~1 minute after melt contact with basemat.

Design incorporated an embedded array of water injection nozzles at a depth of 27.5 cm into the concrete. If debris did not quench, then a second test phase would be initiated to provide additional (category 2) data on bottom water injection cooling. Results demonstrated that:

- early cavity flooding significantly enhances debris coolability, even for siliceous concrete; and
- melt eruptions are a viable cooling mechanism for siliceous concrete. Test terminated on the basis of debris quench well before water injection nozzles were reached.

A concluding seminar of the MCCI-2 Project was held in Cadarache, France, from 15 to 17 November 2010.

These tests have provided a significant understanding and database for reducing uncertainties related to two-dimensional molten core-concrete interactions. In plant accidents, a significant melt metal fraction may be present that may result in a stratified pool configuration. This type of pool structure was not evaluated in this programme. Thus, additional analysis and testing may be required with melts containing a significant metal fraction to further reduce phemenological uncertainties related core-concrete interaction, and to evaluate the effects of melt metal content on debris coolability.

Final report

"OECD MCCI-2 Project Final Report", M.T. Farmer, S. Lomperski, D.J. Kilsdonk, and R.W. Aeschlimann, MCCI-2010-TR07, ANL, November 2010.

Participants

Belgium, Czech Republic, Finland, France, Germany, Hungary, Japan, Norway, Republic of Korea, Spain, Sweden, Switzerland, United States.

Project period: April 2006 to March 2010.

Budget: USD 1.1 million/year.

SA-8: Behaviour of Iodine (BIP) Project

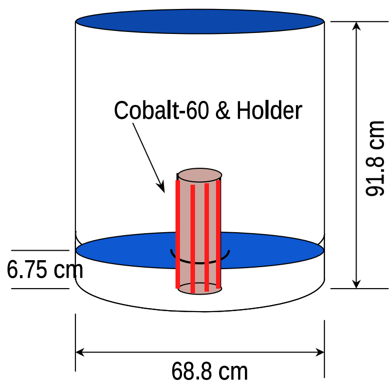
The Behaviour of Iodine (BIP) Project was created to provide separate effects and modelling studies of iodine behaviour in a nuclear reactor containment building following a severe accident. The NEA joint project was designed to complement other national and international experimental programmes which were also studying this phenomenon. As part of the project the results of three radioiodine test facility (RTF) experiments would be provided by Atomic Energy of Canada Limited (AECL).

Project participants have sought to combine international resources to produce a consolidated understanding of the behaviour of iodine and other fission products in this scenario.

The specific technical objectives for this project are:

- quantifying 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;
- measuring adsorption/desorption rate constants on containment surfaces as a function of temperature, relative humidity and carrier-gas composition;
- providing RTF data to participants, for use in collaborative model development and validation.

AECL iodine facilities, Canada



RTF Facility

Deposition coupons after irradiation

Source: OECD/NEA BIP Project.

It is expected that the results provided by this project will be useful for regulators and operators in the context of managing post-accident situations in the containment building. A follow-up of this project was started in April 2011 for a three year duration that is focusing on interaction between iodine and paint coatings and organic iodine formation in accident conditions.

Final report

Issued in December 2011.

Participants

Belgium, Canada, Finland, France, Germany, Japan, Netherlands, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

Project period: July 2007 to March 2011. Followed-up by the BIP-2 Project.

Budget: CAD 1.5 million.

SA-9: Behaviour of Iodine-Phase 2 (BIP-2) Project

The Behaviour of Iodine-Phase 2 (BIP-2) Project is a follow-up to the BIP Project that has provided results from separate effects and modelling studies of iodine behaviour in a nuclear reactor containment building following a severe accident. AECL's Chalk River Laboratories in Canada will be the primary research facility for the project. Key facilities include Co-60 irradiators (Gammacells), active laboratories for performing ^{131}I tracer studies, specialised surface science laboratories (capable of working on active or inactive samples) and extensive hot cell facilities. With these facilities, and within the framework of the OECD BIP Project, the group has successfully performed more than 50 tests dealing with the adsorption of iodine on surfaces and the formation of organic iodides from irradiated paint.

This new NEA joint project will complement other national and international experimental programmes such as SARNET2 which are also studying this phenomenon. Project participants will seek to combine international resources to produce a consolidated understanding of the behaviour of iodine and other fission products in this scenario.

The specific technical objectives for this follow-up project are:

- to obtain a more detailed and mechanistic understanding of iodine adsorption/desorption on containment surfaces by means of new experiments with well characterised containment paints and paint constituents and novel instrumentation (spectroscopic methods);
- to obtain a more detailed and mechanistic understanding of organic iodide formation by means of new experiments with well characterised containment paints and paint constituents and novel instrumentation (chromatographic methods);
- to develop a common understanding on how to extrapolate confidently from small scale studies to reactor scale conditions.

It is expected that an extensive analytical effort will accompany the experimental programme, mainly consisting of data interpretation, result evaluations and modelling for application to reactor situations.

Participants

The project is being supported by safety organisations, research laboratories and industry in the following countries: Belgium, Canada, Finland, France, Germany, Japan, Spain, Sweden, United Kingdom, United States.

Project period: April 2011 to March 2014.

Budget: EUR 0.9 million.

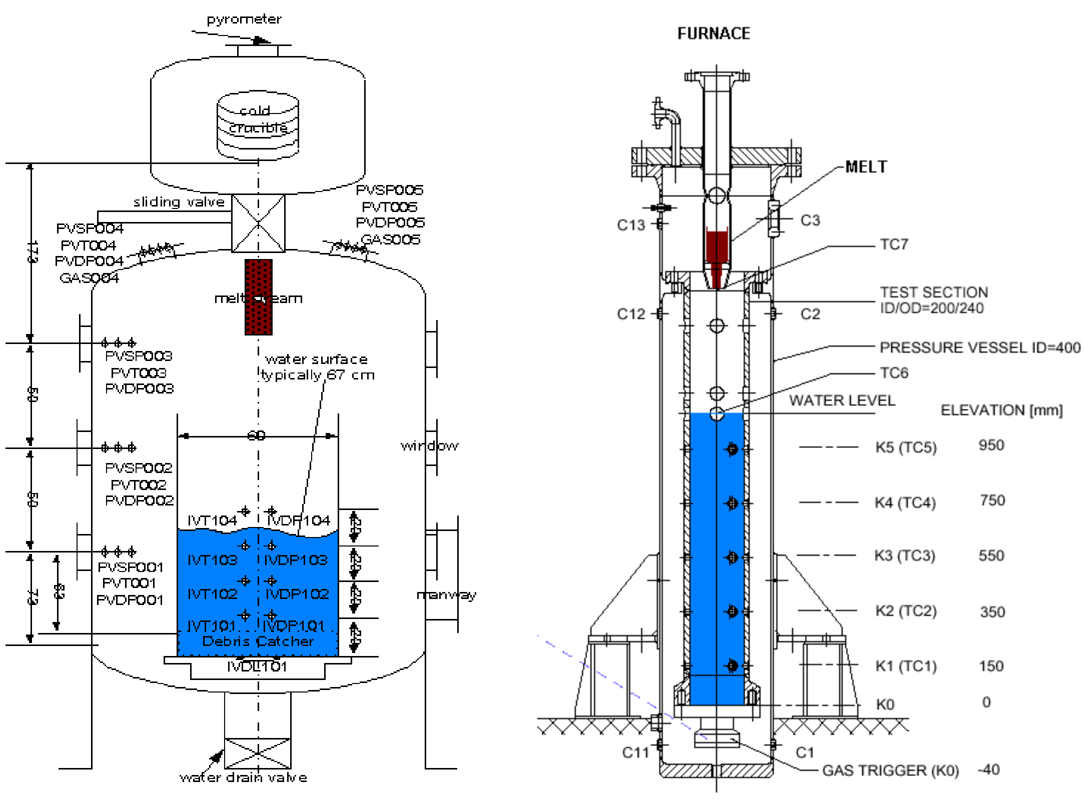
SA-10: Steam Explosion Resolution for Nuclear Applications (SERENA) Project

The SERENA Project was established to assess the capabilities of the current generation of fuel-coolant interaction (FCI) computer codes to predict steam explosion-induced loads in reactor situations. One of the main findings of this programme was that in-vessel FCI would not challenge the integrity of the nuclear reactor containment but that this scenario could not be ruled out for ex-vessel FCI.

The SERENA Project has therefore been set up to resolve any uncertainties on these issues. It is planned to achieve this by performing a limited number of focused tests with advanced instrumentation to examine a large spectrum of ex-vessel melt compositions and conditions. In parallel, analytical work will be carried out to bring the code capabilities to a sufficient level that they may be used in reactor case analyses. The objective of the experimental programme is threefold:

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

TROI facility, Korea/KROTOS facility, France



TROI KROTOS

Source: OECD/NEA SERENA Project.

These goals will be achieved by using the complementary features of the TROI (KAERI) and KROTOS (CEA) corium facilities, including fitness-for-purpose oriented analytical activities. KROTOS is more suited for investigating the intrinsic FCI characteristics in a one-dimensional geometry. TROI is more 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 using 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 overall programme was discussed throughout 2008, including ways to support the experimental projects with analyses and code assessments.

Final report

Scheduled for 2012.

Participants

Belgium, Canada, Finland, France, Germany, Japan, Republic of Korea, Slovenia, Sweden, Switzerland, United States.

Project period: September 2008 to March 2012.

Budget: EUR 2.6 million.

SA-11: Thermal-hydraulics, Hydrogen, Aerosols, Iodine (THAI) Project

The THAI Project was operated from January 2007 to December 2009 by Becker Technologies in its THAI facility in Eschborn, Germany. The objective was to assess the uncertainties related to the distribution of combustible hydrogen and to the behaviour of fission products, in particular iodine and aerosols in severe accident conditions. In the case of hydrogen, uncertainties emerge mainly in the determination of conditions for the occurrence of deflagration flames and in the performance of devices designed to reduce the concentration of hydrogen gas developed in a hypothetical accident, such as passive autocatalytic recombiners. Some concern also prevails 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 set up reliable transport models. Such processes include the exchange of iodine between a turbulent atmosphere and the walls, relocation by wash-down, i.e. washing of walls by 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 radioactivity management.

The THAI facility, Eschborn, Germany

Source: OECD/NEA THAI Project.

The experiments of the THAI Project were designed to fill these knowledge gaps by delivering data for the evaluation and simulation of the hydrogen and fission product interactions mentioned above, thereby supporting the validation of accident simulation codes and models. The experiments were conducted in the THAI facility in Frankfurt, the name of which is an acronym for "Thermal-hydraulics, Hydrogen, Aerosols, Iodine". Within the THAI Project, the following test series have been performed:

- helium/hydrogen material scaling (HM);
- hydrogen deflagration (HD);
- hydrogen recombiner (HR);

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- interaction of metal iodides with passive autocatalytic recombiner;
- passive autocatalytic recombiner poisoning;
- aerosol wash-down (scoping test)

Altogether 70 tests have been performed between January 2007 and December 2009. In addition to the experimental work, an analytical workgroup was established in the frame of the THAI Project, aimed at the evaluation of the test results for further development and validation of the predictive capabilities of advanced LP codes and CFD codes currently in use in the reactor safety field. For this purpose, a number of experiments have been selected for blind and open post-test calculations, some of them for an ISP.

A concluding seminar on the main outcome of the THAI Project will be organised on 6-7 October 2010. The database developed in the frame of this project has been and will be extensively used to validate and improve lumped parameter and CFD tools available and under development for containment analysis. The investigations have filled existing knowledge gaps to a large extent. As an example, in case of the PAR investigations, an almost complete picture of their performance under typical accident conditions has been obtained. The results of the experimental programme have provided assurance of PAR performance under realistic conditions. The remaining open questions regarding hydrogen combustion under spray and low oxygen PAR behaviour are expected to be addressed in future investigations. After completion of this project, an OECD THAI-2 Project was envisaged to be started to the follow-up this programme.

Final report

"OECD/NEA THAI Project Final Report – Hydrogen and Fission Product Issues Relevant for Containment Safety Assessment under Severe Accident Conditions", NEA/CSNI/R(2010)3, December 2010.

Participants

Canada, Finland, France, Germany, Hungary, Republic of Korea, the Netherlands and Switzerland.

Project period: January 2007 to December 2009.

Budget: EUR 2.8 million.

SA-12: Thermal-hydraulics, Hydrogen, Aerosols, Iodine-Phase 2 (THAI-2) Project

The objective of this THAI-2 follow-up project is to address remaining questions and to provide experimental data relevant to HTGR graphite dust transport issues, specific water-cooled reactors aerosol and iodine issues and hydrogen mitigation under accidental conditions. The project addresses open questions concerning the behaviour of: a) graphite dust transport in a generic HTGR geometry, b) release of gaseous iodine from a flashing jet, iodine deposition on aerosol particles, c) hydrogen combustion during spray operation and passive autocatalytic recombiner operation in case of extremely low oxygen content. The understanding of the respective processes is essential for evaluating the challenge posed on next generation reactors (HTGR), for evaluating the amount of airborne radioactivity during accidents with core damage (iodine and aerosols) and for containment integrity (hydrogen). The programme will generate valuable data for evaluating atmospheric flows and subsequently graphite dust transport in a generic multi-compartment geometry. Concerning fission products the programme will focus on iodine release from a flashing jet and gaseous iodine deposition on aerosols. Regarding hydrogen mitigation the programme will focus on its combustion during spray operation and on its effective removal by means of passive autocatalytic recombiners when approaching oxygen starvation. An analytical effort will accompany the experimental programme, mainly consisting of code calculations for pre-test assessments, result evaluations and extrapolation to reactor situations.

Participants

Canada, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Netherlands, Sweden, United Kingdom.

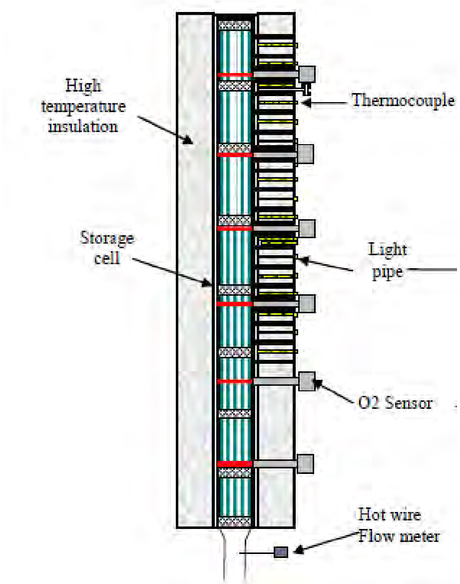
Project period: July 2011-June 2014.

Budget: EUR 3.6 million.

SA-13: Sandia Fuel (SFP) Project

The goal of the Sandia Fuel Project was to provide experimental data relevant for hydraulic and ignition phenomena of prototypic water reactor fuel assemblies. The intended scope of work was defined in an experts' meeting organised by the OECD/NEA. The project content includes 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 in representative fuel configurations.

The SFP test facility, Sandia, United States



Source: OECD/NEA SFP Project.

The proposed experiments focus on thermal-hydraulic and ignition phenomena in pressurised water reactor (PWR) 17x17 assemblies and supplement earlier results obtained for boiling water reactor (BWR) assemblies, which were made available to the SFP participants. It is believed that 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 was scheduled to run for three years and to be conducted in two phases:

- Phase 1 focused on axial heating and burn propagation;
- Phase 2 addressing radial heating and burn propagation and including effects of fuel rod ballooning.

Final report

Scheduled for 2013.

Participants

Czech Republic, France, Germany, Hungary, Italy, Japan, Norway, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

Project period: July 2009-June 2012.

Budget: USD 5.2 million.

SA-14: Source Term Evaluation and Mitigation (STEM) Project

Past, recent and ongoing R&D programmes (e.g. Phebus FP, ISTP, ARTIST, OECD/BIP and THAI Projects…) are mainly focused on the reduction of uncertainties on the evaluation of the potential source term to the environment in case of a LWR severe accident. This effort is sustained within the EU/SARNET Network of Excellence. However, it has been recognised that known phenomenological uncertainties remain for some complex phenomena with a significant impact on source term. For public acceptance of lifetime extension of existing reactors, it must be demonstrated that their safety level continues to be enhanced.

Therefore a new OECD/NEA project named STEM (Source Term Evaluation and Mitigation), operated by IRSN in its facilities in Cadarache, France, was initiated in 2011 to improve the general evaluation of the source term. In addition the reduction of known phenomenological uncertainties on certain phenomena is expected to help:

- providing better information and tools to emergency teams in order to help them making a more robust diagnosis and prognosis of the progression of an accident and a better evaluation of potential release of radioactive materials;
- investigating phenomena involved in possible complementary mitigation measures, natural or engineered, so as to minimise releases to the environment.

STEM facility, Cadarache, France

Source: OECD/NEA STEM Project.

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The new STEM Project deals with three main issues:

- Radioactive iodine release in mid and long term

In complement to previous programmes, it is proposed to perform experiments to study the stability of aerosol particles under radiation and the long-term gas/deposits equilibrium in a containment.

- Interactions between iodine and paints

No experiments are planned but a literature survey especially focused on the effect of paint ageing that is likely to lead to the definition of experiments in a possible follow-up project.

- Ruthenium chemistry

In complement to previous programmes, it is proposed to perform experiments to study the Ruthenium transport in pipes.

Within the frame of the STEM Project, the first series of experiments (iodine behaviour under radiation) are dedicated to the analysis of radiation effects and will thus be realised in benches built on the EPICUR facility (Experimental Programme of Iodine Chemistry Under Radiation). The second series of experiments (ruthenium transport) will aim to analyse the chemistry of ruthenium in pipes including the reactor coolant system and in filters. They will be performed in dedicated benches allowing the injection of different chemical compounds followed by their transport through high temperature gradient tubes up to aerosols filters and bubblers for gas trapping

Participants

Canada, Czech Republic, Finland, France, Germany, Republic of Korea, United States.

Project period: July 2011-June 2015.

Budget: EUR 3.5 million.

SY-1: SCORPIO Project

Under an agreement with the Japanese Science and Technology (STA) and in co-operation with the OECD Halden Reactor Project, the NEA supported an international research project, the “SCORPIO” Project, to improve the surveillance capabilities for VVER reactor cores in various operational conditions.

The Norwegian *Institutt for Energiteknikk* (IFE), the operator of the OECD Halden Reactor Project, carried out the project under the supervision of an international committee set up by the NEA.

The SCORPIO VVER reactor core surveillance system was installed in the Dukovany nuclear power plant in the Czech Republic, for the purpose of developing a general system framework for VVER type reactors. The design phase started in April 1996. The overall structure of the man-machine interface was implemented and several pictures (screens) were made for testing purposes.

Final report

“VVER Core Surveillance System (SCORPIO) for Dukovany NPP (Target Plan)”, IFE, February 1999.

Project period: January 1996 to December 1998.

Budget: USD ~0.5 million.

SY-2: PLASMA Project

Under an agreement with the Japan Atomic Energy Research Institute (JAERI), the NEA supported an international research project aimed at developing a Plant Safety Monitoring and Assessment System, known as the PLASMA Project, for use with VVER nuclear reactors. This project was completed in the course of 2000.

The Norwegian *Institutt for Energiteknikk (IFE)*, the operator of the OECD Halden Reactor Project, carried out this joint research project with the Hungarian Academy of Sciences (KFKI), the Atomic Energy Research Institute of Hungary (AEKI), and PAKS Nuclear Power Plant Ltd, where the system was implemented and demonstrated.

The PLASMA system is intended to be generic in nature and thus applicable to various VVERs. In addition to the target installation in the PAKS nuclear power plant, a comprehensive user training programme was implemented.

Final report

"Plant Safety Monitoring and Assessment System (PLASMA) for the PAKS NPP (Target Plan)", IFE, December 2000.

Project period: December 1998 to December 2000.

Budget: USD ~0.5 million.

EDB-1: International Common-cause Failure Data Exchange (ICDE) Project

Common-cause failure (CCF) events can significantly affect the availability of nuclear power plant safety systems. In recognition of this, CCF data are systematically collected and analysed in several countries. A serious obstacle to the use of national qualitative and quantitative data collections by other countries is that the criteria and interpretations applied in the collection and analysis of events and data differ. A further impediment is that descriptions of reported events and their root causes, which are important to the assessment of the events, are usually written in the native language of the countries where the events were observed.

To overcome these obstacles, the preparation for the International Common-cause Data Exchange (ICDE) project was initiated in August of 1994. Since April 1998, the NEA has formally operated the project. The objectives of the ICDE Project are to:

- collect and analyse CCF events over the long term so as to better understand such events, their causes, and their prevention;
- generate qualitative insights into the root causes of CCF events which 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 CCF phenomena, including the development of defences against their occurrence, such as indicators for risk based inspections;
- generate quantitative insights and record event attributes to facilitate quantification of CCF frequencies in member countries; and
- use the ICDE data to estimate CCF parameters.

The ICDE project operates with a clear separation between data collection and analysis. The data collection and analysis firstly results in qualitative CCF information that can be used for the assessment of the effectiveness of defences against CCF events and the importance of CCF events in the probabilistic safety assessment framework. The qualitative insights on CCF events generated by the analysis are being made available to CSNI countries through published reports. Data collection guidelines have been developed during the project and are continually revised. They describe the methods and documentation standards necessary for the development of the ICDE databases and reports.

The format for data collection is described in the general coding guidelines and in the component-specific guidelines. Component-specific guidelines are developed for all analysed component types as the ICDE project evolves. The ICDE prepares reports containing conclusions on the analysis performed whenever major steps of the project have been completed. The intention is to make the project results known to a larger audience.

Data analysis and exchange covers for the time being the following components:

- centrifugal pumps;
- diesel generators;
- motor-operated valves;
- safety and relief valves;
- check valves;
- batteries;
- reactor protection system components (level measurement, control rod drives, etc.);

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- circuit breakers;
- heat exchangers;
- Mid-2011 several thousands of event records were in the database.

Reports

CSNI reports have been produced for pumps, diesel generators, motor-operated valves, safety and relief valves, check valves and batteries. Full list in "ICDE Project Report: Collection and Analysis of Common-Cause Failures of Level Measurement Components – June 2008", NEA/CSNI/R(2008)8.

Participants

Canada, Finland, France, Germany, Japan, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, United States.

Project period: Initiated in 1994.

Budget: EUR ~0.12 million/year.

EDB-2: Piping Failure Data Exchange (OPDE) Project

The goals of the Piping Failure Data Exchange (OPDE) Project 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 influence factors to facilitate estimation of piping failure frequencies.

The OPDE Project was envisaged to include all possible events of interest with regard to piping failures. It will cover piping components of the main safety systems (e.g. ASME Code Class 1, 2 and 3). It also covered non-safety piping systems that, if leaking, could lead to common-cause initiating events such as internal flooding of vital plant areas. As an example, raw water systems such as non-essential service water could be a significant flood source given a pipe break. Steam generator tubes are excluded from the OPDE project scope. Mid-2011 several thousands of event records were in the database.

Report

"OECD/NEA Pipe Failure Data Exchange (OPDE) Project – (2002-2008) Status Report", NEA/CSNI/R(2009)19.

Participants

Canada, Czech Republic, Finland, France, Germany, Japan, Republic of Korea, Spain, Sweden, Switzerland, United States.

Project period: June 2002-June 2011. To be followed-up by the COPDAP Project (2011-2014).

Budget: EUR ~60 000/year.

EDB-3: Fire Incidents Records Exchange (FIRE) Project

The main purpose of the project is to encourage multilateral co-operation in the collection and analysis of data relating to fire events. The objectives of the FIRE Project are to:

- collect fire event experience (by international exchange) in an appropriate format in a quality-assured and consistent database;
- collect and analyse fire events over the long term so as to better understand such events and their causes, and to encourage their prevention;
- generate qualitative insights into the root causes of fire events in order to derive approaches or mechanisms for their prevention and to mitigate their consequences;
- establish a mechanism for efficient operation feedback on fire event experience including the development of policies of prevention, such as indicators for risk-informed and performance-based inspections; and
- record characteristics of fire events in order to facilitate fire risk analysis, including quantification of fire frequencies.

Coding guidelines and a quality assurance manual have been developed and validated by the project's participants. The project participants have set up structures within their country to collect and validate data for the project, which is now widely seen as the reference international database for fire events.

After having established the project quality guidelines and the quality-assurance procedure, data acquisition has proceeded according to plan. An updated version of the database is provided to all participants every year. Mid-2011 the event database contains more than 350 events.

Report

"Collection and Analysis of Fire Events (2002-2008) – First Applications and Expected Further Developments", NEA/CSNI/R(2009)6, September 2009.

Participants

Canada, Czech Republic, Finland, France, Germany, Japan, Republic of Korea, Sweden, Switzerland, Spain, Netherlands, United States.

Project period: Initiated in January 2003.

Budget: EUR ~80 000/year.

EDB-4: Computer-based Systems Important to Safety (COMPSIS) Project

Software-based systems are currently being used and retrofitted in operating nuclear power plants worldwide. The failure modes of both hardware and software in these systems are to some extent different from the analogous I&C systems. At present, there is no established international database where the failure modes of computerised systems are collected. The general aim of the COMPSIS (Computer-based Systems Important to Safety) Project is to exchange information on computer-based system reliability in a structured way. The high-level objective is to contribute to the improvement of safety management and to the quality of software risk analysis for software-based equipment. Software and hardware faults in safety-critical systems are typically rare events and, consequently, most countries do not experience enough faults to allow meaningful syntheses. Combined information from several countries, however, is expected to yield sufficient data to help draw conclusions.

The main objectives of the COMPSIS project are to:

a) define a format and collect software and hardware fault experience in computer-based safety critical NPP systems (hereafter called “COMPSIS events”) in a structured, quality-assured and consistent database;

b) collect and analyse COMPSIS events over a long period so as to better understand such events, their causes and their prevention;

c) generate insights into the root causes of and contributors to COMPSIS events, which can then be used to derive approaches or mechanisms for their prevention or for mitigating their consequences;

d) establish a mechanism for an efficient feedback of experience gained in connection with COMPSIS events, including the development of defences against their occurrence, such as diagnostics, tests and inspections;

e) record event attributes and dominant contributors so that a basis for national risk analysis of computerised systems is established.

Work during the first part of the project concentrated on the development of the COMPSIS data collection guidelines, quality assurance and data exchange interface. End 2011 about 90 licensee event report (LER) events have been reported.

Report

“Computer-based Systems Important to Safety (COMPSIS) Project: 3 Years of Operation (2005-2007)”, NEA/CSNI/R(2008)13.

Participants

Finland, Germany, Hungary, Republic of Korea, Sweden, Switzerland, United States.

Project period: January 2006-December 2011.

Budget: EUR ~80 000/year.

EDB-5: Stress Corrosion Cracking and Cable Ageing Project (SCAP)

The Stress Corrosion Cracking and Cable Ageing Project (SCAP) began in 2006. 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 systematically compiling and evaluating collected data and information;
- 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 defining and refining the database performance requirements, data format and coding guidelines, and is currently focusing on populating the database and assessing the data.

The database, together with the knowledge base and the commendable practices which was developed by the members, will provide a tool for assisting the member countries. The final report was published at the end of the project and provides the technical basis for commendable practices in support of regulatory activities in the fields of SCC and cable insulation.

Final report

"Technical Basis for Commendable Practices on Ageing Management – Final Report of the Stress Corrosion Cracking and Cable Ageing Project (SCAP)", NEA/CSNI/R(2010)15, January 2011.

Participants

Argentina, Belgium, Canada, Czech Republic, Finland, France, Germany, Japan, Mexico, Norway, Republic of Korea, Slovak Republic, Spain, Sweden, Switzerland, Ukraine, United States.

Project period

June 2006-June 2010. To be followed-up by the COPDAP Project (2011-2014) for the stress corrosion cracking part and by the CADAK Project (2012-2015) for the cable ageing part.

Budget: EUR ~0.1 million/year.

EDB-6: Component Operational Experience, Degradation and Ageing Programme (CODAP)

The Component Operational Experience, Degradation and Ageing Programme (CODAP) aims to combine the follow-up of two previous OECD projects: the OECD Pipe Failure Data Exchange Project (OPDE) and the stress corrosion cracking (SCC) part of the SCC and Cable Ageing Project (SCAP).

The OECD Pipe Failure Data Exchange Project (OPDE) was established in May 2002 to produce an international database on the piping service experience applicable to commercial nuclear plants. OPDE which is operated under the umbrella of the NEA has been run successfully for eight years and has collected more than 3 700 piping events. Currently 11 countries have signed the OPDE agreement for the third term June 2008-May 2011 (Canada, Czech Republic, Finland, France, Germany, Republic of Korea, Japan, Spain, Sweden, Switzerland and the United States). The OPDE Project was completed at the end of May 2011.

In 2006 the SCAP Project was established by the NEA to assess two subjects, stress corrosion cracking (SCC) and degradation of cable insulation, due to their implication on nuclear safety and their relevance for plant ageing management. The project ran successfully from June 2006 to June 2010. Fourteen NEA member countries joined the SCAP Project in 2006 to share knowledge and by 2010 17 countries had joined the project. The International Atomic Energy Agency (IAEA) and the European Commission also participated as observers.

Following the completion of the SCAP Project, SCC participants were interested in some form of continuation and discussions were initiated to explore possible alternatives. It was recognised that there are many similarities between the OPDE and SCAP SCC Projects and therefore, the concept of a new project was envisaged to combine the two projects into a new project called: OECD Component Operational Experience, Degradation and Ageing Programme (CODAP).

Building on the success of OPDE and SCAP SCC, the objectives of the proposed CODAP Project are to:

- Collect information on passive metallic component degradation and failures of the primary system, reactor pressure vessel internals, main process and standby safety systems, and support systems (i.e. ASME Code Class 1, 2 and 3, or equivalent), as well as non safety-related (non-code) components with significant operational impact.
- Establish a knowledge base for general information on component and degradation mechanisms such as applicable regulations, codes and standards, bibliography and references, R&D programmes and pro-active actions, information on key parameters, models, thresholds and kinetics, fitness for service criteria, and information on mitigation, monitoring, surveillance, diagnostics, repair and replacement.
- Develop topical reports on degradation mechanisms in close co-ordination with the CSNI Integrity and Ageing of Components and Structures Working Group (WGIAGE).

Participants

Canada, Czech Republic, Finland, France, Germany, Republic of Korea, Japan, Spain, Sweden, Switzerland, United States.

Project Period: June 2011-December 2014.

Budget: EUR 0.12 million/year.

EDB-7: Cable Ageing Data and Knowledge (CADAK) Project

In 2006 two subjects – stress corrosion cracking (SCC) and degradation of cable insulation – were selected as the focus of the SCC and Cable Ageing Project (SCAP), funded by Japan and operated from 2006 to 2010, due to their relevance for plant ageing assessments and their implications on nuclear safety. In December 2010 CSNI agreed to support the two SCAP follow-up activities, that being the convening of an experts meeting on the cable database and secondly, the convening of an experts meeting on the merging of the SCC and OPDE database projects [NEA/SEN/SIN(2010)2, item 33] to become the CODAP (Component Operational Experience, Degradation and Ageing Programme) Project [NEA/NE(2011)5].

The CADAK (Cable Ageing Data and Knowledge) Project will follow up the cable ageing part of the SCAP Project. CADAK aims to establish the technical basis for assessing the qualified life of electrical cables in light of the uncertainties identified following the initial (early) qualification testing. This research will investigate the adequacy of the margins and their ability to address the uncertainties.

As result of the first CADAK expert meeting the following objectives with four main topics has been drafted:

Completion and maintenance

The content (executive summary) of the final SCAP report, the corresponding IAEA report on cable ageing, as well as the NUREG/CR-7000 should be transferred to the different parts of the data and knowledge base.

For a number of member countries cable data and information needs to be edited into the system e.g. technical standards being applied in the qualification of cables and inspection methods being used regularly. The SCAP Cable Working Group has created an encyclopaedia on cables useful for both novice and experienced NPP regulators and operators which will be used for this project.

Performance monitoring methods

For cable ageing, the crucial point is the knowledge about the qualification procedure for harsh environments and the predictive capability to estimate the remaining qualified lifetime. The cable condition-monitoring techniques shared by the participants within SCAP have become an up-to-date encyclopaedic source to monitor and predict the performance of every unique application of cables.

Research activities are going on in different countries to improve these methods. An extensive info exchange will be beneficial in this technical area. Further on some of the national research activities may be used for international benchmark activities.

Commendable practices

Information supporting Ageing Management Programmes (AMPs) is of vital importance given that ageing management is an essential and important aspect to be taken into consideration in connection with safe long-term operation (LTO). The use of the accumulated knowledge specifically in the area of equipment qualification (EQ) and condition monitoring can form the basis for commendable practice documents – like a CSNI Technical Opinion Paper – being generated as part of the CADAK Project. Such documents will help regulators and operators to enhance ageing management.

Extension to other technical items

The SCAP Project has yielded crucial knowledge in the area of cable ageing related to qualification procedure for harsh environments, as well as assessing cable degradation. This expertise could also be used to include other technical equipment like cable penetration, pressure/level transmitters, etc., that have common elements with many countries to the data and knowledge base.

Participants

Belgium, Canada, France, Japan, Spain, United States.

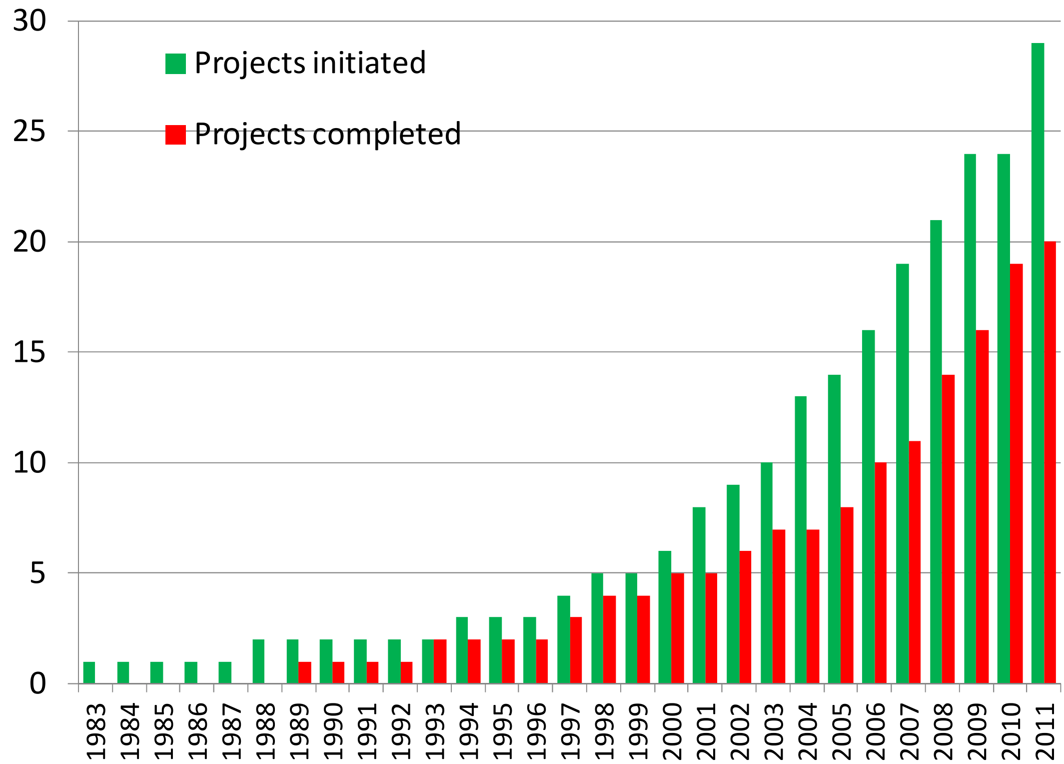
Project period: December 2011-December 2014.

Budget: EUR 120 000/year.

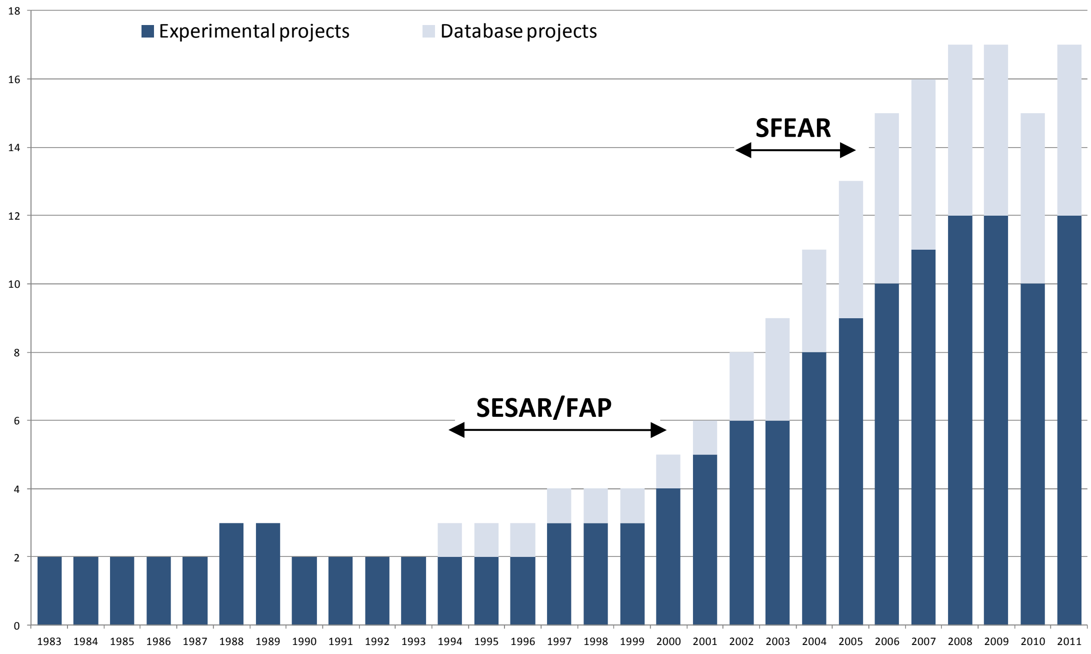
Appendix 2:
Some statistics about OECD/NEA nuclear safety joint projects

The figures hereafter provide some statistical features on the NEA joint project over the last 30 years (before 1983 only the Halden Project was running, related to nuclear safety) regarding the number of project initiated, running and closed, the technical areas concerned, the budgets, the participation and the involvement of countries.

Evolution of the number of NEA joint projects initiated and terminated over 30 years

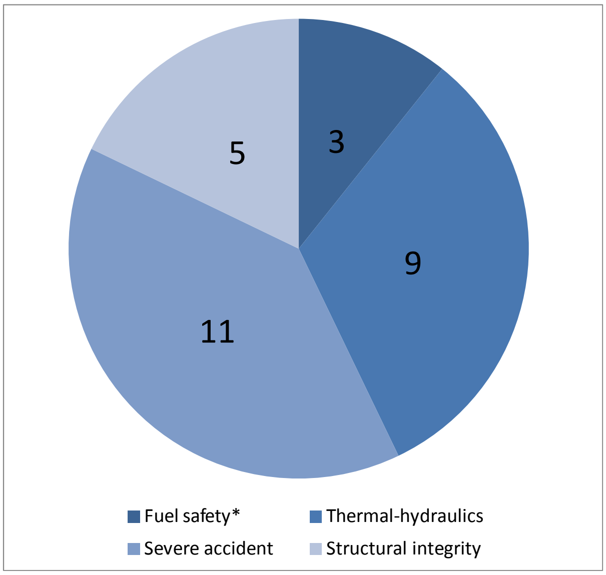


Evolution of the number of ongoing NEA joint projects between 1982 and 2011

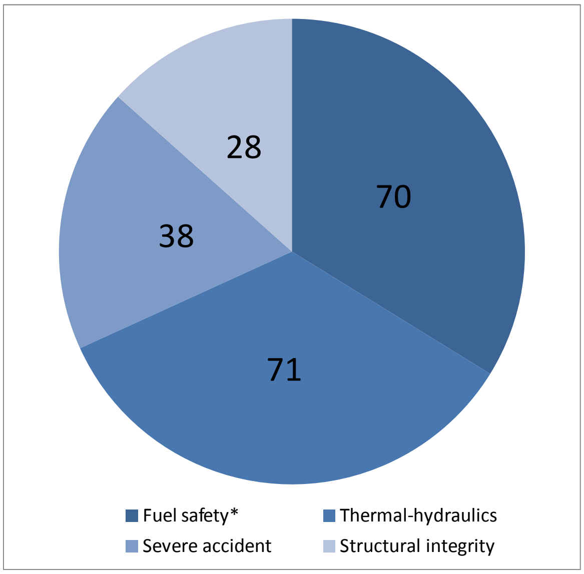


Areas addressed by NEA joint projects over 30 years and indication of budget

Number of projects



Budget (M EUR)



* Halden Project and "system" projects not included.

Some key numbers related to joint projects over 30 years (duration, cost and participants)

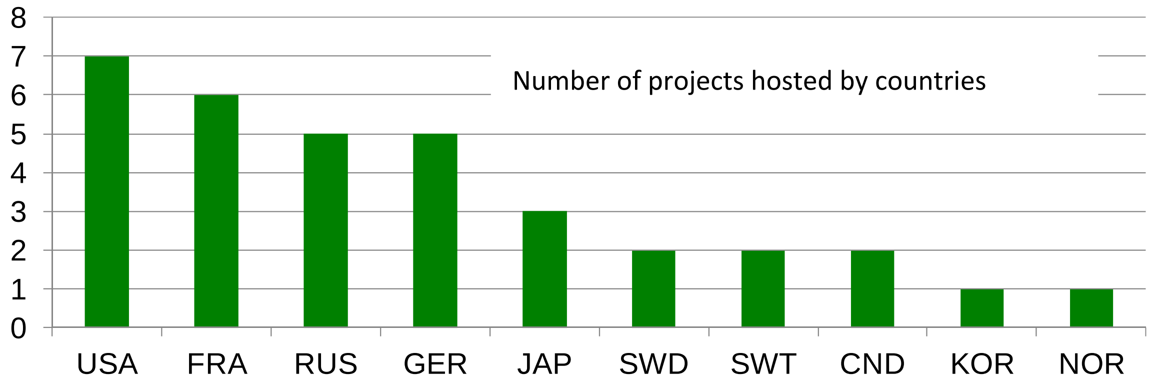
Project duration*	Years
minimum	3
average	4
maximum	6

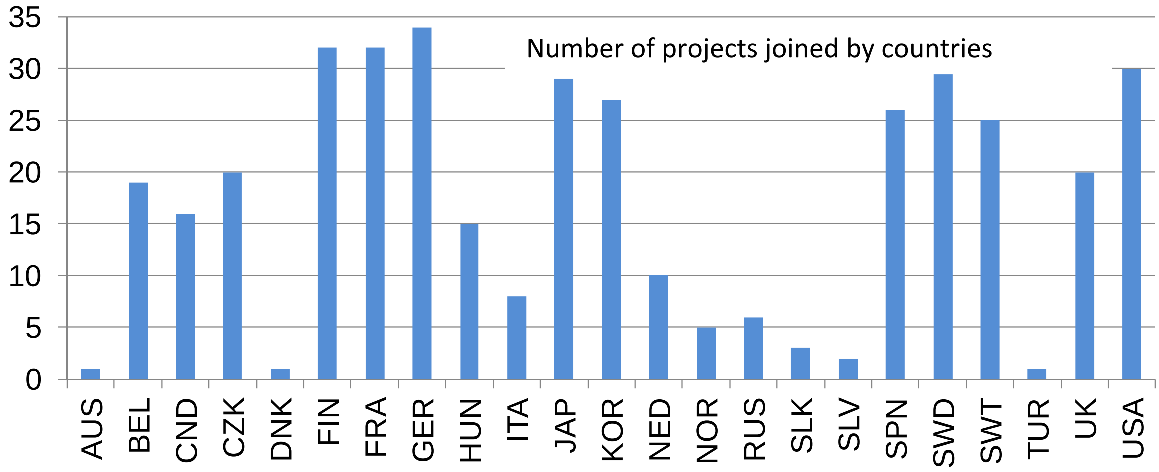
Project budget*	EUR M
minimum	~1
average	~4
maximum	~9

Participants	Total	Safety organisations	Research organisations	Industry
minimum	8	2	2	0
average	15	7	6	2
maximum	23	11	11	9

* Outside Halden Project, LOFT Project and Cabri Project.

Involvement of NEA countries in joint projects between 1982 and 2011





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Main Benefits from 30 Years of Joint Projects in Nuclear Safety

One of the major achievements of the OECD Nuclear Energy Agency (NEA) is the knowledge it has helped to generate through the organisation of joint international research projects. Such projects, primarily in the areas of nuclear safety and radioactive waste management, enable interested countries, on a cost-sharing basis, to pursue research or the sharing of data with respect to particular areas or issues. Over the years, more than 30 joint projects have been conducted with wide participation of member countries.

The purpose of this report is to describe the achievements of the OECD/NEA joint projects on nuclear safety research that have been carried out over the past three decades, with a particular focus on thermal-hydraulics, fuel behaviour and severe accidents. It shows that the resolution of specific safety issues in these areas has greatly benefited from the joint projects' activities and results. It also highlights the added value of international co-operation for maintaining unique experimental infrastructure, preserving skills and generating new knowledge.

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