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Research and Development Needs for Current and Future Nuclear Energy Systems

NEA No. 4453

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities 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 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.

FOREWORD

In 2001, the OECD/NEA Nuclear Science Committee (NSC) initiated a study on research and development (R&D) needs in nuclear science. The initial phase of this study comprised a review of the outcome of past and ongoing studies performed under the aegis of the NSC. This phase was then followed by a workshop on R&D Needs for Current and Future Nuclear Systems, held in Paris, France on 6-8 November 2002.

The workshop was considered timely for two reasons. The first was the recent publication of a number of documents related to future nuclear systems, including accelerators for the transmutation of nuclear waste and the Generation IV initiative looking at next-generation power reactors and related R&D needs. The workshop reviewed both of these areas of activity. The second reason was that the NEA was preparing a new five-year strategic plan for 2004-09, and the findings of the workshop would provide useful input into the nuclear science part of that plan.

The theme of the workshop – R&D needs in nuclear science – was also of interest to the NEA Committee on the Safety of Nuclear Installations (CSNI) and the NEA Nuclear Development Committee (NDC). Both of these committees had relevant ongoing activities, which were presented at the workshop.

During the workshop, past and ongoing NSC activities were presented, as well as research programmes in NEA member countries and international organisations on new nuclear reactor concepts. Following these presentations, participants discussed the need for future R&D initiatives in areas of nuclear science of relevance to NSC work. A set of recommendations to the NSC were issued and can be found in the "Conclusions and Recommendations" chapter of this report.

This report presents the results of the NSC study on R&D needs in nuclear science. It will be of particular interest to heads of nuclear science and nuclear research programmes, as well as to those interested in future nuclear energy systems.

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EXECUTIVE SUMMARY

Continuing programmes of nuclear research and development (R&D) are essential for maintaining the safe and efficient operation of existing nuclear power plants and fuel cycle facilities, and also for ensuring the emergence of new advanced and innovative nuclear energy systems in the future. To enable nuclear power to play an important role in meeting future energy requirements in a sustainable manner, it is essential to identify R&D needs and to ensure that adequate expertise and resources are available to meet them.

There has been, however, a significant decline in nuclear R&D expenditures in recent years in many NEA member countries, resulting in the loss of facilities and expertise. This has led to concerns about maintaining an adequate R&D infrastructure to meet future needs. In order to maximise the efficiency of R&D programmes and to make the most effective use of limited resources, increased international co-operation and co-ordination are essential. Encouraging and facilitating such international co-operation is an important part of the NEA's role.

In order to examine these issues, the Workshop on R&D Needs for Current and Future Nuclear Systems was organised by the NEA's Nuclear Science Committee (NSC), in close co-ordination with the Committee on the Safety of Nuclear Installations (CSNI) and the Nuclear Development Committee (NDC). The workshop was held in Paris from 6-8 November 2002, and was attended by 33 participants from11 countries and three international organisations.

Altogether 15 presentations were made during the workshop. The first session comprised presentations providing an overview of past and continuing nuclear R&D activities in which the NEA was involved. These were divided into those relating to partitioning and transmutation (P&T), nuclear data, fuel behaviour, criticality safety, plutonium recycling and innovative fuel cycles, and reactor physics and shielding.

The NEA has been involved in P&T research since the late 1980s, when a long-term programme was initiated. The primary interest has been in accelerator-driven systems (ADS) for the incineration of long-lived radioactive waste. The programme is overseen jointly by the NSC and the NDC, which have established close co-operation in this area of work. In 1992 the NSC established the Task Force on Physics Aspects of Transmutation Concepts, the scope of which included examining the fundamental scientific issues related to proposed transmutation concepts. An overview report was issued in 1994 looking at over 20 such concepts and setting out follow-up actions.

To help in co-ordinating the NEA's work on P&T, in 2000 the NSC set up the Working Party on Scientific Issues in Partitioning and Transmutation (WPPT). The scope of its work includes accelerator utilisation and reliability, chemical partitioning, fuels and materials, and physics and safety. Sub-groups were established to examine each of these four areas. The WPPT is tasked with providing NEA member countries with up-to-date information on the feasibility and development status of P&T concepts, and to advise on R&D requirements. It is scheduled to produce a major report on the status of P&T in 2004.

There are a number of facilities in NEA member countries and elsewhere which are engaged in nuclear data measurements, although the overall effort has declined over recent years. International co-operation is important to ensure that the nuclear data work of the various facilities around the world is co-ordinated, to avoid duplication and to ensure that the required data are produced. The NEA, together with other organisations, has played an important role in fostering such co-operation over the years. The NEA Data Bank is part of an international network of data centres, each of which compiles and facilitates the evaluation of nuclear data.

The NSC Working Party on International Evaluation Co-operation (WPEC) was established in 1989 to review nuclear data evaluation activities. Specialist sub-groups are regularly established to examine problem areas. More than 20 short-term sub-groups have been set up to date, most of which have issued reports. Four long-term sub-groups have also been set up, covering nuclear model codes, formats and processing, nuclear data standards, and the High Priority Request List for nuclear data. The latter is an international collaborative effort through which data requests from different countries are complied and assessed. It contains close to 500 requests, and can be viewed on the NEA web site.

The need for international co-operation in improving scientific knowledge of nuclear fuel behaviour has long been recognised by the NSC. In 1995, a task force of the NSC produced a report on this topic which initiated a series of actions by the NSC over subsequent years. The report found that, while existing nuclear fuel was highly reliable in terms of both performance and safety, this was partly due to generous operating margins. To meet pressures to improve the economic efficiency of reactor operation, it would become increasingly important to predict fuel performance more precisely. This showed the need for good fuel performance computer codes supported by high-quality data.

The task force went on to carry out a review of available experimental data on fuel performance which could be used to improve the performance of computer codes. This led to the creation of the International Fuel Performance Experiment (IFPE) database, organised and maintained by the NEA Data Bank. It covers all commercially-operated thermal reactor systems, principally focusing on standard zircaloy-clad UO_2 fuel. The database is now well established and is widely used by organisations involved in code development. It contains well over 400 data sets, and additional data continue to be added.

A Criticality Computations Working Group was originally set up by the CSNI in 1980, later coming under the auspices of the NSC. The aim of the group was to examine the computational methods and data available for performing criticality safety assessments. The great diversity in the accuracy and quality of the results gave impetus to further work, leading to a series of criticality safety studies by the NEA. The Expert Group on Burn-up Credit was established in the early 1990s to develop methods for performing criticality calculations for spent fuel which took into account the actual condition of the fuel, potentially leading to significant savings in cost and storage space. It has completed several phases of work, making it the longest-running NEA expert group.

An international effort to collect data from nuclear criticality experiments performed at facilities around the world was begun by the US Department of Energy in 1992. This was transferred to the NEA in 1994 under the title International Criticality Safety Benchmark Evaluation Project (ICSBEP). The project has since grown rapidly and continues to make a great contribution to nuclear criticality safety. The data are published as the International Handbook of Evaluated Criticality Safety Benchmark Experiments.

Work on criticality safety, including the Expert Group on Burn-up Credit and the ICSBEP, is currently overseen by the NSC Working Party on Nuclear Criticality Safety. The working party has established a further five expert groups, on minimum critical values, experimental needs, source convergence analysis, criticality excursion analyses and sub-critical measurements.

The Working Party on Physics of Plutonium Recycling was set up by the NSC in 1992, later becoming the Working Party on Physics of Plutonium Recycling and Innovative Fuel Cycles (WPPR). The working party has since carried out several benchmark studies, covering recycling in PWRs, void reactivity effect in PWRs, fast plutonium-burning reactors, recycling in fast reactors, multiple plutonium recycling in standard and advanced PWRs and mixed oxide (MOX) fuel in BWRs.

These benchmark exercises have led to improved and newly developed computer codes, supporting the plans of several NEA member countries for increased use of MOX fuel. Further benchmark exercises have been proposed for the continuing work of the WPPR, including for high-temperature reactors and MOX core transients. Additional work is being carried out in co-operation with the Expert Group on Reactor-based Plutonium Disposition.

In the field of reactor physics, the NSC has undertaken, through various expert groups, a series of benchmark exercises covering computational methods for assessing the behaviour of reactor cores and components. An expert group was established to deal with modelling and computational methods in the field of three-dimensional coupled neutronics and thermal-hydraulics transients. These included LWR core transients, uncontrolled withdrawal of control rods in PWRs, PWR main steam line breaks, BWR stability analysis and turbine trips in BWRs. The aim was to develop techniques applicable to both existing reactors and new reactor concepts.

The NEA Data Bank is the lead organisation for the Shielding Integral Benchmark Archive Database (SINBAD), which plays an essential role in the validation and benchmarking of computer codes and data used for radiation transport and shielding calculations. With the increasing interest in accelerator systems, in 2000 the NSC set up an Expert Group on Shielding Aspects of Accelerators, Target and Irradiation Facilities (SATIF) to promote information exchange and international co-operation. A series of six SATIF specialist meetings has been held.

The maintenance of essential nuclear safety research capabilities and facilities has been of concern to the CSNI since the early 1990s. In 1992 it established the Senior Group of Experts on Safety Research (SESAR), bringing together senior managers of major research programmes in NEA member countries. The group reviewed existing research and examined future requirements and priorities. It identified three main drivers of research at that time, namely plant life management (including life extension), optimisation of operating margins and measures related to severe accidents.

The work of SESAR in identifying the key areas of safety research which need to be maintained has been followed up by the CSNI. A review group monitors the status of research infrastructure and compiles a list of facilities at risk. The NEA has also expanded the number of internationally funded research projects which it sponsors. In 2001 a major workshop entitled *Research in the Regulatory Context* was held. As new reactor concepts are developed, regulatory authorities will need the appropriate competencies and facilities to evaluate their safety performance. In view of this, a workshop on advanced nuclear reactor safety issues and research needs was held in 2002.

Following the review of past and continuing activities related to R&D in the first session, the workshop turned to the question of future R&D needs, particularly to support the development of new nuclear energy systems and accelerator-driven systems (ADS) for P&T.

Ten countries are participating in the Generation IV International Forum, which aims to develop a new generation of nuclear energy systems capable of being in operation by 2030. These would provide competitively priced and reliable electricity supplies while addressing concerns over nuclear safety, waste and proliferation. The roadmap identifies six advanced reactor concepts as meeting these goals and meriting further development.

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was established by the International Atomic Energy Agency in 2000. It aims to identify the actions required to ensure the continued availability of the nuclear energy option, including innovations in reactor design and fuel cycles.

In France, the Commissariat à l'Énergie Atomique (CEA), recognising the need for technological breakthroughs which go beyond LWRs, has selected for further development the concept of a high-temperature gas-cooled reactor with a fast neutron spectrum, refractory fuel, direct conversion with a gas turbine and an on-site fuel cycle. The construction of large experimental facilities, including a technology-testing reactor, is foreseen during the next decade.

Half of nuclear R&D funding in Finland goes toward nuclear waste management issues, with another 40% on reactor safety. More than half of all the funding is provided by the two main power companies. A programme examining advanced LWR technologies began in 1998, focusing on reactors which could be built within 5-10 years. The objective was to support evaluation of the economics and safety of candidate designs for the fifth Finnish reactor.

In the USA, the Advanced Fuel Cycle (AFC) programme is exploring the potential for advanced nuclear technologies to dramatically reduce the difficulty of disposing of spent nuclear fuel. The programme has two main components. The first aims to address the issues of existing spent fuel by reducing the volumes and toxicity of material requiring geologic disposal. The second is concerned with developing new fuel cycles for the longer term, including innovative reactors and ADS. The AFC programme is closely linked to the Generation IV initiative.

There are also programmes in both Japan and Europe for the development of ADS for P&T of minor actinides and long-lived fission products. In Japan, scenarios are being developed which incorporate an ADS into advanced fuel cycles involving, for example, MOX fuel in advanced LWRs and fast reactors. The OMEGA research programme for R&D in P&T aims to reduce the radiotoxicity of high-level waste to that of natural uranium. It includes basic experiments to demonstrate the processes and engineering scale experiments to obtain safety data. The Japan Atomic Energy Research Institute plans to construct the Transmutation Experimental Facility to study the basic characteristics of the ADS and demonstrate its feasibility.

In Europe, a technical working group was set up to co-ordinate activities in different countries on P&T and ADS development. In its 2001 report, this group concluded that P&T using an ADS could contribute to radioactive waste management, that there was a need to demonstrate this technology at an international level and that there should be a co-ordinated European R&D effort with support from the European Commission. Subsequently, European laboratories involved in P&T research established the ADOPT network to co-ordinate projects supported by the Commission, which now covers 15 projects.

Following the presentations described above, the workshop participants divided into three discussion groups in order to produce conclusions and recommendations. These groups concentrated respectively on the topics of nuclear data, reactor physics and systems behaviour, and fuels, materials, coolants and chemistry. The recommendations put forward by each group are summarised below.

The discussion group on nuclear data put forward a number of specific points and recommendations with regard to differential nuclear data, including:

- The High Priority Request List for nuclear data should be reviewed and improvements considered.

- Evaluation and processing codes should be widely and freely distributed within NEA member countries.

- A standard format for multi-group cross-section libraries for use in testing and benchmarking should be considered.

- A standard set of integral experimental benchmarks should be elaborated, covering a wide range of applications, as a common basis for validation studies.

- The international situation regarding experimental facilities and teams active in the fields of measurement, evaluation and validation should be monitored.

In addition, the discussion group made several suggestions for actions in relation to integral data:

- The preservation of information on past experiments should be encouraged.

- Consideration should be given to making new benchmarks available in a common format.

- Increased international collaboration should be encouraged for work at the few existing facilities for integral experiments still in operation.

The discussion group on reactor physics and systems behaviour agreed upon the following four items as the most crucial to be addressed in the future:

- Fuel cycle issues, covering both conventional and advanced systems, and including high burn-up, the physics and safety of advanced fast-spectrum systems, and minor-actinide recycling.

- Uncertainty analysis for nuclear power plant dynamics.

- Refined modelling for materials behaviour, neutronics, thermal-hydraulics and sensitivity analysis.

- The safety of non-electricity energy product systems.

In addition, the group underlined the strong need to match key experimental needs with the possible use of existing and/or planned facilities in order to set up NEA-sponsored joint projects.

The discussion group on fuels, materials, coolants and chemistry concluded that the NSC should support projects and facilities essential for sustaining the development of nuclear systems, and publish summaries of the relevant national and international programmes. In addition, the group recommended that the NSC organise the compilation of handbooks on lead-bismuth coolants, and advanced nitride, carbide, metallic and IMF fuels. The group also recommended that the NSC establish two new expert groups on irradiation defect behaviour and on materials science.

Chapter 1

INTRODUCTION

In the early stages of nuclear development publicly funded research and development (R&D) facilities and programmes played a major role. As nuclear power programmes developed, however, governments began to regard nuclear energy as a mature and commercial technology. Significant government funding continued, but was increasingly focused on areas such as reactor safety and radioactive waste management. As nuclear power plants were commercially deployed, R&D activities were increasingly performed or supported financially by companies in the nuclear industry, including electricity utilities. Much of this work has been aimed at improving the efficiency and performance of existing nuclear plants, including enhancing the performance of nuclear fuel.

As immediate interest in further expansion of nuclear capacity has waned, both governments and industry have scaled back their R&D programmes, closing facilities and reducing the numbers of scientists and engineers employed in nuclear R&D activities. The move to competitive electricity markets has compounded this. Many governments no longer consider that they should decide which power generation technologies should be developed and deployed. Power companies meanwhile have become increasingly concerned with short-term financial pressures.

As a result, there has been a consistent trend over recent years in OECD Nuclear Energy Agency (NEA) member countries for nuclear energy related R&D activities and funding to be cut back. This has been of increasing concern to many in the international nuclear community, including the NEA. Continuing programmes of nuclear R&D are essential for maintaining the safe and efficient operation of the more than 430 existing nuclear power plants and associated fuel cycle facilities, and to support related activities such as radioactive waste management and disposal.

An example of the latter is work to develop accelerator-driven systems (ADS) for the partitioning and transmutation (P&T) of minor actinides and long-lived fission products. This has the potential to significantly reduce the quantities of radioactive waste requiring geological disposal, and it is the focus of several ongoing R&D programmes in Europe, the USA and Japan.

The loss of nuclear R&D facilities and expertise could also have serious implications for the longer-term future of nuclear energy. On a global scale, there is an ever-increasing demand for energy. If adequate energy supplies are to become available to an increasing proportion of the world's population over the coming decades, power generation will have to increase greatly. At the same time, there is a need to restrain the use of fossil fuels, in order to limit emissions of greenhouse gases.

In this context, nuclear energy has the potential to play a vital role in the future alongside other non-fossil energy sources in achieving sustainable development. However, keeping open the nuclear option will require considerable R&D efforts to ensure the emergence of new and innovative nuclear energy systems in a timely manner. There are already a number of programmes underway to design and develop advanced reactors and fuel cycles. These are largely being conducted in the context of international co-operative frameworks such as the Generation IV International Forum (GIF).

For nuclear energy to fulfil its potential it is vital to identify future nuclear R&D needs and to ensure that adequate expertise and resources are available to meet them. Furthermore, in order to maximise the efficiency of R&D programmes, avoiding duplication of effort and making the most effective use of limited resources, increased international co-operation and co-ordination are essential. Encouraging and facilitating such international co-operation is an important part of the NEA's role.

In order to examine these issues, the Workshop on R&D Needs for Current and Future Nuclear Systems was organised by the NEA Nuclear Science Committee (NSC), in close co-ordination with the Committee on the Safety of Nuclear Installations (CSNI) and the Nuclear Development Committee (NDC). The workshop was held in Paris from 6-8 November 2002, and was attended by 33 participants from 11 countries and three international organisations. Altogether 15 presentations were made. This publication incorporates the proceedings of the workshop.

Within the NSC it was felt that action should be taken to create an international platform for discussion of R&D needs for current and future nuclear systems. It was recognised that each country might have a different strategy concerning nuclear power and consequently a different view of R&D needs, and the NSC did not seek to exclude the continuation of national R&D programmes. Rather, the aim was to begin the process of producing a "catalogue" of R&D needs. This should help the NEA, as well its individual member countries, to establish future R&D portfolios, to limit duplication of effort and to support international collaboration.

Before such a catalogue can be established it is necessary that inventories be prepared of R&D which has already been performed and of ongoing R&D. Only after the completion of these two steps can future R&D needs be accurately accessed. It was with this in mind that the organising committee structured the programme for the workshop.

The following section comprises reports from the first session of the workshop, providing an overview of past and continuing nuclear R&D activities in which the NEA has been involved. There are separate reports covering activities related to P&T, nuclear data, fuel behaviour, criticality safety, plutonium recycling and innovative fuel cycles, and reactor physics and shielding. These reports cover the past achievements and the present status of nuclear R&D programmes, particularly projects involving international co-operation supported by the NSC. There are also reports on the related activities of the CSNI and the NDC.

The workshop then devoted two sessions to R&D needs for new nuclear systems. The reports from these sessions cover several national and international nuclear R&D programmes and co-operative projects. Most of these are involved with the development of advanced and innovative nuclear energy systems, or of accelerator-driven systems for P&T.

Following the presentations, the workshop participants divided into three discussion groups in order to produce conclusions and recommendations. These groups concentrated respectively on the topics of nuclear data, reactor physics and systems behaviour, and fuels, materials, coolants and chemistry. Finally, the findings of each group were discussed by all the participants. A summary of the main recommendations produced by the three discussion groups is presented in the final section.

PAST AND ONGOING NUCLEAR R&D ACTIVITIES

Chapter 2

PARTITIONING AND TRANSMUTATION

2.1 General background

A first series of comprehensive studies investigating the feasibility of partitioning and transmutation (P&T) and its possible role in the management of nuclear waste streams was conducted in the 1970s, predominantly in Europe. On the whole, these studies were sceptical about the existence of cost, safety or other incentives for developing this technology. A renewed interest in P&T arose in the 1980s in response to increasing public doubts concerning the long-term safety of conventional high-level waste repositories. Difficulties in the commercialisation of large, conventional fast reactors and progress in the development of high-power accelerators further strengthened the interest in transmutation technologies in general and accelerator-based technologies in particular. It should be emphasised that, at present, P&T is not seen as an alternative to conventional waste management policies, but as a complement which may contribute to reducing the amount of high-level waste in need of geological disposal.

In response to the increased interest in P&T, some NEA member countries launched R&D programmes such as the OMEGA programme in Japan and the SPIN programme in France. Subsequently, "roadmaps" were developed for the demonstration of accelerator-based P&T technologies, for example, the *US Roadmap for Developing Accelerator Transmutation of Waste Technology*, and the *European Roadmap for Developing Accelerator-driven Systems for Nuclear Waste Incineration*. In 1989, the NEA responded to these developments by initiating a long-term programme on P&T. Activities under this programme cover a wide range of issues in the areas of nuclear science, strategy and development. They are carried out under the guidance of the NEA Nuclear Science Committee (NSC) and Nuclear Development Committee (NDC), which have established close co-operation on P&T.

This section overviews the past and ongoing P&T activities of the NEA, with emphasis on nuclear science related activities. In this context, it is worth mentioning that the NEA was already dealing with P&T before the creation of the NSC in 1991. For instance, between 1988 and 1990, the former NEA Committee on Reactor Physics repeatedly discussed the topic in special technical sessions entitled *Engineering and Physics Aspects of Transuranium Burning by Reactors and Accelerators*. Altogether, 16 technical papers were specially prepared for these discussions (see summary records NEACRP/L-314, NEACRP/L-318, NEA/NEACRP/R(91)1).

2.2 Early activities related to nuclear data and target physics

Due to the specific interest in accelerator-based transmutation concepts at that time, early discussions of the NSC concentrated primarily on such concepts. In particular, this included issues related to the prediction of high-energy reactions in the target, neutron transport in the target, and intensity and angular distribution of neutrons emerging from the target, i.e. physics issues uncommon in normal reactor physics. In this context, different benchmark exercises for the comparison of the models used in high-energy transport codes and on integral target characteristics were proposed and carried out.

2.2.1 *Comparison of codes for calculation of intermediate nuclear data*

In 1991, the NSC launched an international code comparison exercise on the calculation of intermediate energy nuclear data for accelerator-based transmutation applications. A first series of benchmark studies aimed at assessing the predictive ability of computer codes used in calculating intermediate energy charged particle data. Double-differential cross-sections as well as neutron yields and mass distributions of spallation products were investigated. The results were collected and analysed in comparison with experimental data. Later benchmark activities concentrated on thick targets and activation yields.

Documents produced include:

- *International Code Comparison for Intermediate Energy Nuclear Data* (1994).
- *Proceedings of the Specialists Meeting on Intermediate Energy Nuclear Data* (1994).
- *Intermediate Energy Thick Target Yield Benchmark* (1996).
- *International Codes and Model Intercomparison for Intermediate Energy Activation Yields* (1997).
- *Proceedings of a Specialists Meeting on Nucleon-nucleus Optical Model Up to 200 MeV* (1997).

2.2.2 *Nuclear data compilation and evaluation*

In the same time period, the NEA Data Bank, with scientific support from the Netherlands and voluntary funding from Japan, embarked on the creation of a special intermediate energy nuclear data library. This work was later continued with guidance from a special sub-group of the NSC Working Party on International Evaluation Co-operation (WPEC). Validation of higher actinide and fission product cross-sections with a view to reactor transmutation applications was also carried out in the framework of the Western European Joint Evaluated Fission File (JEF-2) project.

Documents produced include:

- *Review of Fission Product Yields and Delayed Neutron Data for the Actinides* ^{237}Np, ^{242}Pu, ^{242m}Am, ^{243}Am, ^{243}Cm *and* ^{245}Cm, NEA/P&T Report No. 1 (1990).
- *Review of High Energy Data and Model Codes for Accelerator-based Transmutation*, NEA/P&T Report No. 4 (1992).
- *Requirements for an Evaluated Nuclear Data File for Accelerator-based Transmutation*, NEA/P&T Report No. 6 (1993).
- *Results of an International Code Intercomparison for Fission Cross-section Calculations*, NEA/P&T Report No. 8 (1994).
- *International Nuclear Data Evaluation Co-operation: Vol. 14 – Processing and Validation of Intermediate Energy Evaluated Data Files* (2000).

2.3 Task Force on Physics Aspects of Different Transmutation Concepts

In 1992, the NSC agreed on the mandate of an expert group called the Task Force on Physics Aspects of Different Transmutation Concepts. The scope of this expert group included a global overview of the fundamental scientific issues of transmutation concepts proposed at that time, as well as the identification of the specific issues which required attention and analysis of the associated discrepancies and uncertainties.

The principal product of this activity was an overview report, *Overview of Physics Aspect of Different Transmutation Concepts*, which appeared in 1994. The report describes and examines more than 20 different transmutation concepts and makes recommendations for follow-up actions. As the comparison of the systems was complicated by inconsistencies in the analysis methods used by the proponents of the concepts, analytical benchmark calculations were felt to be particularly useful to improve the understanding of the basic physical phenomena and to provide a basis for more systematic systems analyses in the future.

2.3.1 Benchmark on physics aspects of transmutation concepts

In 1994, the Task Force on Physics Aspects of Different Transmutation Concepts launched an integral systems benchmark exercise aimed at investigating the physics of coupled transmutation systems involving the reprocessing of spent PWR fuel and the subsequent re-use of this fuel in different transmutation systems, i.e. a fast reactor or an accelerator-driven system (ADS). In accordance with this aim, the overall exercise consisted of three coupled benchmark exercises, one for each reactor concept. The parameters to be compared comprised the physics characteristics of the core and the activities of the individual actinides in the irradiated fuel. As many as 15 solutions were contributed, but only three of these referred to the accelerator-driven system.

For the PWR benchmark, the results showed consistency well within the accuracies evaluated by the Working Party on Plutonium Recycling (WPPR) for the multiple recycling of plutonium. Generally good agreement was also obtained for the prediction of the characteristics of the minor actinide loaded core in the fast reactor benchmark. For the ADS, the results of the high-energy calculations (number of source neutrons per proton, axial distribution of these neutrons in the target, etc.) agreed well, but large discrepancies were observed between the results of the neutron transport calculations below 20 MeV, i.e. in the effective multiplication factor and the burn-up characteristics of the sub-critical system.

The results of the exercise were published in 2000 in a report entitled *Calculations of Different Transmutation Concepts: An International Benchmark Exercise*.

2.3.2 Minor actinide burner benchmark

Noticing the significant discrepancies observed in the ADS part of the benchmark on physics aspects of transmutation concepts, the NSC decided in 1999 to launch a follow-up minor actinide burner benchmark exercise. The new benchmark model considered a lead-bismuth-cooled sub-critical system with a core design similar to that of an advanced liquid-metal reactor (ALMR). Two fuel compositions typical for the start-up and the equilibrium core of a minor actinide burner in a double strata fuel cycle scheme were considered.

The analysis of the solutions from seven organisations again revealed significant discrepancies for normal integral reactor and safety parameters as well as neutron flux distributions. Although the energy and space dependence of the spallation neutron source was predefined, large discrepancies also appeared in the estimation of the total number of external source neutrons per megawatt of power generated in the multiplying system.

Important conclusions from the study were that:

- Future benchmark exercises should be directed primarily at validating the basic nuclear data of transuranic actinides and the respective data processing routes at the level of basic fuel cells.

- In this context, there was a need for integral reaction rate measurements on small samples of minor actinides.

- In burn-up simulations, particular attention should be paid to the calculation of the thermal power in the system, the actinide burn-up chains, the treatment of fission products and the choice of parameters related to numerical approximations. It should be noted that most of the burn-up related issues were generic, i.e. not related to specific aspects of the minor actinide burner.

In summary, the outcome of the benchmark exercise indicated that the overall status of the nuclear data and computational tools for the analysis of accelerator-driven minor actinide burners was satisfactory for scoping calculations, but not for detailed design calculations. The results of the exercise were published in 2002 as *Comparison Calculations for an Accelerator-driven Minor Actinide Burner*.

2.4 Utilisation and reliability of high-power accelerators

The concept of the accelerator-driven system combines a particle accelerator with a sub-critical reactor core. Most proposals assume proton accelerators, either linear or cyclotrons, delivering continuous-wave beams with an energy of around 1 GeV. High-power accelerators have been under continuous development, and the construction of machines with the required electrical efficiency and beam power now appears to be feasible. However, for practical application in P&T systems, these machines will have to be improved with respect to the beam losses, which cause radiation damage and activation in the accelerator components, and the frequency of the beam trips, which subject reactor structures to strong temperature and mechanical stress transients.

Noting these issues, the NSC decided to hold a workshop in 1998 at Tokai Mura, Japan. The meeting brought together accelerator specialists and reactor physicists to highlight the problems and explore possible roles for the NSC in addressing them. The conclusions of the meeting recommended strong international collaboration in developing the respective concepts and performing the large amount of R&D which was necessary.

The different issues and concepts were discussed more thoroughly in follow-up workshops held in 1999 and 2002 at Aix-en-Provence, France, and Santa Fe, USA, respectively. At Aix-en-Provence, the discussions concentrated on the requirements for accelerator reliability and the R&D necessary to achieve these, and also on the impact of beam trips/fluctuations on the target and sub-critical reactor, including systems analyses. Considerable potential for reducing the mean time between failures, compared with existing facilities, was identified.

The proceedings of the first two workshops were published in 1999 and 2001 respectively, under the title *Utilisation and Reliability of High-power Proton Accelerators*.

2.5 Fuel cycle chemistry

From its first meeting in 1991, the NSC recognised the general importance of fuel cycle chemistry and discussed possible areas in which it would be appropriate to intervene. A short-term task force was set up to identify possible items for early action. The interest in this topic increased when it became evident that the feasibility of P&T is closely related to fuel reprocessing issues, in particular the ability to separate actinides and fission products with a very high recovery efficiency.

2.5.1 Task Force on Actinide Separation Chemistry

In 1993, the NSC set up the Task Force on Actinide Separation Chemistry to review scientific problems in the field of separation chemistry and to formulate recommendations for research. To this end, the task force prepared a state-of-the-art report which overviewed the existing basic actinide chemistry data, the need for additional data, and the types of actinide waste streams. It also looked at the various hydrometallurgical and pyrochemical separation processes, including new processes such as TRUEX, TALSPEAK, DIDPA, DIAMEX and TRPO, which are being developed specifically for minor actinide separation in P&T systems. In addition, the report prioritised needs for additional separation development work and made respective recommendations taking into account a time horizon of 25 years.

The report was published in 1997 under the title *Actinide Separation Chemistry in Nuclear Waste Streams and Materials*.

2.5.2 Workshops on chemical partitioning and speciation technology

Complementary to the above-mentioned state-of-the-art report, strategies for the separation of long-lived nuclides from nuclear wastes were reviewed at a workshop held in 1997 at Marcoule, France. The recommendations from this workshop proposed follow-up workshops devoted to the application of X-rays to radioisotope chemistry and to the evaluation of speciation technology. In response to these propositions, in 1998 the NEA and the European Commission co-sponsored the Workshop on Techniques and Facilities for Characterisation of Radioactive Materials at Synchrotron Light Sources at Grenoble, France. In 1999, the NEA organised a Workshop on Evaluation of Speciation Technology at Tokai-mura, Japan. The Grenoble workshop, the first in a series with this title, was organised by the Forschungszentrum Rossendorf, Germany, and hosted by the European Synchrotron Radiation Facility (ESRF). Two years later, the NEA also co-sponsored the second workshop in the series and published the proceedings.

The Tokai-mura workshop aimed at discussing and evaluating in different topical areas the advantages, disadvantages and limitations of various methods for the speciation of actinide and fission product elements in nuclear reprocessing and waste streams, and identifying the R&D needed for improving present, and developing new, speciation methods. The final product of the meeting was a report which will guide researchers in this field in choosing the most useful techniques. Documents produced include:

- *Proceedings of the Workshop on Long-lived Radionuclide Chemistry in Nuclear Waste Treatment* (1998).

- *Proceedings of the Workshop on Evaluation of Speciation Technology* (2001).

- *Proceedings of the Workshop on Speciation, Techniques and Facilities for Radioactive Materials at Synchrotron Light Sources* (1999, 2001).

2.5.3 Pyrochemistry

A Workshop on Pyrochemical Separation, co-organised by the NEA and the European Commission with the support of CEA-VALRHO, was held at Villeneuve-les-Avignon, France, in March 2000. The programme of this workshop addressed national and international R&D programmes on pyrochemical separation, requirements for pyrochemical reprocessing in future fuel cycles, and recent progress in the areas of basic data, experimental results, and process simulation and design. To provide a common scientific base on pyrochemical separation and to promote efficient international collaboration, the creation of a working group and the preparation of a state-of-the-art report were recommended. The proceedings of the workshop were published in 2001.

Taking note of these recommendations, the NSC created the Working Group on Pyrochemistry, with a broad mandate not restricted to P&T applications. The working group met at NEA headquarters in October 2000, in May 2001 and in September 2001, and was preparing a final report summarising its findings on the status of the technology and the R&D which was necessary. The P&T part of the mandate of this working group was later integrated into the mandate of a Chemical Partitioning Sub-group of a new Working Party on Scientific Issues in Partitioning and Transmutation (see below). The new sub-group also covers aqueous processes and, in this respect, has a wider scope.

2.6 Working Party on Scientific Issues in Partitioning and Transmutation

To improve the co-ordination of the increasing number of NSC projects and working groups in the P&T area, the committee created a Working Party on Scientific Issues in Partitioning and Transmutation (WPPT) in 2000. The scope of the WPPT covers four disciplines:

- Accelerator utilisation and reliability.
- Chemical partitioning.
- Fuels and materials.
- Physics and safety.

The general objectives of the working party are to provide the NEA member countries with up-to-date information on the feasibility and development status of P&T, provide advice to the P&T community on the required R&D, liaise with and advise other NEA groups, organise the biennial P&T Information Exchange Meetings (see Section 2.8) and publish a final state-of-the-art report in 2004.

Up to mid 2002, the WPPT held four general meetings at which ongoing P&T activities and projects were discussed and a work programme was defined. Regarding P&T scenarios, technology status and R&D needs, the conclusions from the NDC expert group report *Accelerator-driven Systems and Fast Reactors in Advanced Nuclear Fuel Cycles: A Comparative Study* (published in May 2002) provided a particularly useful input to the discussions. Concerning working methods, the WPPT decided that the different technical topics should be handled by separate expert groups, and consequently launched four WPPT sub-groups, one for each of the above-mentioned disciplines.

2.6.1 Sub-group on Accelerator Utilisation and Reliability

The scope and programme of the WPPT Sub-group on Accelerator Utilisation comprise:

- Evaluation of the potential and performance of accelerators, spallation targets and beam entrance windows as components of accelerator-based transmutation systems.

- Evaluation and ranking of potential issues related to these components with regard to system performance, reliability and safety.

- Organisation of relevant workshops, including the general workshops on utilisation and reliability of high power accelerators.

- Preparation of a prioritised list of issues which require particular attention.

Regarding spallation targets, a particularly important issue is the degradation of the beam entrance window due to the combined effects of the thermo-mechanical loads, radiation damage by high-energy particles, and corrosion phenomena. Another issue to be addressed is the radiotoxicity of the spallation products. Evaluation of the performance of the whole system will require close co-operation with other sub-groups, especially the Sub-group on Physics and Safety.

The sub-group held its first meeting in November 2001 and a second meeting in May 2002. The principal product of the sub-group will be a final report to the WPPT, due in 2004. This report will contain information on the technological status of the proposed accelerators and targets, the data available to support an accelerator application, the ongoing R&D, the gaps in knowledge which require attention, the R&D required to fill these gaps and international collaboration.

2.6.2 Sub-group on Chemical Partitioning

The WPPT Sub-group on Chemical Partitioning incorporates members of the former NSC Working Group on Pyrochemistry, together with experts in aqueous separation processes. The work of the sub-group is focused on the separation processes relevant to currently discussed P&T scenarios involving a broad range of fuel types, including oxide fuel (uranium oxide, mixed uranium-plutonium oxide, inert matrices), fertile and non-fertile nitride fuel, composites (cermet, cercer), fertile and non-fertile metal alloy fuel, TRISO-coated graphite particulate fuel, and uranium and/or thorium based molten salt fuel.

For the different scenarios, the sub-group will assess the technical feasibility and the technological maturity of the required separation processes, develop specific mass balance flow sheets, and perform decision analyses on technical issues. The research, development and demonstration necessary to bring preferred technologies to a deployable stage will be identified, and appropriate collaborative international efforts will be recommended.

The sub-group met in September 2001 on the occasion of the third meeting of the Working Group on Pyrochemistry and, at this first meeting, agreed on the scope of separation processes to be covered. The sub-group will organise a joint seminar with the Sub-group on Fuels and Materials in 2003 and intends to deliver its final report to the WPPT in 2004.

2.6.3 Sub-group on Fuels and Materials

The work of the WPPT Sub-group on Fuels and Materials is focused on the evaluation of the performance of fuels and materials for the different P&T scenarios. In particular, the sub-group is reviewing fundamental fuel properties, fuel selection criteria, fuel fabrication and behaviour issues, cladding-coolant compatibility issues, and long-lived fission product issues. A general objective is to provide information on the state of the art of the respective technologies, the availability of pertinent data, and the necessary R&D to close gaps in the databases.

The work programme includes an initial survey of relevant studies by means of a questionnaire, and a peer-reviewed final report due by mid-2004. The latter will include information on:

- Fundamental thermophysical and thermochemical properties of relevant actinide compounds and alloys.
- Fuel selection criteria for specific P&T scenarios (LWR-FBR and double strata scenarios).
- Fuel fabrication (radiation and heat effects, thermochemical and reprocessing issues).
- Fuel behaviour (problems not encountered in normal U-Pu fuels, including uncertainties).
- Cladding and fuel assembly material problems (compatibility with lead/lead-bismuth, high-energy irradiation effects in accelerator-driven systems).
- Long-lived fission products.
- Ongoing R&D and international collaboration.

The sub-group first met in April 2002 and identified the main authors and reviewers for the final report. The sub-group will hold a joint seminar with the Sub-group on Chemical Partitioning in 2003.

2.6.4 Sub-group on Physics and Safety

The scope of the WPPT Sub-group on Physics and Safety comprises the organisation of analytical and experiment-based benchmark exercises for transmutation systems as well as the assessment of the consequences of beam trips in accelerator-driven systems, physics-related sensitivity studies and safety studies for P&T systems. The areas of interest and respective principal objectives are:

- *Reactor physics* – analyses of system performance (core characteristics, transmutation effectiveness, etc.).
- *Safety* – evaluation of safety approaches (lines of defence, defence in depth), reliability and transient safety analyses (including beam trip issues).
- *Reactor control* – evaluation of reactor control options (including the optimisation of the sub-criticality level).

Both homogeneous and heterogeneous concepts for the transmutation of transuranics, minor actinides and fission products are being considered. Regarding benchmark exercises, the sub-group is involved in an experiment-based ADS benchmark exercise carried out within the framework of the MUSE programme at Cadarache, and will shortly launch a beam trip benchmark exercise.

The sub-group first met in April 2002, when it agreed on details of the work programme and its schedule. The sub-group will organise workshops to enhance information exchange, and will deliver a state-of-the-art report. In a later phase, its range of activity could be extended to include scenario studies similar to those performed by the above-mentioned NDC expert group study.

2.7 Other NSC activities of relevance to P&T

2.7.1 Advanced reactors with innovative fuels

In response to recent initiatives for the development of new reactor types which meet the requirements of the twenty-first century, a series of Workshops on Advanced Reactors with Innovative Fuels (ARWIF) was launched. These workshops address the core behaviour (reactor physics and thermal-hydraulics) and fuel material technology of advanced reactors with different types of innovative fuels, such as advanced U-Pu oxide fuels, uranium-free fuels (inert matrices), and non-oxide fuels, including molten salt fuels. Although the workshops are organised in the framework of the Working Party on Plutonium Recycling, their scope is much wider than that of the WPPR and specifically includes the homogeneous and heterogeneous recycling of minor actinides.

The first ARWIF workshop, hosted in October 1998 by the Paul Scherrer Institute at Villigen, Switzerland, considered water-cooled and fast reactors as well as accelerator-driven systems with fast and thermal neutron spectra. Particular goals of this workshop were to exchange information on R&D activities, identify areas where international co-operation could be strengthened, and identify the roles which could be played by existing experimental facilities as well as the possible needs for new facilities. For ARWIF-2001, hosted at Chester, UK, by British Nuclear Fuels, the scope was extended to include high-temperature gas-cooled reactors. In the panel discussion at the end of this workshop, a specific question addressed strategies for transmutation and, in particular, the separation and burning of curium.

Proceedings of the ARWIF workshops at Villigen and Chester were published in 1999 and 2002, respectively. A third workshop in the series will be held in 2004.

2.7.2 Shielding aspects of accelerators, targets and irradiation facilities

A series of specialists meetings on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF) has been jointly organised by the NEA, the Shielding Working Group of the Reactor Physics Committee of Japan and the Radiation Safety Information Computation Center (RSICC) at Oak Ridge, USA. Particular objectives of the SATIF meetings are the promotion of experiments to improve the knowledge of thin and thick target neutron yields and radiation shielding modelling, including computer code and nuclear data aspects. Related to this work is the compilation of peer-reviewed data on radiation shielding experiments (the SINBAD database). SINBAD is intended for the validation of nuclear models and data in computer codes. Currently, the database contains data from 50 experiments with particular emphasis on pressure vessel dosimetry. However, about 15 of these are relevant to advanced reactors and accelerator facilities and hence to transmutation.

The SATIF meetings have become an ideal forum for exchanging views and sharing experiences in the field of radiation shielding applied to accelerators, targets and irradiation facilities. Moreover, they have helped to achieve significant progress in the organisation of respective benchmark exercises, the compilation and exchange of nuclear models, computer codes, nuclear and hadronic data libraries, and the computation of conversion coefficients for high-energy radiation dosimetry purposes. Some of this work is also relevant to transmutation applications.

Proceedings of the SATIF specialists meetings held in 1994 at Arlington, USA, in 1995 at CERN, Switzerland, in 1997 at Sendai, Japan, in 1998 at Knoxville, USA, and in 2000 at Paris, France, have been published.

2.8 Activities under the guidance of the NEA Nuclear Development Committee

In the framework of the NEA long-term programme on P&T and under the guidance of the NDC, the NEA has launched an information exchange programme, sponsored specialists meetings, engaged specialists to produce state-of-the-art reports and nominated expert groups to conduct systems studies.

2.8.1 Information exchange programme, specialists meetings, state-of-the-art reports

The Information Exchange Programme on Actinide and Fission Product Partitioning and Transformation was constituted in 1989 in response to an initiative by the government of Japan. Inspired by a presentation of the Japanese OMEGA programme at a special meeting in Paris, the information exchange started with a very broad scope, ranging from basic science questions to complete P&T scenarios. Reactor-based, spallation-based and other, even more advanced, accelerator-based transmutation technologies were considered at that time and discussed at a first information exchange meeting at Mito, Japan, in 1990.

Two follow-up specialists meetings – one on partitioning technology and one on accelerator-based transmutation – were held, and it was decided to organise general information exchange meetings every two years. Between 1990 and 2000, six information exchange meetings were held, with increasing participation. The seventh meeting in the series was held in Korea in October 2002.

The two specialists meetings were held at Mito and at Villigen in 1991 and 1992, respectively. The Villigen meeting focused on concepts of accelerator-based transmutation systems, design problems with emphasis on the accelerator-target and target-reactor interfaces, and nuclear data and methods including measurements and validation. Proposals for respective target benchmark experiments were also discussed. Already before this meeting, the NEA engaged a consultant to produce a state-of-the-art report on the application of spallation technology to transmutation. Another consultant was engaged to produce a report which highlighted the impact of P&T on the waste management.

Documents produced include:

- *Information Exchange on Actinide and Fission Product Partitioning and Transmutation*, proceedings of information exchange meetings at Mito, Japan (1990); Argonne National Laboratory, USA (1992); Cadarache, France (1994); Mito, Japan (1996); Mol, Belgium (1998); and Madrid, Spain (2000).

- *Proceedings of Specialists Meeting on Partitioning Technology*, Mito, Japan (1991).

- *Proceedings of Specialists Meeting on Accelerator-Based Transmutation*, Villigen, Switzerland (1992).

- *Survey of Codes Relevant to Design, Engineering and Simulation of Actinide Transmutation by Spallation* (including cost estimation of accelerator for incineration and radiation hazard problems), NEA/P&T Report No. 5 (1991).

- *Role and Influence of Partitioning and Transmutation on the Management of Nuclear Waste Streams*, NEA/P&T Report No. 3 (1992).

2.8.2 P&T systems studies

A first P&T systems study, conducted from 1996 to 1998, focused on a review of the progress in the separation of long-lived actinides and fission products, the options for their transmutation and the benefits for waste management. Specific fuel cycle schemes were discussed, covering plutonium recycling and the additional burning of minor actinides in dedicated systems. However, the study did not address the more effective transmutation strategies with fully closed fuel cycles, nor the technology of accelerator-driven systems and their specific role in such closed fuel cycles.

With voluntary funding from Japan, the NEA therefore initiated a second P&T systems study with emphasis on accelerator-driven systems and fast reactors. This second, complementary study, conducted from 1999 to 2002, aimed at clarifying the roles and relative merits of critical and sub-critical fast-spectrum systems in closed fuel cycles, with the help of a set of representative fuel cycle schemes. The report is divided into two parts: a general introductory part which discusses the incentives for closed fuel cycles and the role of fast-spectrum systems in these fuel cycles, summarising the principal results of the comparative analysis of the fuel cycle schemes; and a technical part which deals with the reactor and fuel cycle technology and its status, ADS safety, and R&D needs. The report also includes a preliminary cost comparison of the schemes.

The second P&T systems study confirmed that all transmutation strategies with fully closed fuel cycles can, in principle, achieve similar reductions in the actinide inventory and long-term radiotoxicity of high-level waste, and that accelerator-driven systems are particularly suited as dedicated minor actinide burners in a double strata fuel cycle strategy. An interesting conclusion was that strategies which involve the use of an ADS show an overall economic benefit from burning as much plutonium as possible in less-expensive conventional reactors (the double strata strategy achieves this by burning the plutonium in mixed-oxide-fuelled LWRs and/or fast reactors). However, actinide-burning schemes which are fully based on (critical) fast reactors appear to be similarly attractive. They require higher initial investment in fast reactors, but are less dependent on innovative reactor and fuel cycle technology than the double strata strategy.

With regard to R&D, the study concluded that emphasis should be placed especially on:

- Basic R&D on advanced fuels, reprocessing methods, structural materials and liquid metals.

- Demonstration of advanced fuels and their pyroprocessing at appropriate scale.

- Simulation of the behaviour of materials under irradiation and high temperature.

- Clarification of advantages and disadvantages of different coolants for fast-spectrum systems.

- Safety analyses of accelerator-driven systems.

- Performance assessment studies of geological repositories using a P&T source term.

The report of the first P&T systems study, *Actinide and Fission Product Partitioning and Transmutation: Status and Assessment Report*, was published in 1999. The report of the second study, *Accelerator-driven Systems and Fast Reactors in Advanced Nuclear Fuel Cycles: A Comparative Study*, followed in 2002.

Chapter 3

NUCLEAR DATA

3.1 Introduction

Nuclear data are required for the design, safety assessment and operation of nuclear power plants and associated waste management facilities. They also find applications in various analysis techniques, in the production of radioisotopes and in medicine. Nuclear data are also required for astrophysics studies, and some facilities which were previously used for nuclear industry-related measurements are now engaged in such measurements (e.g. at Karlsruhe, Germany, and ORELA at Oak Ridge, USA). However, in this report only nuclear data required by the nuclear power industry are considered.

The main emphasis of current programmes is on data requirements for waste management (minor actinides and fission products), for new designs of fission reactor [including accelerator-driven systems (ADS)], and for fusion reactor shielding, materials activation and heating studies. Recycling in thermal reactors, high burn-up cycles, the use of alternative coolants and the thorium cycle are also giving rise to data requirements. Work to reduce the discrepancies in calculations for existing thermal reactors is also continuing. There are plans to build experimental ADS facilities and the design of these has given rise to new requirements which are difficult to meet with present levels of funding.

Neutron interaction data in the energy range 0.001 eV to 20 MeV and the yield and decay properties of radioactive isotopes have been the data of primary interest for fission and fusion reactor studies. Data for the reactions induced by alpha and gamma radiation, and in particular for the neutron producing reactions alpha-n, gamma-n and gamma-f, are also needed but the accuracy requirements have not been so stringent. Related data, such as energy production and deposition, build-up factors, gas production and irradiation-induced damage, are also needed.

Recent studies relating to ADS have extended the energy range for neutron interaction data above the traditional 20 MeV upper limit, and have also given rise to the need for proton interaction data to several hundred MeV and even in the low GeV range. In addition, data extending to about 50 MeV are required for design studies relating to materials irradiation facilities for fusion reactor materials, such as the proposed International Fusion Materials Irradiation Facility (IFMIF).

The work involved in providing the nuclear data required by the nuclear power industry can be separated into the following areas:

- *Measurements of "differential" or microscopic data*. The word "differential" is used to distinguish basic data from "integral" or macroscopic measurements in which averages of the data are obtained. Integral measurements can be the average of reactions induced by neutrons having a range of energies or the average neutronics characteristics of a mixture of materials. Differential measurements include measurements of cross-sections as a function of incident particle energy and the multiplicity, energy and angular distributions of secondary particles. (Some types of integral data can be measured on differential measurement facilities.) Radioactive decay data and the yields of products in fission are also basic measurements.

Differential data are openly published and are made generally available. For high-resolution cross-section measurements, from thermal energies to several MeV, pulsed white source time-of-flight facilities are needed. For all of these techniques, samples of the required elements or isotopes must be available and affordable. Differential measurements are still regarded as essential by evaluators and developers of nuclear theory.

- *Integral data measurements*. These can be made on critical facilities or using specially tailored sources or benchmark fields. They are used to validate (and in some cases adjust) the differential data. When the form of the energy dependence of a cross-section is known, an integral measurement is sufficient for the normalisation of the cross-section. Such measurements can be closer to the parameters to be calculated and so provide a more accurate basis for their prediction. Integral measurements are often proprietary and only available to the organisations participating in the measurement programmes.

- *Evaluation of the differential data*. The task of the evaluator is to consider all of the relevant measurements and derive a recommended set of data. Nuclear theory can be used to interpolate and to help in making corrections to the measurements. Some data can be obtained just using theory when the accuracy requirements are not too stringent.

- *Validation of the evaluated differential data using the results obtained in integral measurements*. Validation of evaluated data is an essential step before the data can be recommended for use and this activity requires an effort comparable to data evaluation. It is important to ensure that the integral database is appropriate for testing the accuracy of the data for the proposed applications. The nuclear data used in calculations can be adjusted to improve agreement with integral measurements.

- *Processing nuclear data to the forms used in calculations*. Energy group averaging of cross-section data is usual for routine calculations. Continuous energy Monte Carlo codes can represent the data more exactly, but there are still some approximations (the effect of Doppler broadening of resonances on secondary energy distributions, solid state effects, treatment of unresolved resonance regions, etc.) The NJOY code (developed at Los Alamos) is widely used for processing evaluated interaction cross-sections. Deterministic code schemes include procedures for treating resonance structures in cross-sections and the heterogeneity of the geometry of reactor regions (cell codes). These procedures can be specific to the particular code scheme and require a corresponding pre-treatment of the resonance region data specific to the method used. There are a number of different cell codes, involving greater or lesser approximations, used for this stage of reactor calculations. Standard group structures have been adopted for different applications, but standard representations for resonance regions are restricted to simpler approximate treatments (the Bondarenko method, or narrow resonance approximation, for example). However, these simpler representations, while adequate for most fast reactor and shielding studies, are not suitable for thermal reactor calculations. Applications nuclear data sets have been derived for use in some of the more widely used schemes of calculation and are available from data centres to organisations in NEA member countries. Mention can be made of the IAEA WIMS Library Update Project which has produced libraries for this thermal reactor code scheme. In general, however, libraries processed for use in particular codes are not made freely available.

- *Assessing the accuracy of calculations*. In principle, given the uncertainties in the differential data, and the sensitivity of calculated properties to changes in nuclear data, the accuracy of calculations can be estimated. However, the basic uncertainty information is far from complete and sensitivity calculations are not always easy to make. The uncertainty information is referred to as covariance data because the uncertainties in the values at different incident particle energies and for different types of interaction are correlated. The analysis of integral measurements can give a useful guide to the accuracy of predictions. In many cases it is easier to assess the accuracy of an integral measurement than the evaluated differential data.

- *Assessing the requirements for new measurements and evaluations*. The High Priority Request List is produced by an international collaborative effort co-ordinated by a sub-group of the Working Party on International Evaluation Co-operation (WPEC) of the NEA Nuclear Science Committee (NSC). It is important for requesters to be aware of the number of years required to carry out the various stages between the initial plan and the data being available in a usable form.

- *Nuclear standards*. These are given special consideration because of their importance in measurement and evaluation. Standard neutron cross-sections are used by experimentalists to determine neutron fluence in cross-section measurements; they are also particularly important for obtaining absolute fission cross-sections from fission ratio measurements. A consensus has arisen that certain of the neutron standard cross-sections need to be upgraded, and that a better understanding of the standards evaluation procedure is needed (particularly in relation to the magnitudes of uncertainties and covariance in the data). This has led to international collaborative efforts under the auspices of WPEC and the IAEA Nuclear Data Section, which has sponsored a co-ordinated research programme to upgrade the evaluations. The work has also been supported by the Cross-section Evaluation Working Group (CSEWG) in the USA. The aim is to complete this work by 2004-5. All the evaluation projects regard the accuracy of the data for the standards as most important.

3.2 Differential data measurement activities

In several countries differential nuclear data measurement facilities have been shut down. It was considered that there were no essential nuclear data measurement requirements for the operation of existing nuclear reactors, and that if requirements were to arise the data could be obtained using facilities in other countries or operated by international organisations. In some other cases, facilities have been maintained in operation but used for applications other than nuclear power, examples being the astrophysics-related studies being made using the facilities at Oak Ridge National Laboratory (ORNL), USA, and Karlsruhe, Germany. Because of the dependence of some countries on work being carried out internationally it is necessary to review the situation periodically to ensure that the facilities and teams required for the operation of the measurement facilities, and for evaluation and validation studies, are being maintained at the required level.

Many differential nuclear data measurement facilities are in universities, and there has been a tendency for nuclear physics research in universities to focus on new areas. For example, the Dynamitron at Birmingham University in the UK is no longer used for nuclear data work, and is in fact no longer suitable for such work. The facilities in several universities are now ageing, and as senior scientists retire the replacement staff often consider other areas of research more rewarding.

It should be noted that some facilities only provide beam time to clients on a commercial basis. A case must be made to the facility management for the proposed measurement by the team wishing to make it. The team must make all the arrangements for the sample and ancillary equipment.

The detectors available at each facility influence the types of measurement made, and the development of detectors is an important activity. Many detectors are of the type used for the past

40 to 50 years. However, new detectors and detector arrays are being designed or adapted from other fields, such as high energy physics. One of the biggest advances is in the use of fast computers and their data acquisition and analysis software.

Examples of the types of measurement required include:

- High-resolution total, capture and fission cross-sections in resonance regions. These require white source time-of-flight facilities.

- Energy and angular distributions of scattered neutrons or secondary particles.

- Neutron- and charged-particle-induced activation cross-sections.

- Yields of prompt and delayed neutrons in fission.

- Yields of fission products and their decay properties (measured using the on-line separators).

- Radioactive decay characteristics of unstable nuclei.

The required energy range of a cross-section measurement and the incident particle (neutron, proton, gamma or alpha) also determine the facilities which can be used to make the measurement. It is usually desirable to have several independent measurements of an item of data, using different techniques where possible.

For many measurements, isotopically enriched samples or radioactive materials are needed. In the past, most of the separated stable isotopes came from Calutron separations at ORNL. These electromagnetic separators have now been shut down, but there remains a (shrinking) inventory of stable isotopes. Some stable isotopes are also available from Russia, often through distributors in other countries. Samples of radioactive isotopes, including fission products and minor actinides, are more difficult to obtain because of the cost of separation (chemical or isotopic), the difficulty of handling radioisotopes, regulatory requirements, shipping restrictions and the need to dispose of associated radioactive waste. Over the past 20 years there has been a significant decline in the availability of these materials and in the capabilities to fabricate them into the desired forms. In addition, a detailed analysis of the sample can be required for an accurate measurement and this can present problems.

Measurement facilities and activities are summarised in Appendix 1 (although this might well be incomplete). The following initiatives are being taken to ensure continuity of expertise and facilities and to improve collaboration.

The European Commission’s Institute of Reference Materials and Measurements (IRMM) at Geel, Belgium, operates facilities for differential nuclear data measurements. The electron linear accelerator time-of-flight facility, GELINA, and the 7 MV Van de Graaff make a major contribution. Elsewhere in Europe, there is also an important new spallation source time-of-flight cross-section measurement facility, n-TOF, at CERN in Geneva, Switzerland; some other differential measurement facilities have been renewed and enhanced.

Cross-section measurements at these and other European facilities are being funded by the Fifth Framework Programme of the European Commission/EURATOM. This programme funds nuclear data work for controlled thermonuclear fusion and for fuel cycle safety. Two projects in the second category are on the topic of partitioning and transmutation. These are the High and Intermediate Nuclear Data for Accelerator-driven Systems (HINDAS) project (budget €2.1 million) to provide nuclear data in the energy range 20-200 MeV, and the n-TOF-ND-ADS project (budget €2.4 million) to provide nuclear data for actinides and long-lived fission products in the energy range 1 eV to 250 MeV. These projects group together a number of laboratories.

The major facilities in Western Europe are proposing an integrated project on nuclear data for waste transmutation for the Sixth Framework Programme. Active participants are: CEA Saclay, France; the Centre National de la Recherche Scientifique (CNRS), Orsay, France; the Nuclear Research and Consultancy Group (NRG), Netherlands; and the Vienna University of Technology, Austria. Participating facilities are at GSI, Darmstadt, Germany; the Svedberg Laboratory (TSL), Uppsala University, Sweden; Physikalisch-Technische Bundesanstalt (PTB), Germany; the Université Catholique de Louvaine (UCL), Belgium; and IRMM. But measurements could be carried out at any convenient facility, for example, n-TOF, Bordeaux, etc.

Several doctoral and post-doctoral students have been attached to the Geel laboratory by European Union (EU) member countries (in particular, France, Germany and Sweden) to ensure continuity of expertise, and there have been a number of visiting scientists. Also, efforts have been undertaken to collaborate with national institutes in EU candidate countries (the PECO initiative).

The Joint Evaluated Fission and Fusion File (JEFF) project has an experimentalists sub-group which reviews measurement requirements, assesses priorities and feasibility, and co-ordinates research activities in order to make the best use of the present infrastructure.

In Japan, joint research projects involving universities and/or the Japan Atomic Energy Research Institute (JAERI), the Japan Nuclear Cycle Development Institute (JNC), and the High Energy Accelerator Organisation (KEK) are being promoted and encouraged. Scientists from other university departments, and other countries, participate in the measurements at all of the facilities. For example, the Department of Energy Engineering and Science of Nagoya University has an active measurement programme using the facilities at other institutes.

Almost all nuclear data measurements are conducted with support from JAERI or JNC, or as a part of research projects at KEK or JAERI. The funding for measurements is considered to be insufficient to meet the requirements. Since the mid 1990s Japan has funded several projects in CIS countries through the International Science and Technology Centre (ISTC) in Moscow, each costing between US$200 000 and US$500 000.

In the USA, some measurements are funded by the Department of Energy's Division of Nuclear Physics (DOE-NP), National Nuclear Security Agency (NNSA) and Division of Nuclear Energy (DOE-NE). Funding from DOE-NP and DOE-NE has decreased significantly since the early 1990s, to about US$4 million per year for nuclear data, including US$2 million for the National Nuclear Data Center (NNDC) at Brookhaven. Low energy nuclear physics research receives US$25 million. The NNSA funds measurements and evaluations at DOE defence laboratories, and many of the results are of use in nuclear power applications. Realising that there will be a shortage of personnel and capabilities in nuclear data, the NNSA has initiated a programme entitled *Stewardship Science Academic Alliance* to fund universities to participate or even take leading roles in these nuclear data measurements.

3.3 Integral data measurement activities

At CEA Cadarache in France there are versatile facilities for studying fast reactor assemblies (MASURCA) and water-cooled thermal lattices (EOLE), and for measurements of material reactivity worth (MINERVE). MASURCA now has a DT source to enable studies of accelerator-driven systems to be made. There are also a number of research reactors which can be used for making integral measurements by irradiation of samples. In addition to funding from the French nuclear power industry and the European Commission, there is substantial funding from organisations in other countries, particularly Japan, a partner in the EOLE programme of measurements.

At Mol, Belgium, there is the VENUS facility for studies on water-cooled thermal lattices and pressure vessel dosimetry. There are also the research reactors BR2 and BR3 which can be used for sample irradiations, dosimetry studies (using standard benchmark fields) and shielding simulations. A new reactor facility called MYRRHA has been proposed which will be suitable for studies on accelerator-driven systems. International collaborations have funded several measurement programmes on the VENUS critical facility.

At the Paul Scherrer Institute (PSI) in Villigen, Switzerland, there is the multi-purpose PROTEUS critical facility. This has been used in the past for studying mock-ups of regions of advanced reactor systems, ranging from gas-cooled fast reactors to pebble-bed high-temperature reactors, with international participation. The experimental programme currently underway involves the investigation of advanced LWR fuel designs and operating modes. PROTEUS is currently being operated on the basis of a collaboration between PSI and the Swiss nuclear utilities.

At ENEA Frascati, Italy, and at TU Dresden, Germany, there are 14 MeV sources which are used to simulate conditions in the structures of fusion reactors. Measurements of neutron and gamma transmission, activation and heating in different materials are used to validate nuclear data in the European Fusion File (EFF). There are several other research reactors in Western Europe which can be used for irradiation experiments.

Similar integral data facilities are also in operation elsewhere in the world. At JAERI in Japan there is the fast critical facility FCA, and there are also thermal reactor facilities. There are also a number of research reactors in which samples can be irradiated. Recent measurements include fission product capture and integral decay heat.

Many of the facilities in the USA were shut down in the 1990s; for example, fast critical assemblies at Argonne and the Tower Shielding Facility at ORNL. However, the Los Alamos Criticality Experiments Facility (LACEF) is still carrying out criticality-related experiments. In Russia, the Institute of Physics and Power Engineering (IPPE) at Obninsk has a number of experimental critical assembly facilities and research reactors.

3.4 Evaluation activities

3.4.1 Present evaluation projects

Evaluation involves the careful interpretation of measured data using nuclear theory. Assembling a file of evaluated data is a major task, requiring quality assurance (QA) procedures and testing of the data for consistency. Before the data can be recommended for use they must be extensively validated. Present evaluation projects are:

- The US evaluation project ENDF/B, co-ordinated by the Cross-section Evaluation Working Group (CSEWG) and supported by the NNDC.

- The Japanese Evaluated Nuclear Data Library (JENDL), co-ordinated by the Japanese Nuclear Data Committee and supported by the Japanese Nuclear Data Centre.

- The Western European Joint Evaluated Fission and Fusion File (JEFF) project, co-ordinated by the JEFF Scientific Co-ordination Group and supported by the NEA Data Bank, and the associated European Fusion File (EFF) project, partly funded by the European Commission.

- The Russian nuclear data library BROND, co-ordinated at the Russian Nuclear Data Centre, Obninsk.

- The Chinese nuclear data library CENDL, co-ordinated at the Chinese Nuclear Data Centre.

In addition, nuclear data evaluation activities are undertaken by several other countries and international organisations. These activities provide inputs to the above libraries.

Several of these projects are suffering from insufficient resources being available to maintain the desired progress. The number of scientists engaged on evaluation and validation studies has declined significantly in some countries, with many retirements in the past 20 years. International co-operation has become more important as the number of scientists working in the field has decreased.

The ENDF/B project

Following the distribution of release 8 of ENDF/B-VI, this library is being frozen and new evaluation upgrades will be included in ENDF/B-VII. This will include photonuclear data, neutron- and proton-induced data up to 150 MeV, as well as numerous upgrades to actinide evaluations. Much of the new evaluation work is being supported by applications related to defence, criticality safety and nuclear power. Some changes to the ENDF/B-VI format are also planned. The timing of ENDF/B-VII is linked to the production of new evaluations for standards, expected in 2004-05. Release 8 of ENDF/B-VI included improved gamma-ray spectra (evaluated at Los Alamos) and neutron capture cross-sections of fission product isotopes in the thermal and resonance ranges (evaluated by the Korean Atomic Energy Research Institute (KAERI)). Actinide evaluations have historically been jointly developed by Oak Ridge (resonance region) and Los Alamos (fast region), with data testing at Argonne and other laboratories. The benchmarking carried out by CSEWG and the JEFF project had earlier resulted in a re-evaluation of ^{235}U resonance capture at Oak Ridge, and has also indicated possible problems with the ^{238}U evaluation. In addition to the ENDF libraries, there are libraries for special applications maintained at Lawrence Livermore National Laboratory.

The JENDL project

A new Japanese evaluated nuclear data library, JENDL-3.3, was produced in May 2002. This includes covariance data for the major materials (major actinides, structural materials and coolants) and makes improvements to the data in JENDL-3.2 on the basis of extensive benchmarking and by taking into account recent measurements. The revision of the library began in April 1997. Data for some new materials have been added and gamma-ray production data have been included for fusion reactor materials. The library will be released following extensive testing. In addition to the general-purpose library there are several special-purpose libraries: fusion data, actinide data, dosimetry data, activation cross-section data, alpha-n data, fission product decay data, high-energy data [for neutrons to 50 MeV (required for IFMIF) and protons to about 3 GeV], PKA/KERMA data and photonuclear data. Some of the evaluations and associated measurements have been carried out as ISTC projects with Russia and Belarus.

The JEFF project

The Joint Evaluated Fission and Fusion File project brings together evaluators in Western Europe. The project has been running for 20 years, and in April 2002 it produced the first version of the JEFF-3 general-purpose library (JEFF-3.0), after 10 years of validation work and reassessment of the JEF-2.2 library. It now includes the evaluations for fusion applications contained in the European Fusion File (EFF-3). With the addition of fusion data, the project changed its name from JEF to JEFF. The European Activation File (EAF) is also linked to the JEFF library.

The work is carried out by working groups co-ordinated by the Scientific Co-ordination Group. There is now a Working Group on Measurement which discusses requirements and measurement plans. Other working groups include:

- Fission Yields and Radioactive Decay Data.
- Benchmark Testing, Data Processing and Evaluation (both a JEFF working group and a combined JEFF/EFF working group).
- Intermediate Energy Data.
- NJOY (a data library processing program).

Validation studies have made an essential contribution to all of the developments. The project is now engaged on an extensive programme of validation studies of JEFF-3.0. Well-specified benchmarks are essential for this work. The IAEA co-ordinated International Dosimetry File, IRDF-2002, has been adopted for JEFF-3, and the work of the IAEA co-ordinated research project on fission yield data is used in the evaluation of yields.

Funding the effort needed for all of the required development work is a continuing problem. There is a strong interaction with the French nuclear industry, and the JEFF project would like to see this interaction extended to the nuclear industries in other countries.

The BROND project

Work on the development of the BROND-3 library started in 1998. Reactor calculations are made using the ABBN-93 data set, which is based on BROND-2 with adjustments made on the basis of analyses of integral experiments. These adjustments are being taken into account in the production of BROND-3. Validation of the data in the BROND-2 library continues, including data for the thorium cycle. There is an ISTC project for the measurement, analysis and evaluation of gamma-ray production cross-sections and spectra for the main structural materials. Cross-sections in the intermediate-energy range, to 150 MeV, are being evaluated. There is also the Russian Reactor Dosimetry File, RRDF-98, which is an update of the International Reactor Dosimetry File, IRDF-90.

The CENDL project

In China, the CENDL-3 library has been developed and is being extended and tested. It includes data for over 200 materials and has consistent data for both isotopes and elements. Data for the primary actinides are tested using the Los Alamos critical assemblies. There are also evaluations of fission yields, of data for accelerator-driven sub-critical systems and of thermal neutron capture prompt gamma-ray data (as a contribution to an IAEA co-ordinated research project).

The IAEA Dosimetry File project

An IAEA Nuclear Data Section (NDS) co-ordinated project has updated the International Reactor Dosimetry File, IRDF-90, taking into account the Russian RRDF-98 file and the Japanese JENDL/D-99 file, together with recent measurements. The new dosimetry file is called IRDF-2002.

The FENDL project

A co-operative project co-ordinated by the IAEA NDS has produced a library for fusion reactor design studies called FENDL. This contains a selection of data from the above libraries.

The ENSDF project

There are other international projects for data compilation and evaluation. The Evaluated Nuclear Structure Data File (ENSDF) project compiles radioactive decay data. There is an international network of contributors, co-ordinated by the NNDC at Brookhaven, USA. Resources are dwindling, with 90% of the evaluations now being carried out in the USA. There are no young scientists replacing those who retire from this activity in Western Europe.

3.4.2 Working Party on International Evaluation Co-operation

The NSC's Working Party on International Evaluation Co-operation (WPEC) provides a framework for the exchange of information on both evaluation and measurement activities and plans, following the merger of the Working Party on Measurement Activities with WPEC in 1999. WPEC was established in 1989 and meets annually. WPEC reviews evaluation activities so as to avoid incoherence in evaluations and unnecessary duplication. The files use a common format (ENDF-6), and a common approach is adopted to evaluation procedures, such as the representation of covariance data, which makes co-operation easier. Files are freely exchanged between projects.

The working party organises sub-groups to investigate problem areas (these are listed in Appendix 2). Most sub-group members are not members of WPEC but are chosen for their specialist expertise. There are several long-term sub-groups, one of which has the responsibility to update the High Priority Nuclear Data Request List. Seventeen short-term sub-groups have completed their work and reports have been issued as volumes in the International Evaluation Co-operation series.

As an example, Sub-group 6 on delayed neutron data recommended new data for the major actinides: energy-dependent total yields, eight group time-dependent data and energy spectra and the associated uncertainties in beta-effective calculations. In addition to the sub-group report there was a special issue of the journal *Progress in Nuclear Energy*. The reports from Sub-groups 12, 13 and 14 showed the increasing interest in nuclear data in the intermediate-energy region (above 20 MeV) and the role of theory in meeting some of these requirements. These sub-groups identified developments required in the theoretical methods.

Sub-group 18 on epithermal capture in ^{235}U resulted in a new evaluation being produced at Oak Ridge, and this has been validated by the different library projects. This improved the prediction of ^{235}U capture in thermal reactors, but also worsened the prediction of effective multiplication in some thermal systems. This led to the establishment of a sub-group on nuclear data for improved low-enriched uranium LWR reactivity predictions to investigate the reasons for the underestimation of reactivity in these thermal systems.

The evaluation and validation of several key types of data, such as the resonance region data of ^{235}U and ^{238}U, are being jointly carried out by the various data library projects. In this way the files tend to evolve towards a single library. However, advantages are seen at present in maintaining the separate projects. These are associated with, for example, differences in national priorities and sources of funding. It should be emphasised that the projects are not in competition with each other.

3.4.3 High Priority Nuclear Data Request List

The High Priority Nuclear Data Request List is produced by an international collaborative effort co-ordinated by WPEC Sub-group C. Each country formulates its own requests and these are compiled and commented on by sub-group members on the basis of their feasibility and status.

The list presently includes 12 requests for standards data, 60 for fusion technology data, 231 for fission reactor technology data, 12 for medical and industrial applications data, and 164 for intermediate-energy data. The total is close to 500 requests. The list can be viewed on-line on the NEA website. It can be modified by requesters and commented on by measurers and evaluators.

It should be kept in mind that to complete the various stages between the initial measurement plan and the data becoming available in a usable form requires several years of work. The process includes obtaining samples, making the measurement, interpreting the measurement, evaluating the ensemble of the data, processing it for use in calculations and validation.

3.4.4 The role of international organisations

The international network of data centres is comprised of the NEA Data Bank in Paris, the IAEA Nuclear Data Section (NDS) in Vienna, the US National Nuclear Data Center (NNDC) at Brookhaven, the Russian Nuclear Data Centre at Obninsk, the Nuclear Data Centre at JAERI in Japan and the Nuclear Data Centre in China. Each centre organises the compilation of measured data and acts as a focus for the work of evaluating nuclear data.

International co-operation is accepted as essential for satisfactory progress with the resources available. The network of nuclear data centres compiles the measured differential data in a computer retrieval format, EXFOR, and in the CINDA bibliographic index. Both the measured and evaluated data can be accessed on-line, and they are also made available on CD-ROM together with retrieval programs. There are also a number of editing programs available for data checking, graphical comparison of data and simple processing (e.g. JANIS and Pre-Pro).

The NEA has further developed the earlier JEF-PC program for displaying and comparing files of evaluated and measured data to produce the JANIS program. This is written in the more flexible Java language, which allows both software and web-based access to files of data and additional types of data to be displayed and compared. JANIS can be used to display radioactive decay data, fission yields and interaction cross-sections. JANIS and its associated data libraries are available on CD-ROM from the data centres, as are other editing and display programs.

Some integral data have not been compiled internationally because of proprietary considerations, but there are many data which have been compiled and made generally available. Examples include the NEA *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, the NEA International Reactor Physics Database (IRPDB), and the older CSEWG Benchmark Book. There is an IAEA initiative to collect fast reactor integral data. There exists a danger, however, of many integral data being lost because of integral measurement facilities being closed and due to results being reported only in internal documents which are not properly maintained.

Some measurement activities are carried out by international organisations. In Western Europe, IRMM at Geel in Belgium is the main centre carrying out differential cross-section measurements. Funding comes from the European Commission's Framework Programme, which also funds measurements being carried out in other laboratories such as n-TOF at CERN in Geneva. The International Institute Laue-Langevin in Grenoble, France, also provides facilities for measurements associated with high flux reactors. At the Frank Laboratory of Neutron Physics in Dubna, Russia, there are two pulsed reactors which provide powerful pulsed beams of neutrons.

The European Union, Japan and the USA established the International Science and Technology Centre (ISTC) in Moscow in 1992 to provide funding for science and technology projects in CIS countries. A number of nuclear data projects are funded in this way.

3.4.5 The role of theory in evaluation

Nuclear models are now successful at predicting the average characteristics of cross-sections for many processes above the thermal and resonance ranges. A WPEC sub-group co-ordinates developments and organises code comparisons. A long-term aim could be to link the different codes so as to provide a single source capable of calculating all reactions.

An IAEA NDS co-operative project, RIPL, is devoting effort to producing recommended parameters for use in these models. As a result, the accuracy with which they can predict cross-sections is much better understood now than several years ago. Theoretical methods are making an important contribution to meeting data requirements in the MeV energy range, when the required accuracy is within the capabilities of current models and parameters. Several computer codes have been adopted internationally, such as SAMMY (for analysing resonance region measurements), and the GNASH and ECIS nuclear model codes. It is important that these be maintained on an international basis.

3.5 Nuclear data validation studies

A major activity in the evaluation projects is validation of the evaluated data; this involves perhaps more than half of the total effort. The validation work indicates the accuracy to be expected in calculations and the need for re-evaluation of data. In some cases, the data are adjusted on the basis of the validation studies. Countries participating in evaluation projects have built up extensive databases which they use to validate a new library.

The aim is always to produce a new library that performs better than the previous one, but fortuitous cancellation of errors can sometimes make some results worse. A systematic approach is essential for locating sources of discrepancy. The worsening of calculated values of effective multiplication for some thermal reactor systems following the adoption of an improved evaluation for ^{235}U is presently being investigated.

Much of the integral data used for validation work is proprietary to the countries where the measurements were carried out or to the organisations funding the measurements. There is a danger of results being lost when a country discontinues its integral measurement programme or when staff retires, and both the NEA and IAEA NDS have initiated programmes to save integral data information.

The CSEWG Benchmark Book is a useful source of information. The International Criticality Safety Benchmark Evaluation Project (ICSBEP), a project of the NSC, is a valuable and comprehensive source of data which is being continuously updated. The NEA Data Bank, in co-operation with the Radiation Safety Information Computation Center (RSICC) at Oak Ridge, USA, also maintains the SINBAD database, which contains the specifications for a number of shielding benchmarks relevant to fission reactors, fusion devices and accelerators. Another recent NEA initiative is the International Reactor Physics Evaluation Project (IRPhEP), jointly co-ordinated by the Idaho National Engineering and Environmental Laboratory (INEEL) and the Argonne National Laboratory (ANL). The aim is to preserve integral reactor physics experimental data. The project is closely co-ordinated with ICSBEP.

Several other databases are being maintained, including the International Fuel Performance Experiments Database of axial burn-up profiles of spent PWR fuel provided by Siemens/KWU, and databases maintained by JAERI, such as the Spent Fuel Isotopic Composition Database.

3.6 Processing and reactor cell calculation methods

All of the evaluated nuclear data libraries are in ENDF-VI format, making it possible to use the same processing program for all. NJOY, developed at Los Alamos, is generally available and is widely used to produce group cross-section sets, resonance shielding tabulations and data for continuous-energy Monte Carlo codes. For the treatment of resonance structure, a different processing method is required by some codes. For example, the CALENDF code, developed by the CEA, is used to produce probability table representations of resonance region data for use in the French codes TRIPOLI, APOLLO and ECCO.

The continuous-energy Monte Carlo code MCNP (developed at Los Alamos), with the associated NJOY interface ACE, is widely used (although several organisations have their own continuous-energy Monte Carlo codes). The ENDF/B-VI.5, JEF-2.2 and JENDL-3.2 libraries have been processed for use with MCNP, and these processed libraries have been made available by the organisations which produced them for distribution via the data centres.

There are a number of group cross-section sets available from the data centres for specific applications, such as general-purpose reactor analysis, isotopic build-up, shielding, dosimetry, fusion neutronics and radiation damage applications. A standard 172-group structure for thermal reactor applications was agreed by the JEF project about 10 years ago, although there is no agreed standard representation of group averaged resonance structure data. For example, the French APOLLO code averages the resonance region cross-sections with fluxes calculated using the “fine structure equation”, whereas the UK WIMS code uses fluxes calculated for hydrogen slowing down. The ECCO code uses a fine group structure and sub-group data. A number of data sets have been produced in these group structures by different organisations, but the data sets have not always been made generally available.

The deterministic cell codes themselves are in many cases proprietary, but a version of the WIMS code has been produced by the NEA. The IAEA NDS WIMS Library Update Project (WLUP) made some further changes and produced group cross-section libraries for the code. Both 69-group and 172-group libraries have been produced from JEF-2.2, JENDL-3.2 and ENDF/B-VI.8 and also IAEA recommended libraries based on a selection from the various data libraries.

Data sets are also available to users of the codes in the widely used SCALE system (although not necessarily the most recent libraries). This system includes methods for treating resonance effects: the BONALUMI module for the Bondarenko (or narrow resonance) approximation, and the NITAWL-II module for the resolved resonance region. There are other cell codes available, such as the Canadian code DRAGON, but it is necessary for the user to develop a group library for the code (perhaps by conversion of a library from another format).

In addition, group cross-section sets produced by KAERI have been made available (based on JEF-2.2 for fast reactor applications and on ENDF/B-VI.5 for shielding applications). The data sets produced by ENEA Bologna in Italy for fusion applications have also been made available.

The European Fusion File (EFF) project has produced the EASY system, and its associated nuclear data library the European Activation File (EAF), for calculating activation in regions of fusion reactors. The data in EFF and EAF are being extended to 55 MeV so that the proposed International Fusion Materials Irradiation Facility (IFMIF) can be modelled.

There is a significant advantage in using an internationally-distributed code, such as MCNP, when associated data sets generated from the current nuclear data libraries are also being made available. It might be considered whether this advantage should be extended to selected deterministic codes which are generally available and widely used (as has been done for WIMS-D by the IAEA's WLUP).

3.7 Conclusions

Ensuring the continuity of facilities and expertise

The High Priority Nuclear Data Request List shows that there is a continuing and important need for high-accuracy nuclear data measurements. However, the number of scientists working in this field has declined over the past 20 years, and many differential and integral measurement facilities have been shut down. In Western Europe, there is a new facility at CERN. However, the measurements being made at CERN relating to nuclear technology applications, and those at IRMM in Geel, are funded by the European Commission, and there is no guarantee that the funding for these will continue long into the future.

University nuclear physics departments have made a major contribution in the past, and their input remains significant. However, nuclear data is no longer regarded by some younger researchers as at the frontiers of nuclear physics research. It can be anticipated that the contribution from universities will decline, making it necessary to consider how it can be replaced in the future. This trend is also a problem for recruiting well-trained researchers to replace those who retire.

In the USA, an important contribution is made by retired scientists who continue to make measurements and carry out research on a part-time basis. In France, retired scientists are also encouraged to play a part in guiding junior scientists and to participate in research. It is less common for retired scientists to be provided with the necessary facilities in other countries, except in universities. In some countries (an example being France) an effort has been made to train young scientists to ensure that the necessary expertise is maintained. Other countries might consider similar schemes.

There are several programmes which are monitoring the situation. Examples include the initiative being undertaken by the NSC and the FRAMATOME project on "Nuclear Expertise and Research Facilities in Europe".

Provision of samples for measurements

Obtaining suitable samples for use in measurements is becoming increasingly difficult. This applies not only to radioactive samples. Problems can arise at each stage: obtaining funding, finding suitable material, having the target fabricated, analysing the sample and disposing of it after irradiation.

Continuity of support by the international data centres

Basic measured and evaluated nuclear data are shared internationally and so it is appropriate that international organisations should support work in this area. However, the nuclear industry, design and research organisations, and government departments with responsibility for long-term planning of energy supply and waste disposal, must monitor the situation to ensure that the appropriate level of support is provided.

Nuclear data requirements

Confidence in the justification for some of the requests in lists of nuclear data requirements is not high, and it is considered that more effort should be made in this regard. This would need to be done through collaboration between users, evaluators and measurers. An example of this approach is the study being undertaken by the French nuclear data committee, CFDN. Account must also be taken of the time required to prepare for and carry out measurements and the subsequent evaluations.

Well-justified requirements should be funded by the nuclear industry, to demonstrate to the bodies that fund the facilities that the work they do is valued. Representatives of the measurement community should attend meetings of evaluation projects and take into account the views expressed there as concerns requirements and priorities.

Maintenance of nuclear data evaluation activities

The number of scientists working on the evaluation of nuclear data has declined significantly. Several countries which previously made significant contributions are no longer doing so. An exception worthy of note is the maintenance of cross-section evaluation activities in France. The effort devoted to evaluation of nuclear structure data through the Evaluated Nuclear Structure Data File (ENSDF) project has declined seriously, with a significant reduction in the contribution from outside the USA.

Validation of new libraries of nuclear data

Before users can accept new evaluations of nuclear data they must be convinced that there will be benefits. This requires extensive programmes of validation for the particular application the user is interested in. Users should ensure that the validation studies being undertaken by the data library projects cover their requirements.

Preservation of integral data

The work being done to preserve integral data should be examined to ensure that it is sufficient. At Argonne National Laboratory (ANL), laboratory documents detailing differential nuclear data measurements and associated nuclear data studies have been scanned and are available on the Internet. Other laboratories should be encouraged to do the same for both differential and integral data measurements and related studies.

Nuclear data uncertainty information

The inclusion of uncertainty information in the files of evaluated data has not progressed as rapidly as had been hoped. These are now included in JENDL-3.3 for important materials and in the EFF files, but have not yet been included for all the major materials in JEFF-3. Covariance files are included for a number of materials in ENDF/B-VI, but more needs to be done.

Production of data sets for applications

Applications data libraries have been produced from the current data files for the Monte Carlo code MCNP and for fast reactor and shielding applications (in standard group structures and using a simple Bondarenko resonance shielding treatment). Data sets for thermal reactor applications are more problematic because of the greater complexity of the required resonance shielding treatment. The IAEA WLUP has co-ordinated the production of data sets from the current data libraries for the WIMS-D code, but this has not been done for other code schemes.

The development of a standard format for applications data sets for thermal reactor calculations, including a standard resonance region treatment (perhaps based on a fine group sub-group treatment or moments representation), would help significantly in reducing the work involved in processing data for applications and would remove a source of ambiguity in benchmark result comparisons. So far, agreement has been reached only on common energy group boundaries, such as the 172 Xmas structure for thermal reactor applications and the 175-group Vitamin-J structure for shielding and dosimetry applications. Agreement on a common approach to the representation of resonance region data in applications would be desirable, although it would probably be difficult to achieve.

3.8 Recommendations

- Continued international co-operation is vital and should be encouraged. Members of the NSC should ensure that the data centres continue to have the necessary resources and that the various international co-operation projects are adequately supported by representatives from their countries.

- The measurement community, as represented at WPEC, should be asked if the present High Priority Request List (HPRL) provides the guidance needed to plan work programmes or if more needs to be done. The WPEC sub-group responsible for the HPRL should be asked whether, in its opinion, more needs to be done to put the request list on a consistent basis, ensuring that the requests are well justified and clearly defined. Members of the NSC should ensure that data requests from their countries are thoroughly vetted. The NSC should also encourage the nuclear industry to fund key measurements in order to show that they are considered important.

  – The NSC should ensure that the continuity of facilities and expertise is regularly monitored, in relation to the level required, by asking WPEC to provide periodic statements on the situation. Associated aspects of the work, such as the availability of samples, should also be reported on.

– Members of the NSC should monitor the steps being taken in their countries to ensure continuity of expertise.

– NSC members should be asked to take steps to ensure that the results and documents relating to differential and integral measurements and related studies carried out in their countries are being properly archived and maintained. Experimental results should be stored in international databases as benchmarks whenever possible. Arranging for documents to be scanned and placed on the Internet should also be considered.

– Applications data sets have been produced for the internationally available Monte Carlo code MCNP and for several other applications, including WIMS-D. The NSC should consider whether the data centres should be asked to produce additional applications data sets for selected internationally used code systems from the current nuclear data libraries.

Appendix 1

FACILITIES FOR MEASURING BASIC NUCLEAR DATA

This appendix aims to provide summary details of the major facilities which exist for the measurement of basic (or differential) nuclear data. In particular, it includes those facilities being used, or suitable for being used, for measuring nuclear data required for the design, safety assessment and operation of conventional fission reactors, fusion reactors and the associated waste management and decommissioning facilities. However, it does not claim to be a complete listing.

Measurement facilities in Western Europe

The main facilities for differential cross-section measurements of interaction data at energies below 20 MeV are at the European Commission's Joint Research Centre laboratory, the Institute for Reference Materials and Measurements (IRMM) at Geel in Belgium. Waste handling and waste transmutation (including waste management facilities) will be the major topic for the Commission's next Framework Programme. Measurements relating to the development of new innovative reactor concepts will also be undertaken, but with lower priority. Interest is slowly shifting to higher energies. But the energy range accessible at Geel (<20 MeV) is still the most important. It is planned to continue to refurbish the linear accelerator GELINA, the Van de Graaff having already been refurbished.

The n-TOF facility at CERN near Geneva is partly funded by the European Commission's Framework Programme. The measurements here complement those at Geel, and there is close collaboration between the two laboratories. Both facilities have comparable average neutron fluxes below 10 MeV, but n-TOF has a peak flux a factor of 1 000 higher. The CERN facility is thus ideal for measurements on radioactive samples, but due to its high peak flux, transmission measurements cannot be made there. Measurements of, for example, capture cross-sections are made at CERN, while total cross-sections are measured at Geel. The measurement programme for 2002 included capture in ^{197}Au, ^{56}Fe, ^{151}Sm, ^{204}Pb, ^{206}Pb, ^{207}Pb, ^{208}Pb, ^{209}Bi and ^{232}Th, and fission in ^{234}U, ^{235}U, ^{236}U and ^{232}Th. CERN's n-TOF has a lower duty cycle than GELINA, the total flux of the two machines being similar.

The n-TOF collaboration involves the partners from the European Commission contract plus some additional institutes from Central and Eastern Europe and the USA which are not eligible for the EC contract. It consists of almost 150 scientists from 40 institutes, and there are links with several laboratories which have neutron facilities. In addition to IRMM and n-TOF, laboratories endowed with Van de Graaff facilities are at the Instituto Tecnologico e Nuclear in Lisbon, Portugal, at the Bordeaux Gradignan site, France, at the Stellar Nucleosynthesis Group at IK3/Forschungszentrum Karlsruhe, Germany, and at the Legnaro National Laboratories, Italy.

At the Physikalisch-Technische Bundesanstalt (PTB) at Braunschweig, Germany, measurements are made of neutron activation cross-sections and of neutron scattering cross-sections (DX and DDX) between 6 and 15 MeV using an energy-variable compact cyclotron and neutron time-of-flight spectrometer. There is also a Van de Graaff accelerator used for producing monoenergetic neutrons and high-energy photons.

Measurements of activation cross-sections at MeV energies using radiochemical methods are carried out at the Institut fur Nuklearchemie, Forschungszentrum Julich, Germany. Interacting particles are produced using one of the two cyclotrons. Another important part of the programme at this facility is the measurement of thick-target production cross-sections of radioisotopes used in medical applications. Also in Germany, the programme at GSI Darmstadt using the heavy-ion accelerator facility and associated on-line isotope separator includes studies of the characteristics of radioactive nuclei.

At the Studsvik Neutron Research Laboratory of the University of Uppsala, Sweden, the OSIRIS radioactive ion source and on-line isotope separator facility, associated with the R2-0 research reactor, is used to study the yields and decay properties of fission products. The reactor and the facility have recently been refurbished. There are several other facilities associated with the reactor. Measurements are carried out on a commercial basis.

The mass separator LOHENGRIN at the Institute Laue-Langevin in Grenoble, France, is also used to study fission product yields. The low-energy neutron beams from the high-flux reactor can be used to measure cross-sections in the thermal energy range. There is also provision to irradiate samples in the high flux of the reactor and to obtain integral cross-section data by analysis of the composition of the samples following irradiation. Teams of scientists from different laboratories carry out the measurements.

There are several other facilities, including Van de Graaff accelerators, which are being used for relevant differential cross-section measurements. A number of facilities is available for making measurements at 14 MeV. Some of these are used for studies relating to activation and heat generation in fusion materials.

There are also some facilities for studies at energies above 20 MeV. At the Université Catholique de Louvaine (UCL) in Belgium, a cyclotron is used for neutron interaction measurements in the energy range 25 to 70 MeV. Measurements of double-differential cross-sections for light charged particle emission, e.g. (n,px), in neutron-induced reactions on both light materials (C, Al, Si) and heavy materials (Fe, Co, Pb, Bi, U-nat) have been completed recently.

In the neutron beam facility at the proton accelerator of the Svedberg Laboratory at the University of Uppsala, Sweden, monoenergetic neutron interaction measurements are carried out. Studies are made of neutron-proton scattering, charge exchange (n,p) reactions in nuclei, neutron elastic scattering on nuclei and fast-neutron fission dynamics.

Measurement facilities in Japan

Facilities operated by JAERI include the 14 MeV Fusion Neutronic Source (FNS) and a 20 MV tandem accelerator providing monoenergetic neutrons in the ranges 9-13 and 17-30 MeV. Recent measurements include activation cross-sections around 14 MeV and secondary neutron and gamma spectra for integral benchmark experiments. In collaboration with the High Energy Accelerator Organisation (KEK) in Tsukuba, a multipurpose intense proton accelerator complex is being built which will include an intense spallation neutron source for neutron scattering. Proposals are being made to the project to install beam lines and/or targets suitable for nuclear data measurements from thermal to the GeV region. If equipped in this way the accelerator and neutron source will be very useful for nuclear data measurements.

At Tohoku University in the Department of Quantum Science and Energy Engineering there is the pulsed Dynamitron, providing monoenergetic neutrons between 8 keV and 20 MeV and broad energy sources. In the university's Cyclotron and Radioisotope Centre is the K = 110 Azimuthally

Varying Field (AVF) cyclotron, providing quasi-monoenergetic neutrons to 90 MeV and activation cross-section data in the 30 to 90 MeV region. Measurements of double-differential thick target neutron yields are in progress for (p,n) and (d,n) reactions for tens of MeV protons and deuterons.

At the Tokyo Institute of Technology there is the 3 MV Pelletron, providing monoenergetic and continuous spectrum neutrons between 10 keV and ~1 MeV by means of the $^{7}Li(p,n)$ reaction. Measurements are made of (n,γ) reactions and gamma spectra for astrophysical targets and elements (interesting for nuclear transmutation) using high-efficiency and/or high-resolution gamma-ray detectors. In the Department of Nuclear Engineering of Osaka University there is the OKTAVIAN intense 14 MeV neutron source which can be operated in 2 ns pulsed or continuous mode, and is used for activation cross-section and benchmark experiments. At the Research Centre for Nuclear Physics there is a 400 MeV ring cyclotron, mainly used for nuclear physics experiments.

At the Research Reactor Institute of Kyoto University there is a research reactor which is provided with a number of beam tubes and irradiation positions. There is also a pulsed electron linear accelerator time-of-flight facility and associated lead slowing-down spectrometer. In addition, there is a 14 MeV neutron generator associated with the Kyoto University Critical Assembly (KUCA), which can be operated in either pulsed or continuous mode. There is the associated flight path for time-of-flight measurements. Recent measurements have been of the total, capture and fission cross-sections in the thermal to resonance range of the minor actinides, and total and capture cross-sections of fission product isotopes and absorber elements.

At Kyushu University there is a tandem Van de Graaff accelerator which provides polarised and un-polarised proton and deuteron beams to about 19 MeV. Studies are made of proton-induced reactions. At the KEK, a 12 GeV proton synchrotron provides protons and pions in the hundreds of MeV and GeV ranges. These beams are used for double-differential neutron emission measurements for proton- and pion-induced reactions by Kyushu University and JAERI.

There are several on-line isotope separators suitable for studying fission-product yields and decay properties: KUR-ISOL is associated with the Kyoto University research reactor; TIARA-ISOL is associated with the AVF cyclotron at JAERI, Tasasaki; and JAERI-ISOL is associated with the tandem accelerator at JAERI, Tokai.

Measurement facilities in the USA

At Los Alamos National Laboratory (LANL) the Neutron Science Center (LANSCE) has two spallation sources. At the WNR facility the neutrons are un-moderated and range from about 100 keV to 800 MeV. At the Lujan Center the neutrons are moderated and range from sub-thermal to about 100 keV. There are a number of flight paths for the time-of-flight measurements at each source. Including all of the capabilities, nuclear data measurements can be made over 16 orders of magnitude in neutron energy.

A number of enhancements have been made in recent years. The emphasis of the measurements made recently has been on total cross-sections for a range of materials in the energy range 5 to 560 MeV to an accuracy of 1% or better, and on (n,xg), (n,xp) and $(n,x\alpha)$ cross-sections to several hundred MeV. Comprehensive fission measurements have also been made in recent years, together with a new capability to measure fission prompt neutron spectra. Measurements of neutron-capture cross-sections of radioisotopes have begun. There is funding for work relating to the transmutation of waste, including requirements for advanced applications for accelerator transmutation studies.

At Oak Ridge National Laboratory (ORNL) there is the ORELA electron linear accelerator pulsed white neutron source time-of-flight facility with energies ranging from sub-thermal to 80 MeV. Improvements have recently been made to the neutron capture measurement facility. Recent measurements include ^{233}U transmission and fission, Al transmission and capture, and capture in Si, Cl and K. Construction of the new Spallation Neutron Source (SNS) began in 1999, with the planned completion date being 2006 at a total cost US$1.4 billion. This will use a mercury target and a beam of 1 GeV protons, providing the most intense neutron beam in the world. Although this facility will concentrate on neutron scattering for condensed matter research, one beam line is being proposed for nuclear data measurements and another for fundamental physics measurements.

At the National Institute of Standards and Technology (NIST) there is a research reactor with a liquid-hydrogen-cooled source and a Cf source facility. Using neutron interferometers, very accurate coherent scattering lengths have been measured at the reactor for H, D, ^{3}He, Si and ^{208}Pb. These data can be used in the evaluation of nuclear cross-sections. Cross-section measurements are carried out using facilities in other laboratories, the emphasis being on high-accuracy measurements related to the standard cross-sections, such as the H(n,n) angular distribution, ^{3}He total cross-section, and the $^{10}B(n,\alpha)$, $^{235}U(n,f)$ and $^{238}U(n,f)$ reaction cross-sections.

In the Department of Physics at Ohio University, there is a pulsed tandem accelerator producing beams of p, d, ^{3}He and ^{4}He, and there are associated flight paths. Neutron source characterisation studies have been performed for stopping targets for the reactions Be(p,n), Be(d,n) B(d,n) and Al(d,n). Measurements continue of reactions which provide information on nuclear level densities. Measurements have been made of the $F(p,\alpha\gamma)$ reaction for proton energies up to 6 MeV. This is a project designed to provide a way of looking for special nuclear materials via the (γ,n) and (γ,f) reactions. Collaboration between Ohio University, NIST and LANL recently led to a precision measurement of the neutron-proton scattering cross-section near 10 MeV for neutron standards work.

At the Renssellaer Polytechnic Institute the Gaerttner Laboratory linear accelerator facility has recently been refurbished. Neutron transmission and capture measurements have been made for Cs, Sm, ^{155}Gd and ^{157}Gd, and transmission for ^{236}U. At the University of Massachusetts Lowell there is a pulsed Van de Graaff accelerator which produces monoenergetic neutrons. Neutron elastic and inelastic scattering and total cross-sections are being measured. Recent work is $^{159}Tb(n,n'\gamma)$ below 1 MeV and total cross-section measurements for ^{235}U, ^{159}Tb and ^{169}Tm from 200 to 400 keV. At the Colorado School of Mines there is a small Cockroft-Walton accelerator used to generate nuclear data for astrophysics, mainly regarding low-energy proton and deuteron reactions on very light nuclei.

There is close co-operation both nationally and internationally between scientists working in the different laboratories, with the facilities being used by international teams. For example, scientists from Argonne National Laboratory, Ohio University and NIST have co-operated on measurements made in laboratories in the USA, Western Europe and Japan. These include the measurement of gamma rays from the reaction $^{19}F(p,\alpha\text{-}\gamma)^{16}O$, which are suitable for materials interrogation. Scientists at the Colorado School of Mines carry out charged-particle measurements using the Holifield Radioactive Ion Beam Facility (HRIBF) at ORNL.

Measurement facilities in China

At the China Institute of Atomic Energy there is a tandem Van de Graaff accelerator producing pulsed beams of deuterons and protons from which neutrons are produced using deuterium or tritium gas targets. There are also facilities at Peking University Institute of Heavy Ion Physics (Van de

Graaff), at Sichuan University and at Lanzhou University (14.7 MeV). Measurements have been made of double-differential (n,n) and (n,α) cross-sections, activation cross-sections, gamma spectra and fission product yields.

Measurement facilities in Russia

At the Institute of Physics and Power Engineering in Obninsk, there are a number of facilities:

- A Van de Graaff, providing a continuous or pulsed beams of protons or deuterons, producing monoenergetic or broad spectrum neutrons.
- A cascade accelerator of protons or deuterons producing monoenergetic neutrons.
- A tandem accelerator of protons or deuterons producing monoenergetic neutrons.
- A 14 MeV neutron source, pulsed or continuous.
- A 30 MeV electron accelerator.

Recent measurements include the fission cross-sections of Cm isotopes, fission product yields in ^{232}Th and ^{237}Np, delayed neutron yields and relative abundances as a function of incident neutron energy, prompt fission neutron spectra, and total cross-sections and inelastic scattering.

At the Frank Laboratory of Neutron Physics in Dubna there are two pulsed reactors which provide powerful pulsed beams of neutrons, a sub-critical reactor driven by an accelerator, and a pulsed prompt critical reactor. Recent measurements include delayed neutron yields. It is planned to build a new high-flux facility, INES.

Measurement facilities in Central and Eastern Europe

There are a number of nuclear data activities in Central and Eastern Europe. At the ATOMKI Institute at the University of Debrecen, Hungary, measurements are made of nuclear data for nuclear techniques, especially yields of reactions to produce unstable isotopes for medical applications and activation analysis. Scientists from a number of Central and Eastern European countries (Bulgaria, Hungary, Romania and Slovakia) participate in nuclear data experiments at IRMM Geel through the PECO collaborations.

Appendix 2

SUB-GROUPS OF THE WORKING PARTY ON INTERNATIONAL EVALUATION CO-OPERATION (WPEC)

Long-term sub-groups of WPEC

- Sub-group A – Nuclear model codes *(Co-ordinators: M. Chadwick and A. Koning)*.
- Sub-group B – Formats and processing *(Co-ordinator: A. Trkov)*.
- Sub-group C – High priority request list *(Co-ordinator: T. Fukahori)*.
- Sub-group D – Nuclear data standards *(Co-ordinator: A. Carlson)*.

Short-term sub-groups of WPEC which have completed their tasks

- 1. Comparison of evaluated data for ^{52}Cr, ^{56}Fe and ^{58}Ni *(Co-ordinator: C.Y. Fu)*.
- 2. Generation of covariance files for iron-56 and natural iron *(Co-ordinator: H. Vonach)*.
- 3. Actinide data in the thermal range *(Co-ordinators: H. Tellier and H. Weigmann)*.
- 4. ^{238}U capture and inelastic cross-sections *(Co-ordinator: M. Baba)*.
- 5. Plutonium-239 fission cross-section between 1 and 100 keV *(Co-ordinator: E. Fort)*.
- 6. Delayed neutron data *(Co-ordinator: A. d'Angelo)*.
- 8. Present status of minor actinide data *(Co-ordinators: T. Nakagawa and H. Takano)*.
- 9. Fission neutron spectra *(Co-ordinator: D. Madland)*.
- 10. Evaluation method of inelastic scattering cross-sections for weakly absorbing fission product nuclides *(Co-ordinator: M. Kawai)*.
- 12. Nuclear models to 200 MeV for high-energy data evaluations *(Co-ordinator: M. Chadwick)*.
- 13. Intermediate energy data *(Co-ordinators: A. Koning and T. Fukahori)*.
- 14. Processing and validation of intermediate energy data files *(Co-ordinator: A. Koning)*.

- 15. Cross-section fluctuations and self-shielding effects in the unresolved resonance region *(Co-ordinator: F. Fröhner)*.

- 16. Effects of shape differences in the level densities of three formalisms on calculated cross-sections *(Co-ordinator: M. Chadwick)*.

- 17. Status of pseudo-fission product cross-sections for fast reactors *(Co-ordinator: H. Gruppelaar)*.

- 18. Epithermal capture of ^{235}U *(Co-ordinator: C. Lubitz)*.

Short-term sub-groups of WPEC with tasks in progress

- 19. Activation cross-sections *(Co-ordinator: A. Plompen)*.

- 20. Evaluation and processing of covariance data *(Co-ordinator: T. Kawano)*.

- 21. Assessment of neutron cross-section evaluations for the bulk of fission products *(Co-ordinator: P. Oblozinsky)*.

- 22. Nuclear data for improved LEU LWR reactivity predictions *(Co-ordinator: A. Courcelle)*.

Chapter 4

FUEL BEHAVIOUR

4.1 Introduction

The NEA Nuclear Science Committee (NSC) has long recognised the need for international co-ordination on improving knowledge of important scientific issues related to fuel behaviour. Consequently, the Task Force on Scientific Issues of Fuel Behaviour (TFSFB) was set up by the NSC in late 1993. The initial objective of the task force, as endorsed by the NSC, was to identify areas of high priority to NEA member countries which would benefit from international co-ordination and co-operation on studies of the basic underlying phenomena of fuel behaviour under normal operating conditions. The task force was also asked to advise the NSC on developments regarding data, models and experiments which were needed to meet the requirements for better understanding of fuel behaviour and for improved predictive models.

The main findings and conclusions of the task force were summarised in a 1995 report [1], discussed in Section 4.2, which set the scene for a series of actions taken by the NEA over the succeeding years. These initiatives and activities are reviewed below, with the aim of highlighting the achievements already made as well as indicating the directions where there is benefit in continuing and extending the current activities in the future.

4.2 Important issues for fuel performance modelling

The first action of the TFSFB was to identify topics and potential authors to review the then-current state of understanding, and to identify priority areas where further work was required. This was accomplished and a report addressing the following topics was issued [1]:

- *Thermal performance* – the calculation of fuel temperatures, the effect of design parameters and the effect of irradiation.
- *Fission gas release* – release under different operating regimes and the effect of high burn-up.
- *Fission product swelling* – the distinction between inexorable solid fission product swelling, which is predominantly a function of burn-up and fission product accumulation, and gaseous swelling, which is also dependent on high-power operation.
- *Stress corrosion cracking* – the effects of stress, time and fission product release on the propensity for clad failure from this mechanism.
- *Water chemistry* – current practice and monitoring; its effect along with zirconium composition and final treatment on clad corrosion.
- *Hydrogen in cladding* – its distribution and measurement.

- *Failed fuel* – detection and modelling of degradation processes.

- *Spent fuel* – long-term storage under wet and dry conditions.

- *Quality assurance* – as applied to materials, experimental tests and data production.

The report concluded that although fuel in nuclear reactors had proven to be highly reliable in its performance and safety, this was often supported by rather generous boundary reactor operating conditions. In the future the requirements would become more onerous. The economic requirement to increase efficiency meant that fuel performance would need to be calculated accurately on a best-estimate basis. To do this, good fuel performance computer codes validated by high-quality data would be required. As a result of this report, the task force arrived at the following recommendations for future NEA activities:

- Countries and organisations should be encouraged to make special research efforts to reduce uncertainties in modelling specific aspects of fuel behaviour, namely:

  – Thermal performance and calculating fuel temperatures.

  – Fission gas release.

  – Fission product swelling and creep of UO_2.

  – Thermo-mechanical behaviour.

  – High burn-up fuel behaviour in transient conditions.

- As a first step, a review should be made of existing data.

- A public-domain database should be assembled of well-qualified experiments and data which could be used for model development and code validation. This database should be organised and maintained by the NEA Data Bank.

- International topical meetings covering high-priority issues should be organised by the NEA in co-ordination with the IAEA. Of particular interest were workshops on thermal performance, on fission gas release (FGR), and on pellet-clad mechanical interaction (PCMI).

4.3 Identification of available experimental data

The second report commissioned by the TFSFB and issued by the NEA was a review of existing data that could be made available to fulfil the aim of improving code performance. This report [2] concentrated primarily on data produced in the joint programme carried out by the Halden Reactor Project, Norway.

Conditions in the Halden reactor are particularly well suited to studies of fuel performance. The boiling conditions ensure a constant coolant temperature, and hence a well-defined boundary condition from which to assess thermal performance from measurements of centreline fuel temperatures. Also, the low system pressure of 32 bar and the low fast flux ensure negligible clad creep down, thus removing one parameter from through-life assessments of fuel dimensions and temperature. However, when more prototypic conditions are required, dedicated in-pile loops are available to simulate the thermal-hydraulic conditions of temperature and pressure, as well as the neutron flux spectrum, in PWRs, BWRs and, most recently, advanced gas-cooled reactors (AGRs), as operated in the UK.

The report outlined some of the experiments that have been performed in the Halden reactor and the most important data that they have provided which are available for use in fuel performance studies. As to be expected in a project that has been producing data for many years, several reviews of individual topics have already been written. The task force's report made extensive use of these, with the aim of demonstrating the extent of the information available collated in a form that is most useful for the development of fuel performance codes and their validation.

The report divided the data into a number of categories depending on the use to be made of them. In this respect, the data were not of universal interest. The simplest division was into three broad categories, as follows:

- *Data useful for model development and validation* – e.g. radial flux depression, fuel creep, fuel densification and swelling, and clad creep down. These are the "unsung needs" of the code developer, not often apparent in code predictions but essential to obtain a good description if reliable predictions are to be obtained. Very often these data require special measuring techniques and non-prototypic rig designs.

- *Data of direct relevance to licensing requirements* – e.g. fuel temperatures, stored heat, fission product release and rod internal pressure, and waterside corrosion. Such data are of particular use for validation purposes.

- *Data for fuel development and optimisation* – e.g. performance of fuel variants and new cladding materials, effects of changes in fuel design. These data are of most interest to fuel vendors in developing and supporting new products.

Within these categories, data were discussed under the following phenomenological headings:

- Radial flux depression.

- Thermal performance.

- Fuel densification and swelling.

- UO_2 grain growth.

- Fission product release.

- Clad properties.

- Pellet-clad mechanical interaction (PCMI).

- Integral behaviour.

- High burn-up effects.

The report reviewed experiments where single parameters were isolated for study (e.g. fuel centre temperature as a function of fuel pellet to clad gap), as well as experiments which addressed integral behaviour, where several effects were studied simultaneously (e.g. fuel temperature, fuel and clad axial extension and rod internal pressure as a function of power and burn-up).

In addition to Halden Project data, the report briefly reviewed data from certain ramp tests conducted at Studsvik, Sweden, and from three fission gas release projects carried out at Risø Laboratories, Denmark. These had already been shown to be compatible with Halden experiments, particularly in the areas of pellet-clad interaction (PCI)/stress corrosion cracking (SCC) failure and post-irradiation examination (PIE) of ramp-tested fuel.

Despite the comprehensive nature of the data available, as outlined in the report, it was clear that there were a number of areas where further experiments were necessary. With the move to higher discharge burn-up, it had become necessary to extend the data to cover the extremes of burn-up and power expected in commercial reactors. This implied a progressive extension of well-qualified data to burn-ups in excess of 70 MWd/kgU. New effects were being discovered at these levels of burn-up and further data were required on such topics as "the rim effect", the effect of burn-up on fuel thermal conductivity, clad corrosion and hydriding, and clad mechanical properties.

Although there were data on temperature changes during reactor scrams, for transient analysis there was a need for data on the evolution of temperatures during rapid power changes. There were (and still are) few data on this topic, and there is clearly a benefit to be claimed in reactivity faults; for example, to exploit the lag between power increase and consequent increase in temperature.

The report supported the TFSFB initiative to assemble an International Fuel Performance Experiment (IFPE) database, and recommended the inclusion of experimental data identified in the report from all three international programmes.

4.4 International Fuel Performance Experiment (IFPE) database

From the outset it was recognised that the IFPE database should cover all commercially operated thermal reactor systems, and that it should include both prototypic data, i.e. originating from power reactor irradiations with pre- and post-irradiation characterisation, and data from test reactor experiments with in-pile instrumentation and PIE, exploring normal and off-normal behaviour. It was recognised that experiments have been performed where the data remain of commercial interest; for example, many of the details of modern MOX fuel performance remain proprietary to the manufacturers, and it was not the intention to compromise such arrangements. However, it was considered that zircaloy-clad UO_2 pellet fuel was largely a standard product and that, as such, release of what was previously proprietary data could only benefit the nuclear community at large.

A particular aspect of the compilation was the inclusion of data generated within internationally sponsored research programmes whose confidentiality agreements had expired. Such data, although available in principle, had not been widely used. The inclusion of such data was of particular importance where the originating organisation had changed its terms of reference. For example, Risø Laboratories in Denmark no longer performs nuclear research and finds difficulty in resourcing the supply of information from its three fission gas release projects. In such cases, the danger exists of losing access to the data altogether.

4.4.1 Extent of parameters included

The database is restricted to data on thermal reactor fuel performance, principally standard product zircaloy-clad UO_2 fuel, although the addition of advanced products with fuel and clad variants is not ruled out. Recent additions to the database include MOX fuel, $(U,Gd)O_2$ fuel, and defective fuel. Data encompass normal and off-normal behaviour, but not accident conditions entailing melting of fuel and clad, resulting in loss of geometry.

Of particular interest to fuel modellers are data on fuel temperatures, fission gas release (FGR), clad deformation (e.g. creep-down, ridging), and mechanical interactions. In addition to direct measurement of these properties, every effort is made to include PIE information on the distribution of grain size and porosity, and electron probe micro-analysis (EPMA) and X-ray fluorescence (XRF) measurements on caesium, xenon, other fission products and actinides.

Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. For example, cases are included which specifically address the effect of gap size and release of fission gas on fuel-to-clad heat transfer. Also in the context of thermal performance, the effect of burn-up on UO_2 thermal conductivity has been addressed. This is illustrated by cases where fuel temperatures have been measured throughout prolonged irradiation and at high burn-up, where sections of fuel have been re-fabricated with newly inserted thermocouples. Regarding fission gas release, data are included for normal operations and for cases of power ramping at different levels of burn-up for fuel supplied by several different fuel vendors. In the case of power ramps, the data include cases where in-pile pressure measurements show the kinetics of release and the effect of slow axial gas transport due to closed fuel-to-clad gaps. Supplementing these in-pile studies, there are data sets of out-of-pile annealing studies measuring FGR under well-defined conditions of temperature and time.

To date, data sets of about 418 rods/samples from various sources encompassing PWR, BWR, PHWR and VVER reactor systems have been included in the database (see Table 4.1).

Table 4.1. Data included in the IPFE database

Halden irradiated IFA-432	5 rods
Halden irradiated IFA-429	7 rods
Halden irradiated IFA-562.1	12 rods
Halden irradiated IFA-533.2	1 rod
Halden irradiated IFA-535.5 and .6	4 rods
Third Risø Fission Gas Release Project	16 rods
Risø Transient Fission Gas Release Project	15 rods
SOFIT VVER Fuel Irradiation Programme	12 rods
High Burn-up Effects Programme	81 rods
VVER rods from Kola-3	32 rods
Rods from the TRIBULATION programme	19 rods
Studsvik INTER-RAMP BWR Project	20 rods
Studsvik OVER-RAMP PWR Project	39 rods
Studsvik SUPER-RAMP PWR Sub-programme	28 rods
Studsvik SUPER-RAMP BWR Sub-programme	16 rods
Studsvik DEMO-RAMP I – BWR	5 rods
Studsvik DEMO-RAMP II – BWR	8 rods
CEA/EDF/FRAMATOME Contact 1 and 2	3 rods
AEAT-IMC NFB 8 and 34	22 samples
CEA/EDF/FRAMATOME PWR and OSIRIS ramped fuel rods	4 rods
CENG defect fuel experiments	8 rods
CANDU elements irradiated in NRU	36 rods
Siemens PWR rods irradiated in GINNA	17 rodlets
CEA failed PWR rods irradiated in SILOE: EDITH-MOX 01	1 rod
CNEA six power ramp irradiations with (PHWR) MOX fuels	5 rods
BN GAIN $(U,Gd)O_2$ fuel	4 rods
INR Pitesti – RO-89 and RO-51 CANDU fuel type	2 rods
Total	**422 cases**

4.4.2 Format of files

The criterion adopted for the file format was one primarily of simplicity. It was considered that users should be able to read the files independently of commercial software. It was recognised that the majority of codes were written in FORTRAN and therefore all files are in simple ASCII format for easy interrogation by file editors. Even text files are in this format despite the limitations imposed by this approach. The adopted ASCII format does not preclude reformatting into a commercial database system sometime in the future if so desired. Each data set has the following common elements:

- *Summary file*. This is a text file which describes the purpose of the experiment or test matrix and the scope of the data obtained.

- *Index file*. This is also a text file, listing all file titles with a brief summary of their contents.

- *Pre-characterisation*. Includes information on the fuel pellets and cladding used, their manufacturing route, dimensions and chemical composition including impurities. For the fuel, this is augmented by details of enrichments, porosity distribution, re-sintering test data and microstructure. For the cladding, additional information includes mechanical properties, corrosion characteristics and texture, as and when available. Details of the fuel rod geometry include relevant dimensions, fuel column length, weight, fill gas composition and pressure. Details of reactor irradiation conditions are also included.

- *Irradiation histories*. All histories are in condensed form with care to ensure that all important features are preserved. Where there was a significant axial power profile the history is provided in up to 12 axial zones. For each time step, the data are provided are time, time increment over which power was constant, clad waterside temperature and local heat rating, for the prescribed number of axial zones. Information is provided to calculate the fast neutron flux and its spatial variation if the information is available.

- *In-pile data*. When applicable, separate files tabulate data from in-pile instrumentation as a function of variables such as time, burn-up or power. For example, IFA-432 fuel rods were equipped with centreline thermocouples and cladding elongation detectors. Files were created tabulating temperatures at constant powers of 20 and 30 kW/m as a function of local burn-up throughout life. At several times during the irradiation, at approximately 5 MWd/kgU intervals, temperature is tabulated against local power during slow ramps. In this case, the data are reproduced directly from the original files where signals were logged every 15 minutes. A similar procedure was adopted for clad elongation measurements; during short periods of variable power, 200-5 000 hours duration, elongation is tabulated against rod average power showing cases where there was pellet-clad mechanical interaction.

- *PIE data*. Where such examinations were made, data are recorded either in tabular or text form. Dimensional data include axial and diametral dimensions before and after irradiation, and post-irradiation ridge heights where available. Data on fission gas release include rod averaged values obtained by puncturing and mass spectrometry, local values from whole pellet dissolution, and across pellet spatial distributions as measured by gamma scanning, EPMA and XRF. Before and after irradiation porosity and grain size distributions are given, as is the radial position for the onset of grain boundary porosity where such measurements have been made from metallographic examination.

All data files are held centrally on the NEA Data Bank computer system from which all files are dispatched. This single source for distribution is necessary for quality assurance purposes, particularly for the tracking and release of upgraded or corrected files. With the NEA's experience of data bank management, this arrangement ensures long-term availability of the service.

Often PIE data are only available in graphical form or as photomicrographs which are difficult to preserve in ASCII file format. For this reason, the possibility of scanning figures and photographs for storage and retrieval using the medium of the CD-ROM was investigated. As a result, all reports available on the data sets compiled to date have been scanned and copied onto a single CD-ROM. The files are all in PDF format and can easily be read and printed using widely available software.

4.5 International workshops

4.5.1 Thermal performance

The initiative to hold international workshops commenced with the Thermal Performance of High Burn-up LWR Fuel Seminar held 3-6 March 1998 at CEA Cadarache, France. The workshop was co-organised by the NEA in co-operation with the IAEA and hosted by the Département d'Études des Combustibles. The meeting was attended by 66 experts from 19 countries and four international organisations [3].

The first two sessions were devoted to fuel thermal conductivity data and thermal conductivity modelling. This was then followed by a first panel discussion, chaired by Hubert Bairiot and Louis-Christian Bernard, addressing emerging thermal conductivity questions. The third session covered fuel clad gap evolution modelling, and was followed by a second panel discussion on gap evolution and heat transfer questions, chaired by Gary Gates and Marc Lippens. The two final sessions covered experimental databases and advances in code development on thermal aspects.

A final panel, chaired by J.A. Turnbull and Daniel Baron, reviewed the conclusions of the two panels on thermal conductivity and gap conductance and summarised the key conclusions of the seminar. These conclusions are described below.

Fuel thermal conductivity

- *Fuel thermal conductivity correlations*. The initial thermal conductivity correlations are well established for almost all fuel oxides loaded in commercial reactors. The influence of parameters such as temperature, stochiometry, plutonium content and gadolinium content are relatively well known. Some models are able to account for all these parameters simultaneously.

- *MOX fuels*. The non-homogeneity of MOX fuel can be accounted for using mathematical homogenisation techniques. However, even for high Pu content the fuel thermal conductivity is close to the uranium oxide fuel conductivity providing O/M is close to 2.000. Participants seemed to agree on a degradation of 4-5% per 10% Pu. However, it has been shown that a deviation from stochiometry has a stronger effect.

- *The burn-up effect*. The burn-up effect on fuel thermal conductivity has been assessed up to 80 MWd/KgU, thanks to in-pile centreline temperature measurements and out-of-pile thermal diffusivity measurements. There is some agreement between these two methods.

- *Specific heat capacity (Cp) variation with burn-up*. Analysing shutdown temperature records from the Halden reactor shows that a slight increase of Cp with burn-up is likely. However, it is assumed that Cp variation with burn-up can be neglected. Experimental data expected in the near future will certainly allow this assumption to be checked.

- *Heat transfer in the rim*. The evolution of rim thermal conductivity is not yet clearly known. The thermal degradation due to the onset of numerous micrometer porosities could be balanced by the cleaning of the matrix subsequent to this porosity build-up. In order to investigate the net effect of the rim structure, samples irradiated to high burn-up at temperatures lower than 800 K are required. Other questions are still open about the rim fission gas release and the rim volume variation during the porosity build-up.

- *Further needs*. There is a lack of data on burn-up degradation at temperatures higher than 1 800 K. A decrease in the conductivity improvement due to electronic heat transport is likely with high burn-up. This effect has already been observed when increasing gadolinium content.

Gap conductance

- *Pellet fragmentation and relocation*. The stochastic nature of this phenomenon prohibits analytical modelling of gap conductance. Reliance must be placed on benchmarking models against in-pile data. There is no general consensus as to a definitive formulation for gap conductance. Fortunately a large database exists and providing this is employed, there are limited difficulties in formulating an adequate gap conductance model which correctly reflects the effects of gap dimension, fill gas composition and pressure.

- *Surface roughness*. In principle this should be an important contributor to gap conductance by its effect of inhibiting heat flow. In practice, at both the beginning of line and at high burn-up, there appears to be little effect on heat transfer between fuel pellet and cladding.

- *Formation of inner clad oxide layer*. At high burn-up, when the fuel and clad have been in intimate contact for some time, an inner oxide layer is formed, 6-10 microns thick. It has been shown that this has a complex structure, which can include zirconium oxides, uranium and caesium, depending on the heat rating and burn-up. It is supposed that its contribution to gap heat transfer is small and if anything beneficial, as it would tend to eliminate effects due to surface roughness and misaligned pellet fragments.

- *Closed gap conductance*. There is evidence to suggest that under these conditions, heat transfer is good and substantially independent of the fill gas composition and pressure, since it corresponds to a solid bound between fuel and cladding. There is some debate as to whether or not the conductance depends on interfacial pressure.

4.5.2 Fission gas release

The second workshop in the series was the *Fission Gas Behaviour in Water Reactor Fuels Seminar*, held 26-29 September 2000, also at CEA Cadarache, organised and hosted under the same arrangements. The seminar was attended by about 100 participants from 24 countries, representing 46 organisations. The objective of this seminar was to emphasise more recent developments, from both the experimental and modelling points of view. The areas covered by the seminar included diffusion coefficient of rare gases including helium, properties of bubbles, the effect of irradiation on re-solution and diffusion, FGR as a function of operating conditions, modelling and code validation [4].

The meeting was divided into four sessions over three days of presentations. The sessions were on feedback from experience, basic mechanisms, analytical experiments, and industrial modelling and software packages. On the final morning there was a panel discussion at which the findings of each session were presented and discussed. In view of the maturity of the subject, each presentation addressed three questions:

- What do we know and what is new?
- What do we not know, what are the uncertainties and what remains unclear?
- What initiatives should be implemented in the future?

From the first session, *Feedback from Experience*, it was concluded that there was an enhancement of fission gas release at high burn-up (>50 MWd/kgU), both in steady-state operation and in transient overpower, but there was no accepted phenomenon to explain this. One result of this was that the well-established Halden 1% FGR criterion relating fuel centreline temperature to burn-up overestimated the onset temperature at these levels of burn-up. That is, the temperature for the onset of release at high burn-up was *lower* than predicted by this criterion.

It has been observed that high burn-up fuel contains a region of restructured fuel microstructure close to the pellet periphery, the so-called rim structure. It was tempting to see this as a reason for the enhanced release, but a direct correlation could not be made. Indeed, a measure of the ratio of released krypton and xenon isotopes implied that their origin was the hot central regions of the fuel and not the colder plutonium-rich restructured region in the pellet rim. The implication of this observation was that the restructured rim served mainly to increase the thermal resistivity of the fuel, thus increasing fuel temperatures.

A discussion of this dilemma continued into the second session, *Basic Mechanisms*. Investigations have shown that the rim structure consists of a refinement of the original ~10 micron diameter grains into ~0.1 micron diameter grains and a micron-size population of bubbles which could account for a swelling up to ~10 vol.%. In addition, the fission gas concentration in the matrix falls to a low level. The impression of most researchers was that the shortfall in fission gas resided in the bubbles and was not released from the fuel; however, the exact distribution of gas remained uncertain.

The fission gas release from the rim was not more than 15-20%, showing a high retention capacity of this restructured material. The mechanism of rim formation was not clear. One suggestion was that the build-up of irradiation damage caused the grain refinement and that this was followed by the collection of gas as porosity increased. Alternatively, it was postulated that the increased porosity developed first and that the sub-micron grains were nucleated from the pore surfaces. It remained a matter of debate whether this rim restructuring was beneficial or detrimental.

Several presentations addressed FGR from the rim both in steady state and in transients such as a reactivity-initiated accident (RIA), concentrating on mechanisms and experimental observations. Measurement of krypton and xenon isotopic ratios in released gas was clearly a valuable technique to determine the relative contribution of the rim region to FGR, and application to RIA tests had proved informative. It was concluded that although details were known of this new high burn-up structure at the micron level, it was clear that a goal for the future was to understand it at the nanometer or atomic level, and that there was scope for simulation studies using high-energy accelerators to reproduce the restructuring without the constraints of time and radioactivity.

A major contribution to the release process was known to be single gas atom diffusion, and quite satisfactory models could be constructed using this along with irradiation resolution from grain boundaries. Many models employed the diffusion coefficient formulated by Turnbull, White and Wise as described at an IAEA meeting at Preston, UK, in 1988. Here the authors gave a two-term diffusion coefficient, with a third low-temperature term to be applied close to surfaces and used for short-lived radioactive species. There was much discussion on this topic in the meeting and it transpired that White has an alternative description using only the two high temperature terms with a fractal treatment of the surface to account for the difference in kinetics between long- and short-lived species. Subsequent to the meeting this was published in open literature.

Extensive study of retained caesium and xenon using EPMA carried out at the European Commission's Institute for Transuranium Elements (ITU) in Germany showed that, systematically, caesium diffusion was some three times slower than for the rare gases. This was an important result since in many accident calculations it is assumed that their diffusivities are comparable; this assumption is therefore reassuringly pessimistic.

In addition, it was known from electron microscopy that intragranular bubbles are formed within grains, and intergranular bubbles and tunnels along grain boundaries. Theoretical studies of krypton atoms in the UO_2 matrix have concluded that both krypton and xenon are insoluble, thus casting doubt on the existence of thermal resolution as a means of moderating absorption of single gas atoms into intragranular bubbles. In the absence of intragranular bubble mobility, such a mechanism was used in several models in order to predict substantial release in transients.

It was agreed that observations support only slow or limited random migration of small bubbles, but a directed movement towards grain boundaries due to a vacancy concentration gradient was an interesting proposal from Evans, UK. If this mechanism makes a significant contribution, the resulting microstructure would be quite distinctive and amenable to future testing. Topics which required further attention were the re-solution of gas atoms from grain boundary bubbles, particularly for large-grain fuel where the grain size was greater than the fission fragment range, and the response of grain boundary porosity in fast transients.

It was clear that there was much more information available on these features than currently accessible through the open literature. There was general support for a more open distribution of information, and all participants were urged to publish as much information as possible in the interests of a general improvement in models and their application to safe reactor operation. It was noted that although the level of fundamental research in many establishments had been reduced, there was still original and comprehensive work going on in ITU, for example, to improve investigation techniques (e.g. microhardness tests, lattice parameter measurements). Also, there was a refreshing intake of young engineers and scientists within CEA Cadarache. The participants looked forward to publication of new and original work from these quarters.

As a digression from fission gas, it was clear that helium generation in long-term storage of high burn-up fuel can pose a significant problem, in particular for high burn-up MOX fuel. Further data were required on low-temperature helium diffusion coefficients.

The session entitled *Analytical Experiments* started with two presentations describing a novel method of determining the disposition of gas between the matrix and the grain boundaries. Such studies complement those performed with XRF and EPMA, and provide vital information for modellers. The techniques still needed refining, particularly regarding the effect of small intragranular bubbles on the results, but promised an evaluation of the grain boundary capacity for gas prior to and during interlinkage. As mentioned previously, data have been obtained at Halden from high burn-up fuel which suggest that the temperature for the onset of FGR is lower than previously expected. It was clear that further data were required to substantiate this observation. In addition, there was a need to obtain further data on MOX fuel to compare it with UO_2 fuel performance under identical conditions.

Enhanced release at high burn-up could cause the rod internal pressure to exceed the coolant pressure, and the effect of this required attention. Experimental data suggested that the gap does not reopen by clad creep-out and positive feedback does not occur. However, the data were sparse and further experiments were necessary.

Both mechanistic and empirical approaches to FGR modelling were presented in the session on *Industrial Modelling and Software Packages*. It was clear that empirical modelling can be very valuable, but is limited in applicability and is essentially only valid within the confines of parameters and irradiation conditions covered in the database on which it is developed. Modellers were also cautioned about employing multiple mechanistic models and expecting to obtain good predictions by using appropriate fitting parameters. There is a case for independent assessment of mechanistic models and their supporting data before they are considered for inclusion within fuel performance codes.

The requirement for further data comparing UO_2 and MOX fuel behaviour was re-affirmed in a paper by Struzik of CEA which showed that there was a significant difference in behaviour in ramp tests above 30 MWd/kgU. Below 30 MWd/kgU the differences in behaviour could be explained in terms of the radial power profile, but at higher burn-up FGR and swelling in MOX fuel was greater than expected.

In conclusion, there was unanimous agreement that the seminar had been a success, with an excellent range of presentations which initiated extensive and in-depth discussion. It was clear that the topic of FGR modelling was in a mature state, with agreement about most of the important mechanisms contributing to the phenomena involved. High burn-up and MOX fuel behaviour are the current challenges, where there is incomplete understanding, but the meeting produced some good suggestions where future work should be focussed. Participants were urged to publish their work in the open literature to help others working to ensure the continued safe operation of commercial reactors.

4.5.3 Pellet-clad interaction

The third and final workshop is to be on the topic of pellet-clad mechanical interaction (PCMI), as this was deemed the least advanced area of modelling. This meeting is scheduled for late 2003, and will again be held at CEA Cadarache.

4.6 Survey of safety-related activities in NEA member countries

While the TFSFB covers mainly scientific issues underlying fuel behaviour in normal and transient operating conditions (see Appendix 3), fuel safety issues are addressed within the work programme of the NEA Committee on the Safety of Nuclear Installations (CSNI). NSC and CSNI co-operate and co-ordinate activities in the area of fuel behaviour.

Under the auspices of CSNI a report was issued in 2002 reviewing the status of safety-related research activities in NEA member countries [5]. The report is in response to a request to members of the Special Experts' Group on Fuel Safety Margins (SEGFSM) to compile information about ongoing and planned fuel safety research in NEA member states, with the aim of providing CSNI with an overview of related international R&D programmes and projects, and identifying current and future needs and priorities.

The report is based on a questionnaire distributed to SEGFSM members in October 2000, requesting them to identify fuel safety research programmes and to provide information on achievements and future plans. The questionnaire required respondents to provide information on the ongoing R&D programmes under the following headings:

A. Title.

B. Research laboratory/sponsor(s).

C. Objectives/goals.

D. Status of work.

E. Brief description/presentation of the main results achieved.

F. Future plans.

G. References.

Replies were received from organisations in the following countries: Canada, Czech Republic, France, Germany, Hungary, Japan, Korea, Norway (Halden Reactor Project), Switzerland, United Kingdom and USA. The report is based only on the information provided in the replies received. As a consequence it cannot be viewed as comprehensive – programmes may well be in progress in addition to those detailed. It is also possible that the detailed results of some programmes may remain proprietary and therefore not available in the short term.

The report is organised in topic sections relating to: fuel and clad studies; integral fuel rod tests and PIE, loss of coolant accident (LOCA) and RIA studies including whole rods and bundles as well as single effects studies of fuel and cladding; code development for both steady state and transient fuel behaviour; thermal-hydraulics; reactor physics codes; and severe accident studies.

The main issues for the current generation of reactors are those of high burn-up performance in normal operations, in LOCA and in RIA conditions. The main goal for the industry is to consolidate the safety issues in order to bring all countries up to a licensed discharge burn-up of ~60 MWd/kgU, and possibly 65 MWd/kgU. The principal issues requiring attention can be broken down as follows:

- Normal operation:
  - Fission gas release and rod over-pressure.
  - Properties of the high burn-up structure (HBS) at the pellet rim, its effect on thermal performance and fission gas release.
  - Cladding oxidation, hydriding and embrittlement.
- Loss of coolant accidents (LOCAs):
  - Possibility of fuel slumping into the ballooned region; the effect of fuel-clad bonding at high burn-up.
  - Increase of pressure in the ballooned region due to fission gas release from slumped fuel.

  - Response of irradiation hardened and hydride embrittled cladding.

  - Review of the 17% equivalent clad reacted (ECR) criterion.

- Reactivity-initiated accidents (RIAs):

  - PCMI loading mechanism(s) on the cladding, the effect of the HBS at the pellet rim.

  - Effect of HBS on fuel dispersal.

  - Response of embrittled cladding to transient PCMI.

In addition to these, for those countries that load both MOX and UO_2 fuel assemblies, there is a requirement to bring the MOX fuel database to the same level as that for UO_2 fuel, with the aim of treating MOX fuel indistinguishably from UO_2 fuel as far as safety is concerned.

The survey of international research programmes outlined in the report demonstrates the large amount of activity to address these issues. When put together, the individual programmes add up to a tremendous effort in both time and money, and will ultimately lead to a much better understanding of materials and component behaviour in a wide range of postulated scenarios. It is to be noted that all countries involved have extensive modelling and code development programmes to best utilise the data generated from the experimental programmes.

It is therefore very important that these activities be well supported and that their results be made available to the widest possible audience. This is vital to support a common global culture of safe and economic production of electricity from nuclear power generation.

4.7 Future activities

4.7.1 The IFPE database

Now that the IFPE database is firmly established, there is a continuing need for it to be supported and improved. Indeed, on the assumption that the database will continue to be maintained by the NEA, several organisations are using the database as the sole data source for their code development and validation. Further data will be added to the database in the future, including the following for which agreement for release has already been obtained:

- BR-3 High Burn-up Fuel Rod Hot Cell Program (DOE/ET 34073-1, Vols. 1 and 2).

- Risø-I experiment.

- IFE/OECD/HRP FUMEX 1-6.

- Studsvik/SKI data from TRANS-RAMP I, II and IV.

- Zaporozhye VVER-1000 fuel behaviour data (4-8 cycles, burn-up >50 MWd/kgU).

- HRP He/Ar/Xe gas flow, Nb-doped fuel, IFA-504.

- VNIINM ramp data from VVER-440 (up to four cycles) and VVER-1000 (up to three cycles).

- IFA-508 and IFA-515 conducted by JAERI at HRP – PCMI behaviour data on different cladding thickness by means of diameter rig.

- CEA failed PWR rods irradiated in SILOE; EDITH-3 and EDITH-MOX 02.

- IMC (UK) swelling data from ramping CAGR UO_2 fuel.

Based on the success of its FUMEX-1 Code Comparison Exercise, the IAEA has agreed to launch a follow-up project, FUMEX-2. The overall outcome of the FUMEX-1 programme was very encouraging, with a high degree of participation from member countries. It is generally agreed that it was a worthwhile exercise, and that the cases chosen were stringent tests of the performance of models and codes. The exercise was useful in demonstrating the strong points of the codes, as well as highlighting deficiencies where improvements were necessary. As a consequence, most of the codes underwent some development during the exercise. It was also apparent that many of the codes had been developed using limited databases, and the FUMEX-1 cases provided a valuable addition.

However, the exercise showed that limited knowledge had been gained in the extended burn-up range (above 50 MWd/kgU). Burn-up extension is a general trend in the field of fuel management, and reliable prediction of fuel behaviour at high burn-up constitutes a basic demand for safe and economic operation of nuclear power plants. This was the basis for launching FUMEX-2, which was due to run from December 2002 until 2006. Central to the project are selected cases in the IFPE database. This will help promote the use of the database and, at the same time, will generate feedback and peer review from users, thus assisting the quality assurance of the data sets within the database. The cases chosen for FUMEX-2 are given in Table 4.2, with cases already in the database shown in bold type.

It is therefore important that the NEA provide additional support to maintain and improve the database over this period, in addition to the introduction of further data as they become available.

4.7.2 Other activities

Based on the survey of member countries and their safety-related R&D activities, the NEA has developed a good understanding of the priorities for current and future work. It is active in supporting these activities, particularly those related to high burn-up studies, the introduction of MOX fuel into commercial reactors, and the issues arising from the use of weapons-grade MOX (W-MOX) fuel in order to reduce the inventories held by both the USA and Russia. Two workshops were held on *Advanced Reactors with Innovative Fuels* (ARWIF) in 1999 and 2001. Fuel behaviour issues were discussed during these meetings [6,7].

The commercial nuclear industry is currently at a very critical stage. Concern over global warming has focused attention on fossil fuel usage and its emissions. As a result, many countries are extending the life of their currently-installed nuclear plants, as well as showing a renewed interest in building new nuclear generators. However, such new plants are unlikely to use current designs.

There are several new designs available. Some are extensions of the current Generation II LWR, but some are more innovative and as such are classified under the generic term Generation IV designs. Common features of these new designs include passive safety, high thermodynamic efficiency and proliferation-resistant fuel. Before entering full-scale commercial operation, they will require extensive qualification of their designs and of the materials used in their construction and operation. It is a goal for the NEA to assist in the quest for improved nuclear design and installation by supporting R&D into these new concepts and the associated new fuel types and cladding alloys.

Table 4.2. Cases from IFPE database chosen for FUMEX-2

No.	Case identification	Measurements made for comparison
1.	Halden IFA 534.14, rod 18	EOL FGR and pressure, grain size 22 µm, ≈ 52 MWd/kgU
2.	Halden IFA 534.14, rod 19	EOL FGR and pressure, grain size 8.5 µm, ≈ 52 MWd/kgU
3.	Halden IFA 597.3, rod 7	Cladding elongation at ≈ 60 MWd/kgU
4.	Halden IFA 597.3, rod 8	FCT, FGR at ≈ 60 MWd/kgU
5.	Halden IFA 507, TF3	Transient temperature during power increase
6.	Halden IFA 507, TF5	Transient temperature during power increase
7.	GONCOR	FGR and cladding diameter during and after transient at ≈ 48 MWd/kgU
8.	**Kola-3, rod 7 from FA222**	**FGR, pressure and creepdown at ≈ 55 MWd/kgU**
9.	**Kola-3, rod 52 from FA222**	**FGR, pressure and creepdown at ≈ 46 MWd/kgU**
10.	**Kola-3, rod 86 from FA222**	**FGR, pressure and creepdown at ≈ 44 MWd/kgU**
11.	**Kola-3, rod 120 from FA222**	**FGR, pressure and creepdown at ≈ 50 MWd/kgU**
12.	**Risø -3 AN2**	**Radial distribution of fission products and FGR-EOL, ≈ 37 MWd/kgU**
13.	**Risø -3 AN3**	**FGR and pressure-EOL, FCT, ≈ 37 MWd/kgU**
14.	**Risø -3 AN4**	**FGR and pressure-EOL, FCT, ≈ 37 MWd/kgU**
15.	**HBEP, rod BK363**	**FGR-EOL, ≈ 67 MWd/kgU**
16.	**HBEP, rod BK365**	**Fission products and PU distribution, FGR-EOL, ≈ 69 MWd/kgU**
17.	**HBEP, rod BK370**	**Fission products and Pu distribution, FGR-EOL, ≈ 51 MWd/kgU**
18.	**TRIBULATION, rod BN1/3**	**Pressure, FGR, cladding creepdown, ≈ 52 MWd/kgU**
19.	**TRIBULATION, rod BN1/4**	**Pressure, FGR, cladding creepdown, ≈ 51 MWd/kgU**
20.	**TRIBULATION, rod BN3/15**	**Pressure, FGR, cladding creepdown, ≈ 51 MWd/kgU**
21.	EDF/CEA/FRA, rod H09	Fission products and Pu distribution, FGR-EOL, ≈ 46 MWd/kgU
22.	Kola-3 + MIR test	Temperature during ramp, FGR-EOL, ≈ 55 MWd/kgU
23.	Kola-3 + MIR test	Pressure-EOL, ≈ 55 MWd/kgU
24.	RIA	To be specified (real data or simplified case)
25.	LOCA	To be specified (real data or simplified case)
26.	Simplified case	Temperature vs. burn-up for onset of FGR

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Appendix 3

NSC EXPERT GROUP ON SCIENTIFIC ISSUES OF FUEL BEHAVIOUR (TFSFB) – SCOPE AND OBJECTIVES

Scope

- The Expert Group deals with the status and trends of scientific issues of fuel behaviour.

Objectives

- Compile high quality experiments for the International Fuel Performance Experiments (IFPE) database. Emphasis is given to high burn-up in water reactors under normal operating conditions. Priority goes to completed international programmes, the data of which would otherwise be lost, but also to data released from national programmes.

- Co-ordinate the qualification of these data through review and by organising user-group meetings. Take initiatives so that it may be adopted as an international standard.

- Co-ordinate computer code validation and benchmark studies.

Work plan

- Organise co-ordination meetings combined with IFPE database user group meetings every 12-18 months.

- The IFPE database will be consolidated during the next two years through updating and addition of new experiments.

- Seminars and Experts' Meetings will be held as needed to address high-priority issues for validating the modelling of phenomena of particular concern (thermal performance, fission gas behaviour, pellet-clad mechanical interaction, etc.).

Co-operation

- This activity will be carried out in co-ordination with the IAEA, the NEA CSNI Special Expert Group on Fuel Safety Margins and the OECD Halden Reactor Project.

Chapter 5

NUCLEAR CRITICALITY SAFETY

5.1 Criticality codes for spent LWR fuel transport containers

At a meeting of the NEA Committee on the Safety of Nuclear Installations (CSNI) in 1979, it was decided that the committee should sponsor a working group to examine the ability to determine the criticality safety of large heavily-shielded containers designed to transport highly-irradiated nuclear fuel, particularly spent fuel. The Oak Ridge National Laboratory in the United States was asked to prepare a series of problems that would comprise a "Standard Problem Exercise on Criticality Codes for Spent LWR Fuel Transport Containers". The basic concept was that results from different computational techniques should be compared with actual experimental data, as well as among themselves, for specific test problems. Each participant in the exercise would be expected to provide solutions to the problems so that comparisons could be made to determine whether a consensus solution could be obtained.

In May 1980 the Criticality Computations Working Group was convened at the NEA in Paris for the purpose of discussing and studying the problem set that had been prepared at Oak Ridge. After reviewing and modifying the set, it was agreed that each country would provide solutions to the various problems using the computational techniques and cross-section data that they normally used for this type of criticality safety review.

The set of problems chosen was intended to provide a step-by-step procedure for establishing the validity of computational methods in determining the k_{eff} of a LWR spent fuel shipping container which uses a boron-aluminium sheet, Boral, as a neutron poison, is moderated by light water, and uses steel or lead as the biological shield around the outside of the container. The first three groups of problems in the set were critical systems containing, respectively, the fuel alone, the boron-aluminium sheet and the thick lead or steel shielding; in each case these were systems for which experimental critical data were available. Two further groups of problems were intended to combine all of these materials into a simulated mock-up of a spent fuel container.

The logic used in choosing these problems and the corresponding experimental data was to establish the validity of the computational methods in a stepwise fashion, a new parameter being introduced with each new problem. In this way the effect of the new parameter on the validity of the method could be observed. This was intended to prevent the masking of computational errors (mainly from inadequate cross-section data) by a combination of negative and positive bias in the results of the full cask mock-up, which could lead to unwarranted confidence in the results. This stepwise approach has been adopted by the working group for all subsequent studies and is considered to be one of the most important components in assuring the validity of the studies.

As might be expected from such an initial study, the early results showed a great diversity in the accuracy and quality of the results. Not all the participants had experience with this type of problem and some had chosen computational techniques and cross-section data not appropriate for this type of system. For all those involved this exercise provided a great opportunity to learn. It was shown just how easy it is to make input data errors and to misinterpret experimental data specifications. This gave great impetus to a project discussed in Section 5.5 below, the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*.

The report documenting this work, CSNI Report No. 71 of May 1982, provides a roadmap for performing spent fuel container criticality safety calculations. More importantly, it was the start of what has been a very successful series of criticality safety studies by the NEA.

5.2 Criticality codes for large arrays of packages of fissile materials

As the Criticality Computations Working Group neared the completion of its study on spent fuel shipping containers, the impending completion and expected results of this study were noted by an IAEA group which was reviewing criticality safety aspects of the regulations for the safe transport of radioactive materials, particularly fissile materials. In a letter to the NEA, the IAEA requested the working group to consider extending its study to cover calculations of arrays of transport packages, particularly those packages defined as Fissile Class II.

The problem as posed consisted of several parts. Packages which meet the Fissile Class II category must be able to pass a variety of tests involving accident conditions, many of which affect the criticality safety of the package. The size of the allowable arrays of such packages can be quite large, and the possible co-mingling of packages containing different types of packaging and fissile material must be considered. Although on the surface this seemed to be an easier task than the shipping cask exercise, it was soon realised that the working group had undertaken a daunting task.

As with the earlier task, the Fissile Class II package work was undertaken in a stepwise fashion in order to validate each step of the process by comparing calculations with experimental data where possible. A variety of materials was examined. The fissile materials considered were high-enriched uranium metal, high-enriched uranyl nitrate, 5% enriched uranium oxide, 5% enriched uranium oxide with H/U = 20 and plutonium oxide.

For this work it was necessary to study the criticality safety of large arrays, since typically Fissile Class II packages contain relatively small amounts of fissile material. It was learned that very few experiments had been performed with very large arrays of fissile material. The largest experimental critical system was a $5 \times 5 \times 5$ three-dimensional array of fissile material. Results for a few larger two-dimensional arrays were also available but were generally not benchmark-quality experiments. Hence, a major portion of this study revolved around evaluating the ability to compute k_{eff} for systems with a large number of fissile units. One significant problem that can occur with large arrays is that of under-sampling in Monte Carlo calculations (the only computational tool available to rigorously compute the k_{eff} of most finite arrays). (As will be discussed below, a project recently undertaken by an NEA expert group looks at “source convergence”, which addresses this problem.)

Another interesting issue that was explored was the co-mingling of fissile packages from different shipments. This raised a complex and perhaps unanswerable question – for two different types of package, where an $N \times N \times N$ array of each would be sub-critical, would an $N \times N \times N$ array of the two types co-mingled also necessarily be sub-critical? For the problems examined this appeared to be true. However, one participant believed this to be an unresolved issue. At present no study group is examining this issue.

While the study allowed a recommendation to be made regarding this type of package, and the results were published as CSNI Report No. 78, it was clear that additional experiments would be needed to satisfactorily provide the complete guidance required to assure safe and economic transport of fissile material in these types of packages. The report provides a list of problems that need to be resolved before proper assessment of the criticality safety of this type of system can be assured. At present, however, there is not sufficient economic incentive to improve the data.

5.3 Criticality codes for dissolving fissile oxides in acids

This third effort by the Criticality Calculations Working Group involved the evaluation of criticality safety for the dissolution of fissile oxides in acids. While this work obviously can be applied in a variety of situations, the working group did not specifically address uses to which the participating NEA member countries might apply the data. The obvious applications are: transport accidents in which the cladding of the fissile material is breached; Three Mile Island-2 type reactor accidents; and the dissolution of fuel elements in acid. The unique nature of this exercise compared to the two previous studies was that it involved fissile material in two forms – namely, fissile material in solid form surrounded by fissile material in solution.

(Because of the nature of this undertaking and for other internal reasons at the NEA, the oversight of the Criticality Calculations Working Group moved from the CSNI to the Committee on Reactor Physics at the start of this study.)

For the first two studies, the experimental data, while not complete, were relatively rich as compared to those available for this study. In addition, for some of the data that were available, which were the most directly applicable, there were problems with the descriptions of the solutions. After discussions with the experimentalists, the working group was able to obtain better descriptions of the materials in the solutions. However, doubts will continue to be present regarding these data since the materials used in the experiments no longer exist in their original form. Again, these problems pointed out the need for evaluated benchmark data and additional experiments.

A series of some 18 experimental systems was chosen by the working group to validate the computational systems being used by the various participants. The first set of experiments involved a variety of fissile materials in the form of rods immersed in a variety of non-fissile solutions. For most of the experimental systems studied, the solution contained some form of dissolved neutron poison, since one would expect to add a neutron poison to the acid to reduce neutron multiplication. The second set of experiments involved the data which, while most applicable because they involved fissile material in solution, were nevertheless clouded by experimental uncertainty as noted above. The last set of experimental data examined involved fissile material surrounded by a solution in which hafnium plates had been inserted.

As with the earlier studies performed by the working group, the experiments were designed to serve as building blocks in the validation process. With each experiment examined and each calculation validated, confidence was increased in the ability of a method to provide accurate results. The results of the first part of this study were documented in report NEACRP-L-306 of April 1990.

The next step in this process was to define two hypothetical fuel-dissolving systems to be studied. The purpose of this was to determine if computational methods which had been validated for the 18 experimental systems would produce similar results for these unknown systems, which involved all of the complexities that had been examined individually in the validation process.

While not all of the methods gave the same results, the working group came to a very good understanding of the elements that went into producing useful criticality safety results for systems involving the dissolution of fissile materials in acid. The two reports which describe the results of the studies of the two hypothetical fuel-dissolving systems are essential reading for anyone attempting to make criticality calculations for this type of system. These were issued as NEACRP-l-320 of December 1990 and NEACRP-L-325 of January 1991.

5.4 Expert Group on Burn-up Credit

As the work involving fuel dissolution was coming to an end, the Criticality Calculations Working Group was asked to consider whether a study might be made to provide a validated procedure to perform criticality safety calculations that would take into account the actual condition of the fuel element after it had been burnt in a nuclear reactor. Until this study began, all criticality safety calculations involving the handling, transport and storage of spent fuel elements had been made assuming that the fuel was as originally fabricated, with no allowance for burn-up or the build-up of fission products.

[It should be noted at this point that the oversight of the working group moved from the Committee on Reactor Physics to the Nuclear Science Committee (NSC).]

The impetus for this effort was clear. With the amount of spent fuel constantly increasing, and the likelihood that the enrichment level of fresh fuel would be increased, the advantages of taking into account the actual material content of the spent fuel became too great to ignore. Clearly, if the actual isotopic content of spent fuel could be verified, and computations could be validated, significant savings both in cost and storage space could be achieved.

One very important fact to note regarding this study: while reactor designers have very good, validated computational tools to determine the criticality of fuel as it is burned in the reactor, these tools are not generally useful in determining the criticality of the fuel during handling, transportation and storage. The reason for this is that in the reactor core every attempt is made to minimise leakage and absorption, while out of the reactor every effort is made to maximise these effects. Hence, the neutronic behaviour is greatly different out of the reactor, requiring computational tools which are able to deal with systems containing voids and strong absorbers. Also, nuclide poisons build up over time after the fuel is removed from the reactor; these are not present in the reactor computational models.

The initial part of this study involved 13 systems. The investigated parameters were burn-up, cooling time and the combination of nuclides in the fuel region. The fuel arrangement for the first 13 cases was a simple infinite lattice PWR pin cell. While the principal result of the calculations was the k_{eff} of the system, for the first time the working group also reported absorption reaction rates for the various elements in the system as a way of better understanding the results obtained. Using this approach it was able to determine which elements contributed most to the decrease in k_{eff} as a result of burn-up. However, when using the conclusions reached in this part of the study it must be kept in mind that leakage is missing and can have an effect on observed reactivity losses.

This first part of the burn-up credit study is referred to as Phase I-A, and was documented in NEA/NSC/DOC(93)22. While not actually part of the criticality calculations, a further study called Phase I-B was conducted which consisted of calculations to determine the isotopic concentrations at various burn-up and cooling times. This study was documented in NEA/NSC/DOC(96)06. These initial parts of the burn-up credit study provided the basis for future work and provided information to guide the examination of other parameters important to the effect of burn-up on criticality safety.

The next part of the burn-up credit study, Phase II-A, was initiated to extend the study to consider the effect of realistic axial profiles on the k_{eff} of a finite array of spent PWR fuel elements in a simulated shipping cask mock-up. Again the study considered a variety of parameters which included different burn-ups and cooling times after the fuel was removed from the reactor.

The most significant result of this part of the study was that for low burn-ups, using the axial burn-up profile gave smaller values of k_{eff} than the corresponding case with an averaged axial burn-up. On the other hand, for high burn-ups the axial profile gave larger values of k_{eff} than those calculated with an averaged axial burn-up. The crossover point at which the two approaches give the same answer was also demonstrated to be a function of the cooling time and initial enrichment.

The important conclusion was that it appears not to be possible to predict in advance whether the calculations of k_{eff} for an infinite array of spent fuel pins can be made with averaged axial isotopic densities. This work was documented in NEA/NSC/DOC(96)01. As with the first phase, a further study, Phase II-B, was conducted to determine the isotopic content of the fuel after various burn-ups and cooling times. This was documented in NEA/NSC/DOC(1998)1.

These early studies followed the step-by-step approach that had already been established by the working group in its study of difficult criticality safety problems. Using this approach, the Expert Group on Burn-up Credit has continued its work for additional reactor systems. The following NEA documents report a portion of its work to date:

- *OECD/NEA Burn-up Credit Criticality Benchmarks Phase III-A: Criticality Calculations of BWR Spent Fuel Assemblies in Storage and Transport* [NEA/NSC/DOC(2000)12].

- *OECD/NEA Burn-up Credit Criticality Benchmarks Phase III-B: Burn-up Calculations of BWR Fuel Assemblies in Storage and Transport* [NEA/NSC/DOC(2002)2].

- *Burn-up Credit Criticality Benchmark Results Phases IV-A and IV-B: Analysis of MOX Fuels* (in preparation).

A number of additional publications will complete the documentation of this work. To date the work has covered a wide range of fuel types, including PWR, BWR, MOX and VVER fuels.

This expert group has existed longer than any other NEA group, and the results produced demonstrate the worth of the very extensive study it has undertaken. The group is progressing well with its task of demonstrating which criticality safety computational tools are appropriate for determining the criticality safety of spent fuel systems and that a reasonable margin of safety can be established.

5.5 International Criticality Safety Benchmark Evaluation Project

In 1992, the US Department of Energy initiated an effort to collect data from nuclear criticality experiments which had been performed at various facilities around the world. The goal was to study the data and determine which of the experiments had been sufficiently documented to qualify as benchmark experiments; that is, those experiments which could be qualified to be used in the validation of computational methods.

By 1994 this had grown into an international effort and it became clear that the project needed international sponsorship. While many countries had been willing to share their data with the US effort, an international organisation’s sponsorship would provide additional incentives for the sharing of data. It was also obvious that the Criticality Calculations Working Group at the NEA had a need for such benchmarks. Consequently, it was agreed that the NEA would sponsor this work, which is known as the International Criticality Safety Benchmark Evaluation Project (ICSBEP).

This project has grown rapidly under NEA sponsorship and it continues to be an important part of the international effort to improve criticality safety calculations. The data are distributed on CD-ROM or on-line as the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*. The CD-ROM distributed in 2001 contains 307 experimental series with nearly 2 900 critical configurations, and new data are continuously being added.

This effort may well have made the greatest single contribution to nuclear criticality safety by providing comprehensive and accurate data for use in validating criticality safety calculations. It is a continuing effort that has benefited greatly from sponsorship by the NEA.

5.6 Working Party on Nuclear Criticality Safety

In 1996, work on criticality safety had grown to the extent that the NSC established the Working Party on Nuclear Criticality Safety to oversee and review the activities of the existing Expert Group on Burn-up Credit and the ICSBEP. This working party was also assigned the responsibility of establishing new expert groups to address issues that had been raised by the two established groups. A large number of new topics was considered by the working party, and the following expert groups were subsequently convened:

- Minimum critical values.
- Experimental needs.
- Source convergence analysis.
- Criticality excursion analyses.
- Sub-critical measurements.

5.6.1 Expert Group on Minimum Critical Values

This topic resulted from discussions regarding discrepancies in the minimum critical values (e.g. minimum spherical radius, mass and concentration for specific fissile materials) that are listed in criticality safety handbooks issued by various countries. These data are important in establishing the maximum amount of any given fissile material that can be safely handled without criticality safety concerns.

At this point the reason for the discrepancies in published data has not yet been determined. Each participating organisation with published results will be providing further information about how its values were determined. In addition to this work, the group has defined a set of seven materials and material configurations to be studied by each participant in an effort to define the minimum critical value for each of these configurations.

5.6.2 Expert Group on Experimental Needs

The objective of this group is to identify experimental needs of mutual value to NEA member states, to encourage joint participation in the planning and performance of such experiments, and to provide guidance on quantitative means to evaluate and justify the experimental needs.

In a number of activities by NEA working groups in the past, progress has been hampered by the lack of adequate and/or appropriate critical experiments. One of the major activities of this expert group has been to define the parameter space for which there is a lack of experimental data. One innovative approach has been to use a web-based system using the data found in the ICSBEP database. This method, developed at the NEA and named DICE, searches the existing database to determine if particular characteristics can be found in one or more existing experiments. Another proposed method involves an analysis using the sensitivity/uncertainty approach to evaluate an experimental need.

A topic under consideration involves the need for critical experimental data for damp MOX powders. There are almost no applicable data for such systems, while the need for data is developing as the industry moves forward in the use of MOX fuels.

A major handicap to obtaining any new experimental data will be the limited number of critical experiment facilities still operating, along with the need to obtain funding for the experiments. Identification of the experimental data needed, along with proper documentation of the need, by this expert group should aid greatly in convincing the proper authorities to fund these experiments.

5.6.3 Expert Group on Source Convergence Analysis

The origin of this work was a paper presented in 1971 at an American Nuclear Society meeting entitled "A Difficulty in Computing the k_{eff} of the World". This paper pointed out the real possibility that inadequate sampling in a Monte Carlo criticality calculation would likely result in a value of k_{eff} less than the real value. This problem, sometimes called the Whitesides Problem, has been addressed in a number of forums, but remains one of the outstanding unresolved problems in assessing the criticality safety of complex, heterogeneous systems of fissile materials for which the only method of computing the neutron multiplication is with the Monte Carlo method.

A series of benchmark systems has been proposed, each of which is likely to exhibit source convergence problems. These are being studied by a number of participants using the computational methods which they commonly use. Each participant will apply a variety of techniques in order to try to assure proper source convergence.

The goal of this expert group is two-fold. First, to increase awareness of the problem of possible inadequate sampling. Second, to make every effort to find techniques to minimise or, at least, detect the occurrence of this problem. A variety of techniques have being proposed for implementation in Monte Carlo criticality computational methods. Each of these will be tested in order to evaluate how effectively it either solves the source convergence problem or alerts the user that such a problem exists.

5.6.4 Expert Group on Criticality Excursion Analyses

This expert group met for the first time on 4 December 2001, and its work has not yet been fully defined. The first significant action of the group was to change its name from Expert Group on Criticality Accidents to Expert Group on Criticality Excursion Analyses in order to better describe the scope of the proposed work. The group's task is to provide scientific knowledge on criticality excursions which can be used by the nuclear criticality community in the study and evaluation of criticality accident analysis.

The initial goal of the group is to define the fissile materials to be considered for benchmarks, along with defining the priorities for the different materials. The next step will be to determine the availability of experimental data and the needs for benchmarking evaluation of the data. From this effort a series of benchmark exercises will be developed for the purpose of determining how well the current computational tools compare with experimental data, and ultimately how effectively they can evaluate potential criticality excursions.

From these studies the group expects to identify computational methods that can best model criticality accidents, and to compile a list of experimental needs for criticality accident analyses. A database of existing experimental criticality transient data from experimental programmes and from accident data will also be compiled.

5.6.5 Expert Group on Sub-critical Measurements

While most of the nuclear criticality safety work has relied on critical experiments to determine criticality safety, there are a number of activities designed to measure and determine the multiplication factor for sub-critical systems. The goal of this relatively new activity is to evaluate and review sub-critical measurements for computer code and nuclear data evaluation.

This expert group plans to provide a set of evaluated sub-critical benchmark experiments for inclusion in the data being gathered for the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*. The first such experiment has been prepared and submitted for inclusion in this collection. A further experiment involving PWR fuel was performed at the Dimple facility in the UK and is undergoing evaluation as a potential benchmark experiment.

A further part of this study will be to produce an assessment of the value of sub-critical measurements for criticality safety analysis.

5.7 Other possible NSC activities

The Japan Atomic Energy Research Institute (JAERI) has proposed the transfer of the Spent Fuel Isotopic Composition Database to the NEA. This effort would be handled in the same manner as the International Criticality Safety Benchmark Evaluation Project, in that it would be a repository for isotopic data for various types of spent fuel. These data would be used to validate computational methods for calculating the isotopic content of spent fuel for use in criticality safety calculations.

Chapter 6

PLUTONIUM RECYCLING AND INNOVATIVE FUEL CYCLES

6.1 Introduction

The NEA Nuclear Science Committee (NSC) has been addressing issues related to plutonium fuel for many years, initially for fast breeder reactors (FBRs) and later for mixed-oxide (MOX) fuel in LWRs and fast burner reactors. Much work has been conducted on FBRs, including an international comparison calculation for large sodium-cooled FBRs. For high-conversion LWRs (HCLWRs), an international benchmark of tight lattice cell burn-up calculations was carried out to validate methods and data.

These benchmark calculations for FBRs and HCLWRs led to improved and newly-developed nuclear codes, resulting in data which is of a quality comparable to that from standard LWR benchmark calculations. Especially for HCLWR application, computational codes based on the ultra-fine group method were developed which accurately calculate resonance shielding, including the intermediate neutron spectrum region. Continuous-energy Monte Carlo calculations were carried out as a reference solution. This methodology has subsequently been used for standard calculations in various benchmark exercises.

In June 1992 the NSC set up the Working Party on Physics of Plutonium Recycling. In 1997 this was followed by the Working Party on Physics of Plutonium Recycling and Innovative Fuel Cycles (WPPR). Benchmark studies undertaken by this working party cover plutonium recycling in PWRs, void reactivity effect in PWRs, fast plutonium-burner reactors, plutonium recycling in fast reactors, multiple plutonium recycling in standard and advanced PWRs, and MOX fuel in BWRs. Benchmark calculations for MOX fuel lattice experiments in LWRs have also been carried out.

6.2 Plutonium recycling in PWRs

Three benchmark exercises have been carried out by the WPPR for MOX fuel in PWRs. The first two were simple infinite pin cell problems. These investigated particular issues related to MOX fuel utilisation in PWRs with plutonium of both typical and poor isotopic quality. The typical case had a plutonium isotopic vector with a higher fissile fraction, as used in commercial PWR MOX fuel. The poor-quality case had plutonium of low isotopic vector, expected to become available for self-generation recycling in PWR MOX fuel. The benchmarks were designed to investigate whether present nuclear data and lattice codes required further development and validation to calculate the plutonium fuel core physics performance.

When the WPPR proposed these benchmark exercises, it was expected that good agreement between the various solutions would be obtained. This expectation was not completely fulfilled, even though the spreads were smaller than those seen in an earlier series of benchmarks. In both benchmarks, the spread in k_∞ was 3-5%. This included the results from established commercial codes verified for uranium fuels. If these were not included the spread decreased, but was not less than 1%. Furthermore, this spread in k_∞ translated into a much larger spread in the plutonium content required to achieve the desired reactivity lifetime. This showed that there was still a need for improvement in both the methods and the basic data for higher plutonium isotopes and minor actinides. The computational methods have to take into account resonance self-shielding and mutual shielding, especially self-shielding of ^{242}Pu.

The third benchmark covered the void reactivity effect in PWRs. It specified a more complicated geometry, corresponding to a supercell configuration of a 30×30 array of PWR uranium and MOX fuel cells which was tested in the VENUS experimental reactor at Mol, Belgium. The calculated results showed that infinite lattice calculations with uranium and differently enriched MOX fuels give a non-negligible spread of results, but with a clear tendency to display the void effect, which is positive for a high content of MOX in the fuel and negative for high plutonium density. A similar tendency was shown in the results of two-dimensional calculations for a uranium assembly with a central moderated or voided region fuelled with MOX fuel pins of different plutonium content.

As a final point, it was noted that the design of MOX fuel assemblies required the use of the newest nuclear databases and the application of very detailed and sophisticated spectra and assembly codes. The use of codes only verified for uranium fuel should be avoided. In connection with the void reactivity effect benchmark, more realistic conditions should be considered, such as an extension of the voided region beyond the limits of the MOX fuel sub-assembly. Furthermore, experimental validation would also be needed, in particular for integral parameters such as reactivity coefficients, in the case of plutonium with degraded isotopic composition. In recycling plutonium in PWRs, the important physics issues considered should be, in particular, the plutonium content limitation to avoid positive void effects and to minimise minor actinide production.

These benchmark exercises proved to be a valuable means of making progress in the interim period before practical results started to become available from in-reactor irradiation experience. The conclusions from the benchmark studies can be summarised as follows:

- The nuclear computational methods have to take into account rigorous resonance self-shielding and mutual shielding over the whole energy region for the fuel and cladding nuclides, and appropriate and well-tested calculation methods should be used.

- Sufficient quality basic nuclear data is needed for ^{238}U and the plutonium isotopes, and also for higher actinides and the major fission products. Modern nuclear data libraries such as ENDF/B-VI, JEF-2 and JENDL-3 are essential.

- Experimental verification of maximum plutonium content in the case of plutonium with degraded isotopic composition is needed in clean lattice configurations with different moderator-to-fuel ratios.

6.3 Plutonium recycling in fast reactors

Topics for benchmarks examined for fast reactor systems involved the physics of plutonium recycling in both oxide- and metal-fuelled fast reactors to determine the potential of fast reactors for consuming plutonium and fissioning minor actinides. Previously, burner reactors had not received the same level of attention as breeder reactors. In these benchmarks the physics related to, in particular, consuming plutonium, rather than establishing a breeding cycle, and reducing the source of potential radiotoxicity was investigated. Such fast reactor systems may have an important role in managing plutonium stocks prior to the introduction of fast breeder reactors.

The benchmark excises considered ranged from standard fast breeder reactors to cases with non-breeding fast systems with conversion ratios in the range 0.5 to 1.0. The symbiosis between fast burner reactors and LWRs was investigated. In the initial decades, fast reactors could be employed as burners in order to consume excess transuranics. At a later stage, with excess plutonium and ^{235}U reserves exhausted, fast reactor loadings could be converted to fissile self-sufficient mode or breeder mode. Results indicated a potential for significant nuclear waste toxicity reduction as well as the benefit of additional energy from multi-recycled LWR spent fuel and reprocessed plutonium.

In the metal-fuelled benchmarks, three cases were specified: the beginning of life (BOL) core, the core designed for a once-through fuel cycle and the multiple recycling core. Concerning physics issues, the BOL benchmarks displayed a larger spread of results among the participants than was the case with more conventional breeder designs. High leakage cores and higher contents of minor plutonium and higher actinide isotopes all need further validation work, including the performance of critical experiments.

6.4 Plutonium fuel without uranium

In recent years, with ex-weapons plutonium becoming available for destruction/degradation and with increasing stocks of civilian plutonium, plutonium burning in reactors has been emphasised as a means of minimising the risks of proliferation. From this perspective, the most effective method is to use plutonium without uranium. By using inert carriers for plutonium it is possible to avoid the production of fresh ^{239}Pu from ^{238}U neutron captures. However, work on non-uranic plutonium fuels has been very limited compared with R&D on MOX fuels. A specific task group has been established by the Paul Scherrer Institute (PSI) of Switzerland, the Japan Atomic Energy Research Institute (JAERI), the CEA of France, and the Politecnico Milano of Italy to summarise the present status and major issues associated with R&D of plutonium fuels with non-uranic carriers, concentrating in particular on the physics.

The most suitable inert matrix depends on the reactor type, the reprocessing method, whether once-through or recycle mode is used, etc. It is necessary to select the proper inert matrix for the system, taking into account the nuclear characteristics, physical properties, irradiation performance and reprocessing capabilities. It is recommended that manufacturing feasibility and irradiation tests for plutonium fuels should be given priority.

Some benchmark exercises have been carried out. In the case of a partially-loaded inert matrix fuel (IMF) core in a PWR, the solutions provided showed good agreement for uranium fuel, but discrepancies were larger for MOX fuel. Further problems were encountered with the kinetic parameters in which the discrepancies are observed. An Inert Matrix Fuel Workshop has been held every year since 1995, and R&D is underway to find potential fuel candidates.

6.5 Recycling of plutonium in advanced converter reactors

A review was carried out of advanced converter reactors and their relevance to plutonium recycling. These are thermal reactors in which the conversion ratio is significantly increased compared with conventional thermal reactors. They therefore could have a role as an intermediate step between present thermal reactors and future fast breeder systems.

Several reactor types were reviewed, including the Japanese advanced thermal reactor (ATR), a variant of the CANDU, and HCLWRs. In some lattice experiments with HCLWRs, such as the LWHCR at the PSI, ERASME/S (moderator ratio 0.5) and ERASME/R (moderator ratio 0.9), comparison of measured and calculated lattice parameters appeared to show no limitations imposed by a lack of knowledge of the neutronics and physics of such reactors.

6.6 Multiple plutonium recycling in PWRs

A shortcoming of the first plutonium recycling benchmarks was that the cases considered corresponded only to the present situation (with plutonium of good isotopic quality), and to the scenario which might arise after many generations of MOX fuel recycling (with extremely poor isotopic quality). Analysis of the intermediate steps was missed, and therefore there was no possibility of determining precisely where the nuclear codes and data libraries would start to lose their applicability. Accordingly, the CEA suggested a benchmark in which five consecutive generations of multiple recycling in a PWR would be followed. In the specification of the benchmark, attempts were made to make it as realistic a scenario as possible, taking into account such details as the length of time between recycle generations (accounting for the time delays in pond storage, MOX fuel fabrication, etc.) and the dilution effect when MOX and UO_2 assemblies were co-reprocessed.

As in the previous benchmark exercise, the benchmark was restricted to the level of the lattice codes. This was the logical first step because it did not make sense to progress to the three-dimensional whole core codes until the underlying nuclear data and lattice code calculations showed adequate agreement. Two cases were considered: one for a standard 17×17 PWR lattice such as used in many present PWRs (designated the STD-PWR); and one for a lattice with an increased moderator/fuel ratio (3.5:1 compared with 2:1 for the STD-PWR). The latter case, designated the HM-PWR (for highly-moderated), was intended to cover a proposed PWR design for use with MOX fuel only. In such a PWR it would be possible to optimise the lattice to give increased moderation, thereby improving the efficiency with which the plutonium could be used.

Parameters examined in the benchmark exercise included end-of-cycle reactivity (which determines cycle length), the variation of reactivity with burn-up, reactivity coefficients, microscopic cross-sections, isotopic evolution and isotopic toxicity evolution with time. The broad conclusions and observations made from this comparison are discussed below.

Since the earlier benchmarks considerable progress had been made in nuclear data libraries and methods. The discrepancies between the different data libraries and lattice methods, when applied to multiple recycle scenarios in PWRs, were generally within reasonable bounds. The observed spread of results was consistent with the uncertainties in the underlying nuclear data, and would require further experimental validation prior to practical implementation of multiple recycling. Multiple recycle scenarios therefore appeared to be practicable and feasible in conventional PWRs, at least in the near term. However, questions arising from a possible positive void coefficient would almost certainly preclude recycling beyond the second generation.

In the longer term, the HM-PWR concept had some merit. However, the principal advantage of needing a lower initial plutonium content was largely eroded in later recycle generations. The HM-PWR degraded the plutonium isotopic quality more rapidly than the STD-PWR, and this negated the benefit of the softer spectrum. The HM-PWR also seemed to pose more difficulties for present nuclear data libraries and codes, as evidenced by the larger number of discrepant results seen in the HM-PWR benchmark. Therefore, the HM-PWR was also of questionable practicability with respect to later recycle generations.

In view of these considerations, the WPPR agreed that there was no compelling reason to continue further benchmark studies at the level of the lattice codes.

6.7 Power distribution within MOX fuel assemblies

The main objective of this benchmark was to compare different techniques of fine flux assessment derived from coarse mesh diffusion calculations or transport calculations. Ten institutions contributed and more than 15 calculation schemes were examined, which included the majority of the methods used for reactor design: collision probability, S_n transport (finite difference and nodal), diffusion (finite difference and nodal), Monte Carlo, power reconstruction methods, etc.

From a mathematical perspective, excellent consistency of results was noted for the reference results using fine mesh diffusion calculations, and also for the transport calculations. With regard to the nodal calculations, and more specifically fine flux reconstructions, it was observed that although different nodal methods were in use in various institutions (analytic solutions or nodal expansion methods) these methods were capable of reconstructing the fine flux in the assemblies within a few per cent. Larger discrepancies appeared at U-MOX interfaces, and it was pointed out that discrepancies might also appear in non-symmetric configurations. The conclusion was that fine flux reconstruction could be achieved to a satisfactory precision, except for local singularities.

A second phase benchmark exercise was undertaken for power distribution within MOX-fuelled assemblies. Concerning core calculations, a large reactivity spread of 860 pcm from the average value of k_{eff} was observed. This was due to cross-section libraries and methods used. As for the pin-by-pin power distributions, on average the participants overestimated the power in the central MOX fuel assembly, with a spread of about 3% for each zone.

6.8 BWR MOX fuel benchmark

The previous work of the WPPR addressed fast reactors and PWRs but did not consider BWRs. As MOX use in BWRs was intended in several NEA member countries, the WPPR felt that it was important to address this imbalance in its activities. A specification for the BWR MOX fuel benchmark was provided by Siemens. The benchmark exercise compared calculations for a modern 10×10 BWR fuel design with assemblies containing MOX fuel rods and uranium/gadolinium fuel rods with a large internal water structure. The fissile content allowed for an average discharge burn-up above 50 GWd/t, and the isotopic composition of the plutonium corresponded to that from burnt uranium fuel with a discharge burn-up above 50 GWd/t.

The results from participants showed a similar accuracy to that achieved for PWR benchmarks. For the BOL core, agreement within 1% for k_∞ was observed, and a difference of slightly more than 1% was found when comparing ENDF/B-VI against JEF-2.2. For cases after some burn-up, the agreement was better than that for BOL cases.

6.9 Ongoing work

At the fifth WPPR meeting in November 1996, work to analyse MOX fuel critical experiments in LWRs was proposed as an essential means of validating calculations against experiments. Several experiments were put forward for consideration, such as TCA (JAERI, Japan), VENUS-2 (SCK•CEN, Belgium), EOLE (CEA Cadarache, France), etc. These data sets were not available immediately, and would require the agreement of the original sponsors to release part or all of the data.

6.9.1 Benchmark analysis of VENUS-2 data

SCK•CEN at Mol, Belgium, has carried out a benchmark analysis based on experimental data from the VENUS-2 programme. The core consisted of 3.3% and 4.0% enriched uranium fuel rods and

2.7% plutonium-content MOX fuel rods. Comparison of calculations and experiments was conducted for k_{eff} and pin-power distributions. The first benchmark was a two-dimensional configuration model, and the second phase was a three-dimensional model. Participants adopted the blind benchmark procedure, in which the experimental values are not given to the participants.

Thirteen solutions were submitted, with generally satisfactory agreement (discrepancies of <500 ppm for reactivity, 5% for power in the uranium regions and 10% for power in the MOX region). The higher spread of pin powers in the MOX region was partly the result of the MOX rods being positioned at the core periphery and therefore more influenced by boundary effects.

6.9.2 Proposed future benchmarking exercises

HTR benchmark

High-temperature reactors (HTRs) have recently attracted considerable interest. The IAEA has organised a benchmark study with a prism-type assembly. To avoid overlap it has been suggested that the WPPR address the issue of pebble-bed reactors. The pebble-bed modular reactor (PBMR) concept is more interesting for physics studies than the prismatic design. Proposals for a HTR plutonium physics benchmark have been presented by Delpech (CEA) and Rutten (FRAMATOME). Furthermore, Kasemeyer has presented an analysis of experiments carried out on the HTR at the PROTEUS facility. The experimental data are of interest, but the PROTEUS geometry is complicated and difficult to model. Thus it is a candidate for work in a second-phase study.

MOX core transient benchmark

A benchmark exercise entitled "Kinetic Parameters via Calculation/Experiment Comparison of Reactivity Effect in CROCUS" has been proposed. It is a three-dimensional problem and involves four water heights and two control rods, six cases in all. The first phase would be the uranium fuel benchmark, and a similar exercise on MOX fuel would follow as a further step. In fact, large discrepancies due to data libraries have been found at PSI (20% in some cases). This benchmark has been endorsed by the WPPR.

Co-operation with the Expert Group on Reactor-based Plutonium Disposition

Three benchmark exercises are being developed in co-operation with the Expert Group on Reactor-based Plutonium Disposition (TFRPD):

- A three-dimensional VENUS-2 MOX fuel blind benchmark, for which specifications have been prepared based on VENUS-2 MOX core measurements from May 2001.
- KRITZ-2 benchmarks, consisting of three low-enriched uranium fuel cores and one MOX fuel core.
- A VVER-1000 MOX fuel benchmark proposed by the Kurchatov Institute, Russia.

Furthermore, a study related to plutonium management in the medium term (i.e. beyond LWR MOX fuel, but prior to fast reactors) has been addressed.

Chapter 7

REACTOR PHYSICS AND SHIELDING

7.1 Radiation dose and radiation-induced degradation of reactor components

It is essential to accurately calculate neutron fluence and fluence rates at more than one location in a reactor in order to assess and predict pressure vessel embrittlement and thereby ensure the safe operation of nuclear power plants. The NEA Nuclear Science Committee (NSC) established the Task Force on Computing of Radiation Dose and Modelling of Radiation-induced Degradation of Reactor Components to review this problem. At an early stage of its work the task force reviewed the state of the art in calculation methodologies for neutron and gamma fluence in reactor vessels and published a report [NEA/NSC/DOC(96)5].

As a follow-up to this report, the task force launched two blind benchmark studies to validate and verify the claimed accuracy of the calculation methods used. Both benchmarks were based on the VENUS experiments performed at SCK•CEN in Mol, Belgium, which offered the exceptional advantage of exhibiting a realistic radial core shape and a neutron spectrum that is typical of PWRs. Details of the two benchmarks are:

- *VENUS-1*. A two-dimensional benchmark on ex-core dosimetry computations. Comparison for five measured reaction rates [^{58}Ni(n,p), ^{115}In(n,n′), ^{103}Rh(n,n′), ^{238}U(n,f), ^{237}Np(n,f)], fluxes and displacement per atom (DPA) at 34 points on the mid-plane. The given uncertainty of the measurements amounted to ±5%.

- *VENUS-3*. A three-dimensional benchmark on ex-core dosimetry computations. In a three-dimensional benchmark, a large number of detector positions is needed to describe the streaming of fast neutrons through the complex three-dimensional geometry of the outer core, core baffle, water reflector and core barrel. The number of detector types for comparison was decreased to three reaction rates [^{58}Ni(n,p), ^{115}In(n,n′), ^{27}Al(n,α)], fluxes and DPA at 344 positions. Apart from the mid-plane, the detectors were located at 14 different axial levels between 105 cm and 155 cm.

The benchmark results supplied for VENUS-1 were based on eight two-dimensional S_n-method codes (DORT), one in-house transport code and one Monte Carlo code. The results revealed that the main source of inaccuracy in the VENUS-1 calculations was due to the diffusion-theory-based buckling correction. This worked satisfactorily within the reactor core, but failed outside the core, resulting in an underestimation of reaction rates by about 10% at those detector positions. It was demonstrated that a relative difference between measurement and calculation of ±20% has to be taken into account. The main problem in the two-dimensional dosimetry calculations is determining the right buckling correction in a case where the core height is small, such as the VENUS core.

In choosing the right order of S_nP_l approximation, it would appear that a S_8P_3 calculation was sufficient for in-vessel dosimetry. For ex-vessel dosimetry, however, a higher order approximation was required.

Eight results were contributed for VENUS-3, comprising five three-dimensional S_n calculations, two Monte Carlo calculations and one flux synthesis method calculation. From the VENUS-3 benchmarks the following conclusions could be derived:

- Agreement between calculations and measurements for VENUS-3 was considerably superior to the VENUS-1 exercise. The main reason was that buckling approximation is not necessary in three-dimensional calculations.

- The results of the TORT, PENTRAN and MCNP codes were within the target accuracy of ±10% desired for dosimetry calculations. If necessary, more precise calculations would be possible with deviations of ±5% with regard to the measurements. This could be achieved by adopting finer spatial mesh size and multi-group structure in case of the S_n calculations.

- The calculated DPA rates exhibited a relatively large scatter band. This was because two versions of DPA cross-sections differing by 20% were applied in the S_n calculations. DPA rates calculated with the data from ASTM-82 were significantly underestimated compared with those derived from new iron data from ENDF/B-VI.

7.2 Benchmark on the VENUS-2 MOX core measurements

In the framework of joint activities carried out by the NSC Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles (WPPR) and the Task Force on Reactor-based Plutonium Disposition (TFRPD), a blind international benchmark exercise for the prediction of power distribution in the two-dimensional VENUS-2 MOX core experiment was launched in May 1999. The NSC had already been studying theoretical physics benchmarks and multiple recycling issues related to various MOX-fuelled systems. However, it was felt that there was also a need to link these findings to data from experiments. The aim of this exercise was to investigate the predictive capability of current production codes with the latest nuclear data libraries used for calculations on MOX-fuelled systems by comparing their results with experiments in the VENUS-2 MOX core.

The VENUS facility is a zero power critical reactor at SCK•CEN in Belgium. The core comprises twelve 15×15 sub-assemblies. In the VENUS-2 core, the central part of the core consisted of 3.3%-enriched fuel pins. For the eight assemblies on the periphery of the core, which contained 4.0%-enriched fuel pins, eight rows of the most external fuel pins were replaced by MOX fuel pins enriched 2.0% in ^{235}U and 2.7% in plutonium. One-eighth of the core comprised 325 fuel rods, of which the pin powers of 121 fuel rods were directly measured and those of 204 fuel rods were interpolated from the measured values. Since the benchmark was for two-dimensional models, the vertical buckling was determined by measuring the axial power profile for six fuel pins.

For this benchmark exercise, each participant performed cell calculations of k_∞ and of absorption and fission rates per isotope for each type of fuel cell. They also performed core calculations of k_{eff} and normalised pin power distributions for one-eighth of the core (325 pins).

Ten institutions participated in the benchmark, providing more than 14 solutions. Both deterministic and the Monte Carlo methods were applied, with different nuclear data sets. The deterministic codes used were a S_n code (DORT) together with SCALE/XSDRNPM, the collision probability codes HELIOS and BOXER, and a nodal diffusion code GNOMER in conjunction with a cell calculation code WIMS-D. Various versions of the continuous-energy Monte Carlo codes, such as MCNP-4B, MVP and MCU-B, were also used. The main nuclear data files used were ENDF/B-V, ENDF/B-VI, JEF-1, JEF-2, JENDL-3.2 and BROND. Thus, the most recent nuclear data files used world-wide were benchmarked.

The results of the cell calculations gave averaged values of k_∞ of 1.40593 ± 0.00393 for 3.3% ^{235}U cells, 1.33726 ± 0.00553 for 4.0% ^{235}U cells, and 1.25673 ± 0.00607 for MOX cells. Most of the results reported a deviation of less than 0.5%. In particular, the Monte Carlo calculations gave results with a deviation of less than 0.2%, which is the claimed uncertainty in the reactivity in current nuclear design methods. The results for absorption and fission rates were:

- *^{235}U*. Good agreement among the results was observed for both absorption and fission rates (less than 1% for absorption and less than 0.5% for fission).

- *^{238}U in MOX cell*. Most of the calculations showed less than 3% and 2% deviation from the averaged values for absorption and fission rates, respectively.

- *^{239}Pu*. Most of the results showed less than 0.5% deviation for absorption and fission rates.

- *^{240}Pu and ^{241}Pu*. The trends for both nuclides were very similar, deviations of a few per cent being observed. Overestimation or underestimation was also a few per cent, depending mainly on the nuclear data used.

- *^{242}Pu*. Deviations were higher for this nuclide, up to 10% for absorption rates and up to 6% for fission rates.

- *^{241}Am* – Deviations of about 5% were observed for both absorption and fission rates.

Core calculations results gave an average value of k_{eff} of 0.99758 ± 0.0045. The deterministic transport calculations gave an average of 0.99750 ± 0.0044, and the Monte Carlo calculations gave 0.99983 ± 0.0037. The maximum difference was -1.0%, given by the diffusion calculation with ENDF/B-VI.

On pin power distributions, excellent agreement with measured values was observed for two types of uranium fuel pins (deviation of less than 2.5% for 2.5%-enriched pins, and 1.0% for 4.0%-enriched pins). For the MOX fuel pins, the averaged deviation was larger than for uranium pins in most calculations, with about 4-5% overestimation being observed.

In general, the calculations were systematically high for the MOX region and low for the uranium region. In conclusion, the results were very encouraging and confirmed that present methods using the latest nuclear data sets can adequately calculate the behaviour of MOX-fuelled systems. As a follow-up to this two-dimensional benchmark, a benchmark using three-dimensional VENUS-2 experimental results is planned.

7.3 LWR stability and transient benchmarks

An expert group was established to deal with issues related to modelling and computational methods in the field of three-dimensional coupled neutronics and thermal-hydraulics transients for nuclear cores, and the coupling of core phenomena and system dynamics in PWRs, BWRs and VVERs. The transients considered included:

- LWR core transient benchmark [control rod ejection (PWR), and cold water injection and core pressurisation (BWR)].

- Uncontrolled withdrawal of control rods in PWRs.

- PWR main steam line breaks.
- BWR stability, time series and frequency analysis.
- Turbine trips benchmark for BWRs.
- VVER-1000 coolant transient benchmark (V1000-CT).

The objectives of the NSC in the field of coupled neutronics/thermal-hydraulics computations are to advance the scientific knowledge needed for the development of advanced modelling techniques for new nuclear technologies and concepts, as well as for current nuclear applications. This includes:

- Driving the development of coupled three-dimensional neutronics/thermal-hydraulics codes.
- Validating and benchmarking their performance through comparison with experiments.
- Verifying the correctness of methods and computer codes, building confidence in areas where experimental data are very expensive or lacking.
- Determination of model uncertainties.
- Promoting the use of these methods and codes in production runs and safety analysis.

7.3.1 LWR core transient benchmarks

The subject of this computational benchmark exercise, completed in 1993, was the calculation of reactivity transients in commercial-sized LWR cores, via space-time kinetics codes. The general object of the benchmark was to carry out a first survey of the state of the art in this area of analysis. For the sake of simplicity, all the two-group neutron cross-sections and most thermal-hydraulics data were imposed by the specification document.

For PWRs, the problems considered involved ejection of a control rod. Transients were initiated from hot zero power (2775 W) and full power (2775 MW) states for three different configurations. The reference core consisted of 157 fuel assemblies and 64 reflector elements. Cases A and B were defined in octant symmetry and further characterised by the ejection of a central (A) or a peripheral (B) control assembly. Case C was defined for full core geometry. Solutions were contributed from 13 industrial and national institutions from ten countries.

The problems proposed for BWRs were cold water injection and core pressurisation transients. Cold water injection over the whole core at the initial power of 1 600 MW was simulated by doubling the inlet water sub-cooling through an exponential increase with a 2.5 s time constant. The reference BWR core consisted of 185 fuel assemblies and 64 reflector elements. Solutions were contributed from eight industrial and national institutions from five countries.

The results obtained in the first phase of the PWR and BWR tests can be considered very satisfactory. The comparison was beneficial in different ways for individual codes, given the different stages of development and validation they had reached. Some participants were able to spot weaknesses in their solutions and to take corrective measures. Many codes based on three-dimensional coarse mesh methods were known to have reached a stage of advanced development or testing.

The results of this benchmarking exercise were published as document NEA/NSC/DOC(93)25.

7.3.2 Uncontrolled rod withdrawal at zero power

Due to the great interest in the control rod ejection benchmark described above, a second benchmark exercise was undertaken to validate the codes for another of the standard problems of PWR core safety analysis – a rod withdrawal at zero power accident. This used the same core model as the first benchmark. From the critical initial state at zero power, one or two control rod banks were withdrawn at the maximum speed (72 steps/minute). The evolution consisted basically of a continuous-reactivity insertion, limited by the reactor trip, which would occur with a certain delay after high-flux detection (35% of nominal fission power). Four cases were considered:

- Cases A, B and D differed in the location of the involved control rod banks and, as a consequence, the injected reactivity.
- Case A represented a single bank withdrawal, other banks remaining fully withdrawn until scram.
- Case B represented a double bank withdrawal.
- Case C was the same as Case B, except that the fuel-water heat transfer coefficient was fixed.
- Case D started from the same initial conditions as Case B, but more peripheral rods were withdrawn.

The submitted solutions were compared to a reference result obtained with a nodal code (PANTHER) using finer spatial resolution (3×3 nodes per assembly and 48 axial nodes) and temporal resolution than in a standard calculation. Eleven (11) solutions were submitted from ten countries. The codes used a nodal model for direct three-dimensional calculations, except for one solution using the synthesis method.

A good agreement was obtained for most of the codes on the power evolution and its integral, in particular for the core-averaged parameters. However, the time-of-power surge seemed more sensitive to the model and to the nodalisation. For the "hot pellet", the spread of the results was more important, especially for hot pellet enthalpy and envelope power profiles. It suggested that care must be taken to ensure consistency between the calculation methodology and the criteria applied to the safety analyses.

The axial and radial power distributions were very similar for all of the participants as far as the whole core was concerned. The axial envelope profiles reflected the differences obtained at the hot pellet level. Because of the lack of measurements on reactivity insertion transients, much interest was expressed in the results of this benchmark, which were published as document NEA/NSC/DOC(96)20.

7.3.3 PWR main steam line break (MSLB) benchmark

The PWR MSLB benchmark problem, jointly approved by the NEA and the US Nuclear Regulatory Commission (NRC), used a three-dimensional neutronics core model to further verify the capability of coupled codes to analyse complex transients with coupled core-plant interactions and to fully test the thermal-hydraulics coupling. It was based on real plant design and operational data for the Three Mile Island-1 nuclear power plant. The purpose of this benchmark was:

- To verify the capability of system codes to analyse complex transients with coupled core-plant interactions.

- To fully test the three-dimensional neutronics/thermal-hydraulics coupling.

- To evaluate discrepancies between predictions of coupled codes in best-estimate transient simulations.

The benchmark consisted of the following three exercises:

- A point kinetics plant simulation, which modelled the primary and secondary systems. The purpose of this exercise was to test the thermal-hydraulics system response.

- A coupled three-dimensional neutronics/thermal-hydraulics evaluation of core response. The purpose of this phase was to test the neutronics response to imposed thermal-hydraulics conditions.

- A best-estimate coupled core-plant transient model. This exercise simulated the entire transient and combined the first two exercises, fully testing the neutronics/thermal-hydraulics coupling.

Each computer code used separate temporal and spatial models and numerical models for core neutronics, core thermal-hydraulics and system thermal-hydraulics simulation. The ultimate goal was to enable participants to initiate and verify these models before focusing on the major objective – testing the coupling methodologies in terms of numeric, temporal and spatial mesh overlays. In order to perform such a comprehensive validation of coupled codes, a multi-level methodology was employed. It included the application of three exercises, the evaluation of several steady states and simulation of two transient scenarios. Over the course of the benchmark activities a professional community was created, and its members were involved in an in-depth discussion of different aspects considered in the validation process. International workshops and *ad hoc* meetings played important roles through the evaluation processes.

In Phase I of the exercise, 14 solutions were submitted from eight countries. It was observed that major factors affecting the dynamics of the transient were break flow modelling (critical flow model), liquid entrainment, modelling of the aspirator flow and nodalisation of the steam generator (SG) downcomer. The key parameters were SG mass, break flow rate, coolant and fuel temperatures, and power level. It was useful to analyse other parameters because they helped to determine what was causing the behaviour of the key parameters. It was proven that the SG model has a great effect on the power throughout the transient.

One of the most important lessons learned from the first phase of the benchmark was that it is extremely important, as well as very difficult, to ensure consistency in the modelling for any problem. The results showed that point kinetics analysis may be overly conservative and thus may limit the operational flexibility of nuclear plants in the areas of higher burn-up, longer fuel cycle and power upgrading.

Twenty solutions from eight countries were submitted in Phase II, and 11 solutions from 10 countries for Phase III. For comparison and evaluation of the submitted results, a statistical methodology was established. The reference values were calculated based on the statistical mean value of all submitted results, except obviously outlying solutions. In general, agreement amongst the global core results was good, except for one solution. Most of the deviations were observed in the local parameter predictions, which was a consequence of the thermal-hydraulics nodalisation and spatial coupling schemes used. The reason was partially that participants used different schemes to spatially distribute the decay heat.

Based on the parametric study, the results indicated that the total transient power evolution during the MSLB simulation was not sensitive to the radial heat structure refinement. As in the case with neutronics scheme refinement, the impact was mostly on local distributions. In addition, the MSLB calculations were sensitive to the detail of the thermal-hydraulics core modelling. The results were published as documents NEA/NSC/DOC(2000)2 and NEA/NSC/DOC(2000)21.

During the course of defining and co-ordinating the MSLB benchmark, a systematic approach was developed which contributed to the determination of additional requirements and needs for the validation of best-estimate coupled codes. The resulting improved methodology was subsequently employed to develop a similar benchmark for BWRs.

7.3.4 Ringhals-1 stability benchmark

In NEA member countries there have been several examples of inadvertent power oscillations in BWRs. Such experiences showed the need for further activities in R&D, as well as in licensing and in finding new core design and operation strategies. There are two possible types of instability: oscillation in-phase or global mode, and out-of-phase or regional mode (sometimes with overlapping of the two modes). Some measurements provided data for the validation of codes and methods, but they were limited. It was clear that a comprehensive and well-defined set of data from measurements assembled in an international benchmark exercise would be of interest to many code developers and could also be of value in licensing efforts.

The data came from beginning-of-cycle measurements in cycles 14, 15, 16 and 17 at the Swedish BWR Ringhals-1. The benchmark comprised in total 41 state points from four cycles, each with measured and evaluated decay ratios and natural frequencies.

Nine participants from 8 countries, featuring several calculation models, submitted their solutions and joined the discussion. The codes applied in the best-estimate mode displayed a very small bias in global decay ratio, and an uncertainty (one sigma standard deviation) in the calculated decay ratio in the range 0.06 to 0.10. The codes under validation showed higher fluctuation between cycles. In summary, it was shown that calculation of the global stability characteristics of a BWR could be performed with a precision rather close to that of the noise analysis methods used to evaluate the corresponding stability parameters from the raw data. This was true for frequency domain codes as well as time domain ones.

As for the regional stability characteristics, it was shown that the methods using both time-series evaluation and calculation of oscillation parameters needed to be refined. A regional oscillation is more troublesome for present BWR core monitoring and core protection systems. A calculation tool is needed to predict with some confidence which core mode will be dominant in a reactor with certain operating conditions and core design.

The results of this benchmarking exercise were published as document NEA/NSC/DOC(96)22.

7.3.5 Forsmark-1 & -2 stability benchmark

A recommendation from the Ringhals-1 benchmark was to study the different time series analysis methods in order to obtain a unified methodology to detect and suppress oscillations during reactor operation, as well as better qualification of the applied noise analysis methods. A follow-up benchmark was thus proposed, dedicated to the analysis of time series data and including the evaluation of both global and regional stability.

The benchmark was based on data from several measurements performed in the Swedish BWRs Forsmark-1 and -2 in the period 1989 to 1997. The data were divided into six cases. Fifteen solutions were submitted by participants belonging to ten organisations in 8 countries. The following methods were used for the analysis: auto-regressive methods and domain poles; auto-regressive methods and impulse response; auto-correlation; recursive auto-correlation; auto-regressive moving average method (plateau method); and power spectrum estimation.

The results of the benchmark study were published as a report entitled *Time Series Analysis Methods for Oscillation during BWR Operation* [NEA/NSC/DOC(2001)2]. Its main conclusions were:

- For noise analysis, the decay ratio was associated with the least stable or dominant pole. The definition was clear for a second-order system.
- For determination of decay ratios, the asymptotic part of the transformation function should be used.
- The decay ratio could be determined automatically, without filtering, if expert tuning to the plant was carried out.
- The time duration required for determining the decay ratio was about inversely proportional to the value of the decay ratio.
- The decay ratio of out-of-phase oscillations could be determined for values up to 0.7 ± 0.1, if enough local power range monitor signals per plane were provided. This depended on the expertise of the analyst or the sophistication of the monitoring algorithm.
- Sufficiently accurate limits to the stable behaviour of the reactor core could be determined using codes in the frequency domain. They were efficient but not sufficient. The real margin should be determined on power. The decay ratio is a measure of linear stability and should therefore not be used as the only indicator of BWR stability.

It was verified that the different methods consistently provided the same answer to a good degree.

7.3.6 BWR turbine trip benchmark

The NEA and the US NRC jointly approved a BWR turbine trip benchmark for the purpose of validating advanced system best-estimate analysis codes. The object of the benchmark was to establish a BWR turbine trip exercise, based on a well-defined problem with a complete set of input specifications and reference experimental data, for qualification of coupled three-dimensional neutronics/ thermal-hydraulics system transient codes.

Since this kind of transient is a dynamically complex event with reactor variables changing very rapidly, a well-constituted benchmark problem would need to test the coupled codes on both levels: neutronics/thermal-hydraulics coupling and core/plant system coupling. Consequently, the objectives of the benchmark were comprehensive feedback testing and examination of the capability of coupled codes to analyse complex transients with coupled core/coolant interactions by comparison with actual experimental data.

The benchmark project was established against a turbine trip transient with a sudden closure of the turbine stop valve at the Peach Bottom-2 BWR in the USA. Three turbine trip transients at different power levels were performed prior to shutdown for refuelling in April 1977. The benchmark consisted of three separate exercises:

- Power versus time plant system simulation with fixed axial power profile table (obtained from experimental data).
- Coupled three-dimensional kinetics/thermal-hydraulics BC model and/or one-dimensional kinetics plant system simulation.
- Best-estimate coupled three-dimensional core/thermal-hydraulics system modelling.

Three workshops were held during the course of the benchmark activities. These were held at: Exelon Generation, Philadelphia, USA, 9-10 November 2000 [published as NEA/NSC/DOC(2000)22]; Paul Scherrer Institute, Switzerland, 15-16 October 2001 [NEA/NSC/DOC(2001)20]; and finally, Forschungzentrum Rossendorf, Germany, 28-30 May 2002. A fourth workshop was scheduled for Seoul, Korea.

A summary of the benchmark results was published in four volumes as NEA and NUREG/CR reports [NEA/NSC/DOC(2001)1].

7.3.7 VVER-1000 coolant transient benchmark

During the course of defining and co-ordinating the joint NEA and US NRC PWR MSLB and BWR turbine trip benchmarks, a systematic approach was established to validate best-estimate coupled codes. This approach employs a multi-level methodology that allows not only for a consistent and comprehensive validation process but also contributes to determining additional requirements. It also prepares a basis for licensing application of the coupled calculations for a specific reactor type, i.e. establishing a safety expertise in analysing reactivity. The above examples demonstrate the benefit of establishing such coupled standard benchmarks from the available real plant experimental data.

A Soviet-designed VVER-1000 reactor was selected as the next application of thermal-hydraulics system codes, based on actual plant data. A coupled benchmark problem based on data from the Kozloduy VVER-1000 nuclear power plant in Bulgaria was developed for the purpose of assessing kinetics and three-dimensional kinetics models.

The reference problem chosen for simulation was a main coolant pump activation while the other three main coolant pumps were in operation, which is a real transient in an operating VVER-1000 power plant. This event is characterised by a rapid increase in the flow through the core, resulting in coolant temperature decrease, which is spatially dependent. This causes insertion of spatially distributed positive reactivity due to the modelled feedback mechanisms and non-symmetric power distribution. Simulation of the transient required evaluation of core response from a multi-dimensional perspective supplemented by a one-dimensional simulation of the remainder of the coolant system.

The purposes of this benchmark exercise were: to verify the capability of system codes to analyse complex transients with coupled plant interactions; to fully test the three-dimensional neutronics/thermal-hydraulics coupling; and to evaluate discrepancies between predictions of coupled codes in best-estimate transient simulations. The benchmark consisted of three separate exercises:

- *Point kinetics plant simulation*. The purpose was to test the primary and secondary system model response.

- *Coupled three-dimensional neutronics/core thermal-hydraulics response evaluation*. The purpose was to model the core and vessel only; inlet and outlet core transient boundary conditions were provided.

- *Best-estimate coupled code plant transient modelling*. This combined elements of the first two exercises and was an analysis of the transient in its entirety.

The results of this benchmarking exercise were published as document NEA/NSC/DOC(2002)6.

7.4 Radiation shielding

7.4.1 Shielding Integral Benchmark Archive Database (SINBAD)

The Shielding Integral Benchmark Archive Database (SINBAD) is an international effort between the NEA Data Bank, as the lead organisation, and the Oak Ridge National Laboratory Radiation Safety Information Computation Center (RSICC) in the USA. Co-operation from many organisations, authors and benchmark analysts has helped SINBAD become a living database, which involves continuous information updates, preservation of the original data sources and the addition of additional nuclear experimental benchmarks of suitable quality.

SINBAD is a tool for accessing user-friendly, quality-assured information that is widely available to modellers of nuclear technology information. Through the continued inclusion of contributions from the nuclear community to SINBAD, international nuclear computational modellers and nuclear data users will further contribute to safety and reliability in new nuclear power plants, isotope production, nuclear medicine, nuclear fuel/waste handling and other areas of focus in current radiation applications. The database is playing an essential role in validation and benchmarking of computer codes and nuclear data used for radiation transport and shielding problems.

Several different levels of information are included in the database:

- Primary documents in computer-readable image form, using optical character recognition.

- A description of the facility and experimental results in a standard form.

- Evaluation results, and their interpretation and review through analysis of experiments with state-of-the-art computer codes and data.

- Peer-reviewed benchmark data, given by a hypertext mark-up language (html) based electronic management system.

The mission of SINBAD includes the following key areas that are integral to its success: data collection; information preservation; data integration; compilation format; computational aspects; code and data quality assurance; updating/correcting information and user feedback; standards development; and international co-operation in nuclear science and technology development.

A large number of experiments in the database are related to pressure vessel dosimetry. Data from 60 experiments have been collected. The major emphasis has so far been on fission reactor shielding data (representing 66% of the total). Some facilities used for measurements have now been closed down, leading to an urgent need to preserve the data. Data for fusion blanket neutronics represent 25%, the rest being devoted to accelerator shielding experiments. About 60% of the data have been compiled and peer reviewed, about 30% have been compiled but are awaiting peer review and the rest are in the process of being compiled. More data sets continue to be identified for future release. Emphasis will be on the quality of the experiments, and new compilations will address cases not yet sufficiently covered by existing data sets. Over 68 copies of SINBAD have been distributed to 20 countries since 1997.

A new release of SINBAD was issued at the end of 2000, and a further version was being readied for release in 2002. As of February 2002, SINBAD had received contributions from 22 organisations, institutes and universities. It covered 32 cases of nuclear reactor shielding, 12 cases of fusion neutronics shielding and six cases of accelerator shielding. The total number of materials covered was 22 (for some materials there was more than one measurement with different conditions, e.g. type of source, source energy, geometry or composition). The database is available to organisations in NEA member countries at no charge from the NEA Data Bank.

The SINBAD database needs to be further extended to include additional experiments that increase its comprehensiveness and coverage for validation requirements in support of new designs. Some experiments (about 50 measurements) have been identified and are being included or may be included in the future.

7.4.2 Shielding aspects of accelerators, targets and irradiation facilities

Accelerators are playing an increasingly important role in certain technological areas, such as energy and medical applications. As they become more widely used and their power increases, challenging new problems arise in characterisation of radiation environments and in radiological safety. Beginning in 1994, the NSC sponsored a series of six specialist meetings on this topic. The Expert Group on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF) was set up in June 2000.

This expert group deals with multiple aspects related to radiation safety, activation and shielding modelling, and the design of accelerator systems. These include electron, proton and ion accelerators, spallation sources, synchrotron radiation facilities, transmutation sources, accelerator-driven systems, free electron lasers, and high-power targets and dumps. Its objectives are:

- To promote the exchange of information among scientists within the defined scope.
- To identify areas in which international co-operation could be fruitful.
- To carry out a programme of work to achieve progress in agreed priority areas.
- To encourage free access to computer codes sources and to cross-section and integral experimental data, including making them available at information centres.

The main activities of the expert group are:

- Assessing the needs for experimental data for the validation of models and codes.
- Assessing the needs for evaluated nuclear data and processed data libraries.
- The organisation of shielding experiments.

- Collecting and compiling experimental data sets.

- Assessing the models, computer codes, parameterisations and techniques available for the design of accelerator shielding.

- Validating the computer codes and models available to perform particle transport simulation.

- Holding a specialist meeting at least once every two years, and publishing the proceedings.

- Reporting to the Nuclear Science Committee.

The six SATIF specialist meetings were held in: Arlington, Texas, USA, 28-29 April 1994; CERN, Geneva, Switzerland, 12-13 October 1995; Tohoku University, Sendai, Japan, 12-13 May 1997; Knoxville, Tennessee, USA, 17-18 September 1998; OECD, Paris, France, 17-21 July 2000; and SLAC, Stanford, California, USA, 10-12 April 2002.

The proceedings of the SATIF meetings have been published by the NEA. About 50 specialists attended each meeting, including physicists, engineers and technicians from laboratories, institutes, universities and industries in 7-9 countries and international organisations. During the meetings, special attention has been paid to:

- The availability and compilation of experimental data for different applications.

- The organisation of international benchmark exercises.

- The availability of computer codes and data libraries for the use of the scientific community involved in various aspects related accelerator shielding and applications.

- Identifying areas in which international co-operation could contribute to the solution of existing problems.

The presentations made at the SATIF meetings can generally be categorised into the following topics: data, benchmarks, computer codes and models, facilities, shielding techniques, conversion coefficients, code comparisons, and status of codes.

7.4.3 Three-dimensional radiation transport benchmarks

The Expert Group on Three-dimensional Radiation Transport Benchmarks is concerned with scientific issues in the field of deterministic and stochastic methods and computer codes relevant to three-dimensional radiation transport. Applications encompass transport through large and complex shields including void regions and highly heterogeneous reactor cores. Methods include discrete ordinates, nodal transport, finite elements with spherical harmonics, collision probabilities, Monte Carlo, etc. The objectives of the expert group are:

- To develop benchmarks and comparison exercises for three-dimensional radiation transport computer codes.

- To carry out validation of methods and identify their strengths, limitations and accuracy.

- To suggest requirements for method development.

Several NSC activities are concerned with the validation of computation methods and codes as applied to nuclear technology. One of the main challenges is refined modelling of the full geometrical complexity of real problems in practical applications. Two types of method for three-dimensional modelling have emerged – the stochastic Monte Carlo method and deterministic methods.

In order to address this issue, a series of three-dimensional neutron transport benchmarks, known as the "Takeda Benchmarks" was organised under the auspices of the NSC (see NEACRP-L-330 of March 1991). These concerned small, highly heterogeneous reactor cores. A seminar entitled "Three-dimensional Deterministic Radiation Transport Computer Programs: Features, Applications and Perspectives" was held in Paris in December 1996.

During the above seminar, an additional benchmark study on three-dimensional radiation transport for simple geometry with void regions was proposed to clarify issues of precision regarding the different methods used for flux calculations. The proposal was made by Professor Keisuke Kobayashi from Kyoto University, Japan, to study a pure absorber problem with internal void regions, which was further extended to include cases with 50% scattering.

There were eight contributions for the benchmarks. Six were obtained by using discrete ordinates methods, and two used the spherical harmonics method. In conclusion, the accuracy of the discrete ordinates methods with the first collision source was best for these benchmark problems. It was expected that these benchmark problems would help further improve three-dimensional transport programs based on the spherical harmonics method as well as the discrete ordinates method.

The results of these benchmarks were discussed at a meeting in Madrid, Spain, on 1 October 1999. The results of the analysis were summarised in a NSC report and were also published in a special issue of *Progress in Nuclear Energy* in 2001.

A new benchmark study of deterministic two- and three-dimensional MOX fuel assembly transport calculations without spatial homogenisation was launched in 2001 by the NSC [see NEA/NSC/DOC(2001)4]. It concerns pin-by-pin power distribution within core assemblies using transport theory with cross-sections in seven energy group structures. The final specification was distributed on 28 May 2001.

7.5 The Monte Carlo method

Following discussion of the potential of the Monte Carlo (MC) method in nuclear applications and radiation physics, the NSC has supported and sponsored some international conferences to discuss the impact this method can have in industrial use. Significant progress has been achieved in applications for which using the MC method is advantageous by demonstrating its potential and its successful use.

During the last few years, five conferences co-sponsored by the NEA have taken place during which particle transport MC methods had a prominent place in the programme or were the subject of the conference itself. These were: M&C'99 in Madrid, Spain; ICRS9 in Tsukuba, Japan; PHYSOR 2000 in Pittsburgh, USA; SNA 2000 in Tokyo, Japan; and MC 2000 in Lisbon, Portugal. The NEA also sponsored the Workshop on Advanced Monte Carlo Computer Programs for Radiation Transport, held at Saclay, France, in April 1993.

Each of these conferences showed that the Monte Carlo method is used increasingly widely. The reasons for this are:

- Computer architectures including parallel and vector processing have evolved that are particularly well suited to increasing the speed and reducing the statistical uncertainty of MC codes in solving problems.

- It largely removes uncertainties arising from model simplification.

- Powerful biasing schemes have been developed, which are now well established and used in widely known computer codes. Statistical analysis has been further developed and the method has become more user-friendly as it has matured.

These recent conferences have confirmed that the MC method is now used in many applications. Particularly intensive use is made in radiation physics, diagnostics in material identification, material science, and radiological and medical applications. Other areas of wide use are deep penetration of radiation into matter and radiation shielding, including intermediate particle energies applications. Criticality safety is a field where the MC method is used as a standard analysis tool. It is particularly suitable for calculations of the integral parameter k_{eff} describing the level of criticality.

The use of the MC method in the area of nuclear power has undergone an important evolution. Notable successes are the extensions to compute burn-up in reactor cores and full-core neutronics simulations. Aspects concerned with material or geometry perturbation are beginning to be successful after long development periods. First results from sensitivity analysis with the MC method are promising, but still limited.

However, in many aspects of nuclear power plant simulation the MC method is still not applicable and its use would require much further development, e.g. in the kinetics field. Deterministic methods will continue to play an important complementary role. A symbiosis of stochastic and deterministic methods (including coupled and hybrid methods) can be expected for many more years.

In order to meet the increasing interest and needs of the nuclear community, several training courses have been organised every year. During these courses code users learn how to carry out efficient modelling with the MC method. Participation in these courses has been over 400 mostly young persons during the last few years.

In conclusion, the Monte Carlo method has proven to be very successful, in particular for radiation transport problems. Its use will further increase, particularly if development of the methods is pursued. In order to foster such development this topic should continue to be on the agenda at international conferences, and a series of specific MC conferences is justified and should be maintained. The NSC continues to support such activities.

7.6 Reactor surveillance and diagnostics

The NSC has organised a series of symposiums and specialist meetings in the field of reactor surveillance and diagnostics. These meetings concern research and applications regarding surveillance and diagnostics methods aimed at guaranteeing the safety, availability and efficient operation of various types of nuclear rectors.

The SMORN meetings are international symposiums related to studies and applications of reactor noise for the purpose of monitoring, diagnosing and improving knowledge of the physical processes in nuclear reactors. The first few SMORN meetings concentrated on solutions to direct tasks, i.e. calculating the noise induced by known perturbations. At the very first SMORN meeting even zero-reactor noise problems were treated. At later meetings the emphasis shifted to elaborating inversion procedures, and more and more realistic calculations of the transfer function. The last few meetings demonstrated the possibilities of noise analysis techniques and their application in nuclear reactors, even in the form of complete systems in routine usage.

Altogether, eight SMORN meetings have been held: Göteborg, Sweden, May 2002; Avignon, France, 1995; Gatlinburg, USA, 1991; Munich, Germany, 1987; Dijon, France, 1984; Tokyo, Japan, 1981; Gatlinburg, USA, 1977; and Casaccia, Italy, 1974.

The Workshop on Core Monitoring for Commercial Reactors: Improvements in Systems and Methods (CoMoCoRe'99) was held on 4-5 October 1999 in Stockholm, Sweden. The main objective was to discuss how instrumentation, methods and models used in core monitoring could be validated and, if necessary, improved and further developed to provide more reliable and/or detailed information on local power in the core and on other parameters indirectly affecting fuel duty, e.g. the core decay ratio in a BWR. Another important objective was to show how the core monitoring system can be used to support reactor operation in normal and anticipated transient modes, and to supply the data used to derive initial key core parameters for transient and accident analysis. The workshop dealt with applications for all commercial LWR types, including VVERs.

A series of specialist meetings on in-core instrumentation and reactor core assessment has also been held: in Mito, Japan, 1996; in Pittsburgh, USA, 1991; in Cadarache, France, 1988; and in Fredrikstad, Norway, 1983.

7.7 Advanced computing

The Working Party on Advanced Computing was set up in 1991 by the NSC to address four areas of work relating to hardware and software in the nuclear industry. These areas were: scientific applications software; standards and quality assurance; process control systems; and supercomputing. A task force was established for each area of work.

The Task Force on Supercomputing has prepared a state-of-the-art report on adaptation of computer codes in nuclear applications to parallel architectures. The report identifies "grand challenge problems" in nuclear applications based on present and future needs. It also includes an assessment of high-performance computing, such as indications as to which type of computer architecture is best suited for different applications, and the cost/benefit of developing new algorithms taking advantage of the special feature of emerging high-performance computers. The following application areas were reviewed: Monte Carlo radiation transport; deterministic radiation transport; computational mechanics and fluid dynamics; reactor safety analysis; and waste management.

Chapter 8

CSNI ACTIVITIES RELATED TO SAFETY RESEARCH

8.1 Background

The international nuclear community has been concerned for some time about the ability of countries to sustain an adequate level of safety research. In recent years both government and industry funding of research has decreased in many countries. Governments often believe that nuclear is a mature technology and therefore that increased reliance should be placed on industry to fund the necessary research. Industry, in turn, has often reduced its involvement in funding safety research because there is little commitment to building new plants and because there is a belief that the research needed to operate existing plants and to prevent and manage possible accidents is largely complete. Furthermore, electricity market competition has tended to focus attention on short-term profitability, sometimes at the expense of long-term research.

Excessive reduction in safety research leads to the loss of continuing safety knowledge, the consequent loss of research facilities and expertise, and the loss of academic interest in safety research. It may thus affect the safe operation of existing nuclear power plants in the medium and long terms, the ability of regulatory bodies to meet their obligations in a competent and independent manner, and ultimately the ability to design and build new plants. Closely linked to this is the problem of attracting and retaining a young generation of scientists in nuclear research.

In general, there is recognition that the reduction in safety research may have gone too far. Some action is being taken to deal with the situation. Several countries are conducting studies to assess the needed research capability and are making arrangements to ensure that essential capability is available. Although solutions to this problem must primarily come at the national level, improved co-operation between the industry and regulators, and improved international co-ordination and co-operation, can make a very important contribution. Through the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA), the NEA has been active since the early 1990s in looking into specific issues and promoting international co-operation to deal with the problem of maintaining adequate research capability.

This chapter summarises the actions taken by the NEA to help its member countries deal with the problem of maintaining adequate safety research programmes in order to ensure effective regulation and operation of nuclear power plants. In particular, it summarises the conclusions of a major workshop held in June 2001 at which senior regulators, heads of industry and leading researchers discussed in broad terms what is needed to ensure adequate research capability.

8.2 Activities of the CSNI

Within the NEA, the CSNI and the CNRA have key roles in providing member countries with authoritative advice on matters relating to the safety and regulation of nuclear facilities. The CSNI concentrates on the technological and research aspects of nuclear safety, while the CNRA concentrates on the regulatory and policy aspects. The maintenance of essential research capabilities and facilities has been of concern to the CSNI since the early 1990s. A programme of work was started with the essential aim of gathering information, analysing it and developing a strategy for the efficient management of essential nuclear safety research.

In 1992 a Senior Group of Experts on Safety Research (SESAR) was established. This group brought together senior research managers of major research programmes in NEA member countries. The first task of this group was to review the status of research being carried out and draw conclusions on future requirements and priorities [1]. The group then went on to identify areas of agreement, areas where further action was required and the need for increased collaboration [2].

An important outcome of these studies was the raising of concern about the loss of critical research facilities and competence, which could lead to the ability to adequately regulate and support the safe operation of nuclear facilities being undermined [3]. This concern led the CSNI to request that the group focus more specifically on research capabilities and facilities likely to be at risk, and to put forward proposals for international programmes that could help secure an adequate research infrastructure [4].

In its work, the group started by summarising the main drivers of research for the medium and long terms. These were identified as:*

- Plant life management, including ageing of components, systems and structures (hardware), ageing of analytical tools and documentation ("paperware"), application of modern standards to older plants, life extension and backfitting.

- Optimisation of operating margins, including power uprating, higher fuel burn-up, more extensive use of probabilistic safety analysis (PSA), etc.

- Severe accidents, including the need to further develop practical accident management procedures and design solutions for future plants.

Subsequently, discussion within the group concentrated on determining the areas in which research facilities and programmes were essential, and where internationally co-ordinated action was necessary. In essence, the work of the group consisted of correlating the technical areas that needed to be monitored with the research facilities needed to provide the relevant data and which might be at risk in the short and medium terms.

The technical areas considered were all those relevant to nuclear safety, namely:

- Thermal-hydraulics.

- Fuel and reactor physics.

- Severe accidents.

- Human factors.

- Plant control and monitoring.

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* Since this report was written, interest in new reactor concepts has further increased. Regulatory authorities need to have the proper competencies and experimental facilities to evaluate the safety performance of such concepts.

- Integrity of plants and structures.

- Seismic analysis.

- Risk assessment.

- Fire risk assessment.

To judge the suitability of a particular facility, a set of criteria was developed. The two overriding criteria were whether the facility could provide knowledge to address a safety issue which was considered open based on earlier work of the group, and whether the relevant capability or facility was under threat. Other criteria used were:

- *Uniqueness to the nuclear industry*. Resources should be focused on capabilities and facilities where there are no other front-line industrial or research interests. Examples of areas unique to the nuclear industry are core melt progression experiments, specific thermal-hydraulic test rigs, test reactors and critical facilities, and hot cells. Examples of non-uniqueness are instrumentation, heat transfer, basic computational fluid dynamics (CFD) developments and some aspects of human factors.

- *Applicability to a broad range of conditions*. Capabilities and facilities need to be flexible and able to accommodate different users' needs. Scaling criteria are also relevant, if the technical demands of the subject area require it.

- *Responsibility*. The ownership of the capability (either government or industry) must be clearly established, and there should ideally be a clear commitment from the owner to support any future international programme in its facility. However, for specific and important facilities and programmes, the group raised the possibility of international action even if the host country was not initially supportive.

- *Credibility*. Management provisions must be acceptable and meet current standards, including proper financial, quality and technical controls.

- *Size*. In order to limit the number of possible projects under consideration, an initial cut-off point of projects valued at more than US$1 million was used.

Using these criteria, the group examined a very large number of facilities and identified a number of long-term actions which were required to address the needs of an adequate programme. Obviously, these needs varied from country to country depending upon the nuclear technology in use and the national arrangements in place for funding and conducting safety research. However, a number of collective needs for the NEA member countries could be identified.

This work concluded with a large number of recommendations, some of a strategic nature and some dealing with specific facilities and programmes. Some of the key recommendations were:

- *Thermal-hydraulics*. Maintain one major facility per reactor type because of the need to perform confirmatory tests, to support code development and to provide educational opportunities.

- *Severe accidents*. Address the need for a centre of excellence in fuel/coolant interaction and fission product behaviour.

- *Fuel and reactor physics and integrity of structures*. Maintain the status of hot cells and research reactors.

- *Human factors and plant control and monitoring*. Maintain the Halden project as a centre of excellence.

- *Seismic*. Monitor the availability of large shaking tables.

- *Fire safety*. Establish an international database, and consider possible additional research.

The work of SESAR is being followed up in various ways. The CSNI has created a programme review group, which has among its tasks that of monitoring developments in the research infrastructure of NEA member countries and periodically updating the list of facilities and programmes at risk (given in Ref. [3]). The specialist knowledge of CSNI working parties is being used for this purpose. In addition, the NEA has considerably expanded the establishment of internationally-funded research projects based on the conclusions of the SESAR study and detailed discussions at the technical level. Table 8.1 shows the current status of NEA-sponsored research projects.

Table 8.1. The status of NEA-sponsored research projects

Name	Period	Technical area	Status
HALDEN	2000-2002	Fuel/human factors	Ongoing
RASPLAV	1997-2000	Severe accidents	Completed
MASCA	2000-2002	Severe accidents	Ongoing
CABRI-WLP	1999-2007	Fuel	Ongoing
SANDIA-LHF	1998-2001	Severe accidents	Ongoing
SETH	2001-2005	Thermal-hydraulics	Ongoing
MCCI	2002-2005	Severe accidents	Ongoing
ICDE	2000-2002	Operating data	Ongoing
PLASMA	1999-2000	Human factors	Completed
OPDE	2002-2005	Piping reliability	Ongoing
FIRE	2002-2005	Fire safety	Ongoing
PSB-VVER	2003-2006	Thermal-hydraulics	Ongoing

A number of other recommendations are being followed-up at the level of the CSNI working parties. For example, in the area of fuel/coolant interaction a co-ordinated research programme (SERENA) is being conducted by a number of countries under the sponsorship of the CSNI.

8.3 Workshop on Research in the Regulatory Context

To complement the work of the CSNI on safety research, it was felt important to hold a senior-level workshop with the following objectives:

- To exchange views among the three major groups involved, i.e. regulators, researchers and industry executives, on the needs and vision for research in the regulatory context.

- To identify commonalties and differences, and define additional activities at the international level that may be needed.

The workshop was held in June 2001 and saw the participation of approximately 100 senior experts. It was structured in three sessions. The first dealt with the vision and needs of regulators, and the second with the vision and needs of researchers and industry. The third session was used to identify ways to move the discussion forward.

Prior to the workshop a survey was conducted in NEA member countries to gather basic information on research needs, adequacy of current programmes, trends in programmes and funding levels and action being taken to overcome some of the problems. Some of the major aspects of regulatory-related research highlighted by the results were:

- A research programme in support of the safe operation of NPPs was deemed necessary in essentially all NEA member countries.

- All countries participated in international research projects, though to widely varying degrees. Typically, small/medium-sized countries devoted 20-30% of their budgets to international projects, while major nuclear countries dedicated approximately 10%.

- No clear trend could be seen regarding past, current and future funding of safety research, although there appeared to be a trend toward maintaining or increasing funding in the following five years.

- Sources of funding in NEA member countries tended to reflect the extent to which the government or the nuclear industry was regarded as having the prime responsibility for resolving nuclear safety issues. For example, in Japan, Germany and Spain a very high proportion of the funding came from governments, while in the USA, Sweden and the UK a very large proportion came from industry.

- Essentially all countries shared concern about the need to maintain research facilities. A number of countries had undertaken a strategic review of their capabilities and needs.

In the first session of the workshop, heads of regulatory organisations addressed questions such as why research should be supported, what types of research should be funded and the role of international organisations in setting up and funding research. A common conclusion was that a strong research programme was a central feature of a sound regulatory system. Research was needed by the regulator to provide independent judgement, to determine areas in which improvements might be necessary, to anticipate potential problems and in general to improve the effectiveness of the regulatory system and ensure that regulatory requirements were adequate and practical.

It was recognised that one of the key challenges for regulators was to maintain the proper balance between confirmatory research, such as that conducted to validate methods, and anticipatory research, such as that conducted to anticipate potential problems and improve knowledge. With a decreasing budget it was always easier to justify confirmatory research at the expense of anticipatory research.

International co-operation was felt to be important for several reasons. One reason was simply to be able to leverage budgets and avoid duplication of programmes. Other reasons included the “magnification of intellectual firepower” which came from interaction among researchers, the increased possibility of involving countries with limited resources, and the contribution to harmonisation of safety requirements from achieving common technical positions. Such co-operation could also help by stimulating and motivating young scientists to work in the nuclear field.

In the second session, issues discussed included ways to increase co-operation between industry and regulators in research, while at the same time maintaining the independence of regulatory decisions and preserving the freedom to choose research subjects.

Industry representatives emphasised that the industry should be allowed to choose the method of demonstrating the safety of nuclear installations. Studies, calculations, design modifications, etc. were in many cases an acceptable alternative to further research. They also emphasised the need to establish achievable closure criteria for safety, and to improve the alignment between industrial and regulatory research, particularly with respect to best-estimate analyses and determination of margins.

The basic conclusions of the workshop were documented in a collective statement by the CNRA and the CSNI. The document outlines areas of agreement between the three parties involved in safety research, as well as areas where differences exist. It reports on the consensus achieved and includes a number of recommendations.

In response to these recommendations, the CSNI conducted a study to identify and review issues which hinder closer co-operation on research between regulators and industry, and to propose possible ways of resolving such issues. The study examined issues such as resources and sources of funding, public availability and utilisation of results, independent decision-making, and research on very low-probability events. A report entitled *Regulator and Industry Co-operation on Safety Research: Challenges and Opportunities* summarises the results of this study.

In addition, existing groups of the CSNI will continue to discuss possible types of criteria that could be used by individual countries for closing-out specific research activities and issues.

8.4 Advanced nuclear reactor safety issues and research needs

A workshop on advanced nuclear reactor safety issues and research needs was held from 18-20 February 2002, attended by approximately 80 specialists from 18 countries. The objectives of the meeting were to facilitate early identification of research needs, to promote the preservation of knowledge, and to provide input to the Generation IV International Forum Technology Roadmap. Twenty-six papers were presented, and the proceedings were published by the NEA [5].

The discussion and conclusions of the workshop are summarised below. The programme was structured into four main sessions:

- Session 1 focused on basic safety principles and requirements, and on the implementation of defence-in-depth. It also provided an introduction to major novel designs and their safety cases.

- Session 2 was devoted to the identification of issues important for nuclear safety and their assessment.

- Session 3 discussed how to deal with these safety issues and identified questions and concerns, stressing those that could be solved through research and identifying research needs.

- Session 4 developed conclusions and recommendations regarding safety issues and research needs.

The workshop noted that the basic principle of nuclear safety, defence-in-depth, continued to be employed in advanced reactor designs. However, it was also recognised that advanced reactors posed several questions and challenges for implementation of this principle. In earlier designs, defence-in-depth was achieved primarily through deterministic implementation of provisions and multiple physical barriers against the release of fission products, and by measures to prevent accidents and mitigate their consequences. The various advanced reactor concepts placed different levels of emphasis on prevention and mitigation. The approach to the safety of future reactors would need to be derived from a more advanced interpretation of defence-in-depth, fully integrated with PSA insights. How to achieve the best integration of the deterministic and probabilistic concepts was still a major open question.

The advanced reactor concepts discussed during the workshop were mostly limited to advanced LWRs (ALWRs), high-temperature gas-cooled reactors (HTGRs) and liquid metal-cooled reactors (LMRs). The concepts discussed could be divided roughly into two categories: mature designs which were more or less ready for the market, such as FRAMATOME ANP's SWR-1000 or Westinghouse's AP-600; and preliminary designs, such as IRIS (an ALWR) and most LMR and HTGR designs. A common feature to all the advanced reactor types was that they promised safety enhancement over the current generation of plants. Consequently, their safety significance and the provisions to be made against external hazards were common questions that pertained to all the designs.

The mature ALWR concepts were characterised by increased simplicity and streamlining in the safety system design, a significant number of passive (system) features, and explicit consideration of severe accidents as part of the design basis. Regarding severe accidents, the technical and regulatory treatment varied between Europe and United States. It was noted that European vendors and regulators specifically required qualification of the dependability of severe accident capabilities (although design features were also selected on the basis of PSAs, to effectively eliminate severe accident sequences that would be overly complex to manage). In the USA, however, PSAs were relied upon more extensively to identify severe accident vulnerabilities and appropriate measures to reduce the risk of such accidents.

As for LMRs, it was noted that a considerable experience base from operating sodium-cooled reactors existed, and convergence seemed to be occurring in the treatment of certain major issues such as core disruptive accidents and sodium-related issues. As far as lead/bismuth-cooled reactors were concerned, significant questions related to, among other things, materials and thermal-hydraulics issues (integrity, corrosion, thermal loads and heat transfer, irradiation effects, etc.) remain. However, it was also noted that considerable operating experience (about 80 reactor-years) had been gained with Russian submarines using the same type of coolant. Several research institutions in NEA member countries were building research facilities to intensify experimental and analytical heavy liquid metal (HLM) investigations.

Some actual operating experience also existed for HTGRs, and the main issues for future designs were clear. HTGR safety cases, as presented so far, relied very heavily on the fuel as the main (if not the sole) fission product barrier, and hence fuel issues had become prominent. These included fuel (concept) qualification with a very high confidence level, manufacturing issues, fuel handling during operation, and improved understanding of fuel failure mechanisms and modes. HTGR designs included promising features regarding both criticality and decay heat removal concerns, but ultimately their success would depend on the quality of the fuel. Also, certain well-known systemic safety issues, such as air-water ingress into the reactor and reactor vessel integrity with respect to thermal shock, remained to be addressed to an extent that would convince the entire reactor safety community.

Most of the fundamental research into currently proposed HTGR design features seemed to date back 20-30 years, and there did not appear to be a significant recent experimental effort either to confirm earlier results or to close existing gaps. Proponents of HTGRs maintained that the plants would need no leak-tight containment in the conventional sense against internal threats due to greatly enhanced safety. However, recent attention to external hazards, and the fact that the relative importance of external hazards would increase with enhanced safety against internal hazards, might raise the question of the need for a containment or other protection against external hazards.

It was concluded that identification of specific research needs for any reactor type could only follow the establishment of a consistent overall safety case. Such a safety case would help identify where the remaining research needs were, and what level of uncertainty reduction or confidence was necessary. Ideally, the safety case should render manageable all confidence-related requirements of each individual safety question and safety factor. Only then could research problems be formulated properly, i.e. defined so that the problem will have a definite solution knowable to adequate accuracy and obtainable at reasonable cost. Research supporting the development of passive systems for mature ALWR concepts seemed to fulfil this objective (or to come close).

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Chapter 9

NDC ACTIVITIES RELATED TO R&D NEEDS

The role of the NEA Committee for Technical and Economic Studies on Nuclear Energy and the Fuel Cycle, known as the Nuclear Development Committee (NDC), is to provide authoritative, reliable information on nuclear technologies, economics, strategies and resources to governments for use in policy analyses and decision making. This includes information on the future role of nuclear energy within the context of energy policies that contribute to sustainable development.

The programme of work contains a wide range of studies on the economics and technology of nuclear power, and seminars are organised for the exchange and consolidation of information. Studies on partitioning and transmutation (P&T) were initiated in 1988. Mainly feasibility studies or state-of-the-art reports have been produced. Most of the P&T studies have been performed in close co-operation with the Nuclear Science Committee (NSC). The NDC has also conducted some studies on new reactor concepts. These have been performed in co-operation with the International Energy Agency (IEA) and the IAEA. In the future, the importance of new reactor concepts in the NDC's programme of work may increase, and new working methods, such as co-ordination of R&D projects, might be initiated.

The most recent P&T study, *Accelerator-driven Systems and Fast Reactors in Advanced Nuclear Fuel Cycles: A Comparative Study* (published in 2002), reviewed the potential of these two technologies as part of a series of studies on fuel cycles options. Some recommendations on future priorities in nuclear development of a general nature were presented in the report. However, no particular recommendations of R&D needs were included. The NDC has organised, in co-operation with the NSC, a series of International Information Exchange Meetings since 1989. The purpose of these meetings has been to enhance the value of basic P&T research by facilitating the exchange of information and discussions of programmes, experimental procedures and results. The latest meeting took place in October 2002.

The so-called "three-agency study" (undertaken jointly by the IEA, the IAEA and the NEA) entitled *Innovative Nuclear Reactor Development: Opportunities for International Co-operation* (published in 2002), suggested the following topics for collaborative research and development:

- Natural circulation.
- High-temperature materials.
- Passive (safety) devices.
- In-service inspection and maintenance methods.
- Advanced monitoring and control technologies.
- Delivery and construction methods.
- Safeguards technologies and approaches.

Additionally, the following topics were suggested for more limited co-operation among a few design groups:

- Advanced coolants.
- Advanced fuel design, processing and fabrication.
- Sub-critical systems.
- Component development.

The NDC has conducted some studies on infrastructure for education, training and R&D. A study entitled “International Collaboration to Achieve Nuclear Support Excellence” was started in early 2002. The aim of this study is to survey R&D capabilities, to map recent initiatives on education and training, and to identify best practices on international collaboration. The survey of R&D capabilities is being performed in co-operation with the NSC.

The NDC programme of work for 2003-04 contains a project called “Impact of Advanced Nuclear Fuel Cycle Options on Waste Management Policies”. The intention of this study is to evaluate the amounts and types of wastes generated in various fuel cycle options, and to identify the impacts of these on waste management and disposal. During the project some requirements for further R&D may be identified. Since experts on the fuel cycle, P&T and waste disposal are expected to participate in the project, it will be conducted in collaboration with the NSC and the NEA Radioactive Waste Management Committee (RWMC), as well as with the IAEA and the European Union.

R&D NEEDS FOR NEW NUCLEAR SYSTEMS

Chapter 10

OVERVIEW OF CEA STUDIES ON FUTURE NUCLEAR ENERGY SYSTEMS

Future nuclear energy systems studies conducted by the French Commissariat à l'Énergie Atomique (CEA) aim at investigating and developing promising technologies for future reactors, fuels and fuel cycles, to enable nuclear power to play a major part in sustainable energy policies. Reactors and fuel cycles are considered as integral parts of a nuclear system which should be optimised as a whole. The major goals assigned to future nuclear energy systems are:

- Reinforced economic competitiveness with other means of electricity generation, with a special emphasis on reducing the investment cost.

- Enhanced reliability and safety, through improved management of reactor operation in normal and abnormal plant conditions.

- Minimised production of long-lived radioactive waste.

- Resource-use efficiency, through effective and flexible use of the available resources of fissile and fertile materials.

- Enhanced resistance to proliferation risks.

The latter three goals are essential for the sustainability of nuclear energy in the long term. Additional considerations, such as the potential for applications other than electricity generation (e.g. co-generation, production of hydrogen, seawater desalination, etc.), also take on an increased importance.

Sustainability goals call for fast neutron spectra (to transmute nuclear waste and to breed fertile fuel) and for the recycling of actinides from spent fuel (plutonium and minor actinides). New applications and the need for increased economic competitiveness call for high-temperature technologies (around 850°C), that afford high conversion efficiencies and hence less radioactive waste production and discharged heat.

These considerations indicate a need for technological breakthroughs beyond light water reactors. Therefore, as a result of a screening review of candidate technologies, the CEA has selected an innovative concept of a high-temperature gas-cooled reactor (HTR) with a fast neutron spectrum, robust refractory fuel, direct conversion with a gas turbine and an integrated on-site fuel cycle, as a promising system for a sustainable energy development. This concept partly builds on past experience of the CEA with gas-cooled reactors (gas-graphite and HTRs), and partly on current efforts to revive and update HTR technologies to support the development of modular helium-cooled reactors of about 300 MWe by FRAMATOME ANP and their international partners.

In this context, the CEA decided to focus prospective R&D work on the development of a consistent set of gas-cooled nuclear systems. These systems will range from medium-term reactor projects for electricity generation and other applications (envisaged as export models, allowing for process heat production, hydrogen production at very high temperatures, plutonium burning, etc.) to a longer-term vision of sustainable nuclear systems using fast neutrons with a closed and integrated fuel cycle. This range of gas-cooled nuclear systems will cover a wide variety of high-temperature applications, as well as a broad range of fuel cycles, including synergistic fuel cycles with LWRs (i.e. burning plutonium and possibly also minor actinides from PWR spent fuel).

A specific R&D programme is being implemented to support the development of this consistent set of gas-cooled systems. The major emphases of this programme are on fuel particle re-fabrication and possible adaptation to fast neutrons, high-temperature materials, high-temperature systems technology, and compact spent fuel processing and re-fabrication processes. This work anticipates the construction of large experimental facilities during the next decade, such as an integral test loop by 2008 and a technology-testing reactor by 2012. A substantial effort is also being invested in the validation of computational tools and procedures for feasibility and performance studies. Strong connections with fundamental research work in nuclear physics, materials science and nuclear chemistry will be essential to improve modelling capability and to achieve effective breakthroughs for the development of high-temperature and high-irradiation damage-resistant materials.

The potential of high-temperature modular reactors for massive production of hydrogen – as a future clean energy carrier – without the release of greenhouse gases seems promising enough for the CEA to join the international effort to assess the feasibility and performance of thermo-chemical water-splitting processes. The potential of these reactors to also co-generate electricity and to desalinate sea water with reverse osmosis (at around 40°C) or multiple-effects distillation processes (at around 120°C) is also being assessed, to meet the needs of the international market.

Future nuclear energy systems studies are the subject of active international exchanges, including bilateral collaborations with the United States, Japan and Russia, as well as participation in European technology networks and contributions to the integrated R&D projects of the European Commission's Framework Programme. The CEA and its industrial partners are promoting their ambitious vision of future nuclear energy systems within the Generation IV International Forum which has been set up by the US Department of Energy to screen promising technologies for future international development.

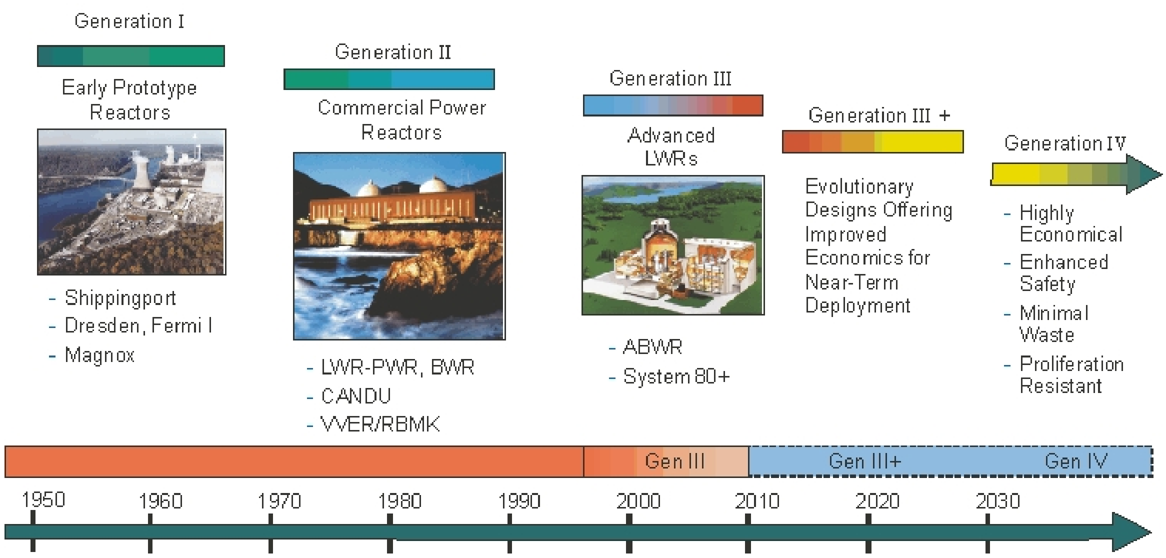
Chapter 11

GENERATION IV NUCLEAR ENERGY SYSTEMS TECHNOLOGY ROADMAP

Many of the world's nations, both industrialised and developing, believe that the greater use of nuclear energy will be required to meet their future energy and environmental quality needs. To enable nuclear energy to meet this expanded world-wide role, ten countries – Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, Switzerland, the United Kingdom and the United States – have been co-operating to advance a new generation of nuclear energy systems. These systems, known as Generation IV, are intended to be deployable no later than 2030 in developed and developing countries for the generation of electricity and other products, such as hydrogen for use as a transportation fuel and fresh water for world regions facing future shortages.

Figure 11.1 gives an overview of the generations of nuclear energy systems. The first generation was operated in the 1950s and 1960s as early prototype reactors. The second generation began in the 1970s with the large commercial power plants that are still operating today. Generation III was developed in the 1990s, comprising a number of evolutionary designs that offer significant advances in safety and economics. A number of reactors has been built to these designs, primarily in East Asia. Further advances to Generation III designs are underway, resulting in several so-called Generation III+ near-term deployable designs that are actively under development and are being considered by several countries. New plants built between now and 2030 will likely be chosen from these designs. Beyond 2030, the prospect for innovative advances through renewed R&D has stimulated interest world-wide in a fourth generation of nuclear energy systems.

Figure 11.1. Overview of the generations of nuclear energy systems



The ten countries mentioned above joined together to form the Generation IV International Forum (GIF) to develop future-generation nuclear energy systems capable of being licensed, constructed and operated in a manner that will provide competitively-priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation and public perception concerns. The objective for Generation IV nuclear energy systems is to have them available for international deployment by about 2030, when many of the world's currently operating nuclear power plants will be at or near the end of their operating licences.

Nuclear energy research programmes around the world had developed numerous concepts that could form the basis for Generation IV systems. Beginning in 2000, the countries constituting the GIF began meeting to discuss the research necessary to support next-generation reactors. Following those initial meetings, work began to define a "technology roadmap" to guide the Generation IV effort. Roadmapping is a methodology used to define, gain consensus for, and manage the planning and execution of large-scale R&D efforts. The GIF agreed to support the preparation of a roadmap, which became the focal point of its efforts. More than 100 technical experts from 10 countries contributed to its preparation.

Preparing the Generation IV Technology Roadmap required the establishment of goals for Generation IV systems. These goals had three purposes:

- To serve as the basis for developing criteria to assess and compare candidate systems.
- To be challenging and so to stimulate the search for innovative nuclear energy systems (both fuel cycles and reactor technologies).
- To motivate and guide R&D on Generation IV systems as collaborative efforts got underway.

The Generation IV goals were defined in four broad areas:

- Sustainability goals, focusing on fuel utilisation and waste management.
- Economics goals, focusing on competitive life cycle and energy production costs, and on financial risk.
- Safety and reliability goals, focusing on safe and reliable operation, accident avoidance and minimisation of consequences, investment protection, and elimination of the technical need for off-site emergency response.
- Proliferation resistance and physical protection goals, focusing on safeguarding nuclear materials and facilities.

Six systems were selected by the GIF as meeting the specified technology goals for Generation IV. Each of these systems comprises a nuclear reactor and its energy conversion systems, as well as the necessary facilities for the entire fuel cycle, from ore extraction to final waste disposal. The systems selected for Generation IV by the GIF are listed alphabetically in Table 11.1.

Table 11.1. Generation IV systems selected by GIF

Generation IV system	Acronym
Gas-cooled fast reactor system	GFR
Lead-cooled fast reactor system	LFR
Molten salt reactor system	MSR
Sodium-cooled fast reactor system	SFR
Supercritical water-cooled reactor system	SCWR
Very high temperature reactor system	VHTR

The motivation for the selection of these six systems was to:

- Achieve significant advances toward the technology goals.
- Ensure that the important missions of electricity generation, hydrogen and process heat production, and actinide management were adequately addressed by Generation IV systems.
- Provide some overlapping coverage of capabilities, because not all of the systems may ultimately be viable or attain their performance objectives and attract commercial deployment.
- Accommodate the range of national priorities and interests of the GIF countries.

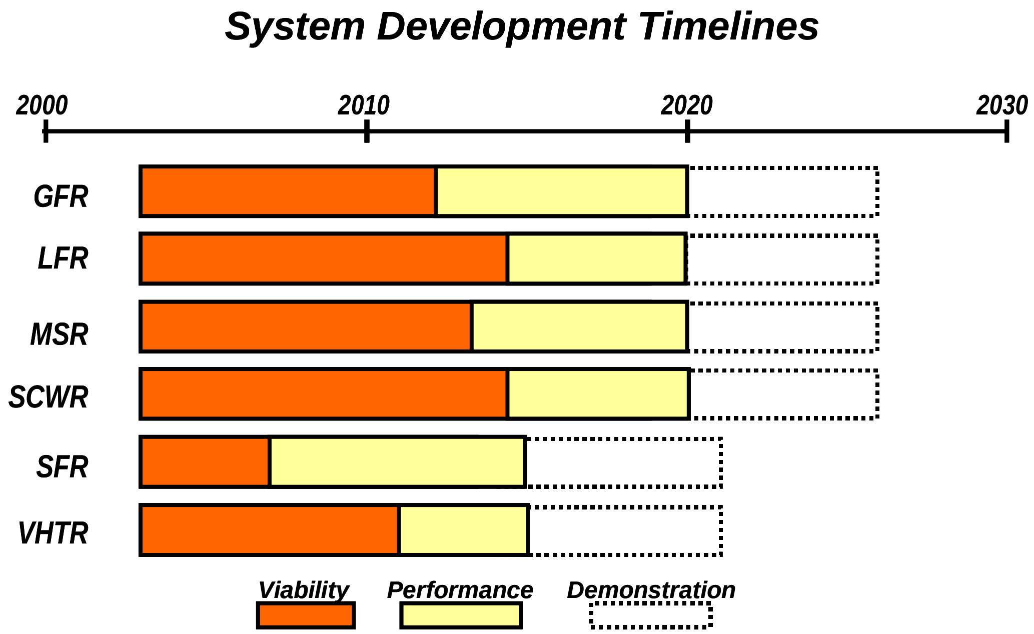
The Generation IV Technology Roadmap describes the system evaluation and selection process, introduces the six Generation IV systems chosen by the GIF, summarises R&D activities and priorities for the systems, surveys system-specific R&D needs for all six systems and collects cross-cutting R&D in areas of common interest to several or all systems. Cross-cutting R&D is defined in the areas of fuel cycle, fuels and materials, energy products, risk and safety, economics, and evaluation methodologies.

The scope of the R&D described in the roadmap covers all the Generation IV systems. However, with six Generation IV systems and 10 countries in the GIF, the approach to building integrated programmes for any of the systems becomes an important issue. GIF countries have expressed strong interest in collaborative R&D on Generation IV systems. Nevertheless, it has been acknowledged that each country will participate in R&D only on the systems that it chooses to advance.

In light of the considerable resources required for the development of any Generation IV system (roughly US$1 billion each), not all six systems are likely to be chosen for collaborative efforts. For systems that are chosen, participating countries will need to assemble the priority R&D for the system and the necessary cross-cutting R&D, and then set the desired pace for the programme. The technology roadmap has been structured to allow the independent establishment of collaborative R&D programmes. Thus, the roadmap provides a foundation for formulating national and international programme plans on which the GIF countries will collaborate to advance Generation IV systems.

With regard to the timing of programmes, Figure 11.2 shows a summary of the expected duration of the R&D activities for each of the systems. This R&D is organised into two phases, viability and performance. Viability R&D is designed to answer basic feasibility and proof-of-principle questions, while performance R&D will undertake engineering-scale development and optimisation to achieve desired levels of performance.

Figure 11.2. Timelines for the development of Generation IV systems



It can be seen that the systems are not projected to complete their viability and performance phases at the same time. For each of the systems chosen for collaboration, the GIF will need to periodically assess the success of the R&D and the progress toward achieving the Generation IV goals. The technology roadmap includes R&D on evaluation methodology that will support these continuing assessments. Following successful completion of the viability and performance R&D for a particular system, at least a further six years and several billion US dollars will be required for the detailed design and construction of a demonstration system.

The GIF plans to focus its future meetings on the development of collaborative programmes on several systems. Of considerable interest is the participation of industry in the Generation IV programme, and its growth as the systems advance. While the prospects for demonstration and entry into commercial markets are a number of years into the future, the need exists for early involvement of industry to provide direction and to keep the focus on the requirements for the systems.

Interactions with regulatory authorities are also planned as system development progresses to enable regulators to understand the system design features and technologies and to provide feedback on licensing issues. To enhance public confidence, the findings of the roadmap and the R&D plans developed based on it will be communicated to the public on a continuing basis. Mechanisms for communicating with interested stakeholder groups will be developed so that their views on the programme are considered and, to the extent possible, incorporated into the objectives of the R&D programmes.

Chapter 12

AN INTRODUCTION TO INPRO AND ITS STATUS

In 2000 the International Atomic Energy Agency (IAEA) initiated the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) following a resolution of the IAEA General Conference [GC(44)/RES/21]. The main objectives of the INPRO project are:

- To help to ensure that nuclear energy is available to contribute to fulfilling energy needs in the twenty-first century in a sustainable manner.
- To bring together technology holders and technology users to jointly consider the international and national actions required to achieve the desired innovations in nuclear reactors and fuel cycles.

As of October 2002, INPRO participants comprised the European Commission and twelve IAEA member states: Argentina, Brazil, Canada, China, Germany, India, the Republic of Korea, the Russian Federation, Spain, Switzerland, the Netherlands and Turkey. These participants have nominated in total 17 experts to INPRO, with assignments ranging from three months to three years.

Phase I of INPRO started in May 2001 and is split into two sub-phases:

- Phase I-A will deal with determining user requirements and developing a suitable methodology to assess the level of fulfilment of these criteria by different future nuclear technologies. For this phase, task groups have been established by the IAEA in order to define user requirements in the following areas:

  - Economics (including resources and demand).
  - The environment, fuel cycle and waste.
  - Safety.
  - Non-proliferation.
  - Cross-cutting issues.

  A sixth task group will define the methodology for assessment of nuclear technologies.

- Phase I-B will apply the assessment methodology in several IAEA member states in order to make a judgement on the potential of innovative nuclear technologies.

This report presents an overview of the preliminary results of Phase-IA. The main initial results of the different task groups are summarised below.

Four representative future scenarios have been selected as the basis for the INPRO project out of the 40 reference scenarios in the *Special Report on Emission Scenarios of the Intergovernmental Panel on Climate Change* (IPCC). The chosen scenarios are characterised by differences in terms of globalisation versus regionalisation, and in the demographic, social, economic, technological and environmental developments which are foreseen.

Some general and special requirements have been defined within the Task Group on Requirements for the Environment, Fuel Cycle and Waste. In the area of nuclear safety, five general and up to 30 special requirements have been created, covering the nuclear reactor and whole fuel cycle including the mining of fissile and fertile material, fuel fabrication, operation of reactors, and the treatment of spent fuel and waste.

One example of a general safety-related requirement for future nuclear systems is that future nuclear reactors and fuel cycle installations shall be so safe that they can be sited in locations similar to other industrial facilities.

In the area of non-proliferation, a definition of principles is to be finalised which will equally cover the intrinsic and extrinsic features of proliferation resistance. The cross-cutting issues considered within INPRO are the changes necessary in existing nuclear power infrastructure in order to enable the implementation of innovative nuclear technologies as foreseen in future scenarios. The INPRO assessment methodology chosen will adopt a top-down approach, and some illustrative examples of its application are available. A first draft report was due to be presented to the INPRO Steering Committee in December 2002.

Chapter 13

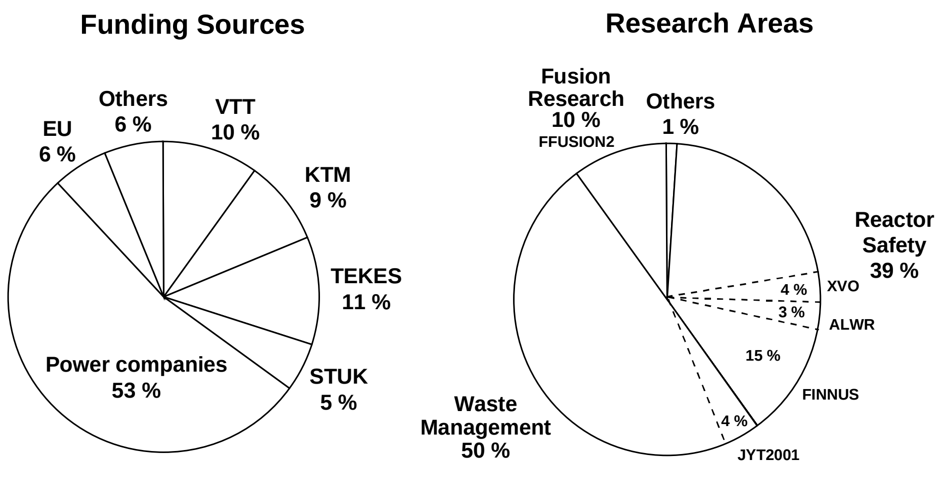
NUCLEAR ENERGY R&D IN FINLAND

A continuous high level of safety is a prerequisite for the use of nuclear energy. About 27% of electricity generated in Finland is nuclear; in 2002 the government made a positive decision-in-principle, which was approved by parliament, concerning the construction of a new nuclear power unit. In 2001, a positive decision-in-principle was made on building a final disposal facility for spent nuclear fuel at Olkiluoto. These decisions mean that the construction of the new unit and the final disposal facility are considered to be in line with the overall good of the country. High levels of operational reliability and careful upgrading have maintained nuclear production costs at a competitive level as the electricity market has been opened.

Maintaining confidence in nuclear safety calls for continuous investment in plant operation and supervision. This needs to be supported by well-directed research in various fields of technology and human behaviour in order to take into account modernisation and upgrading of plant processes, implementation of new techniques, changing production goals and updated safety requirements. Research is also needed to deepen understanding of the new technology needed in the construction and licensing phases of new units.

The total of nuclear-energy-related R&D efforts in Finland in 1999 was about €27 million (see Figure 13.1). No dramatic changes have occurred since then. The power companies TVO and Fortum directly fund more than half of the total, and the public sector about one-third. Half of the total is spent on nuclear waste management issues, mainly conducted by the power companies. Nearly 40% is used for reactor safety, of which about half is allocated for fully or partly public research programmes. Public sector funding comes from the Ministry of Trade and Industry (KTM), the Technical Research Centre of Finland (VTT), the National Technology Agency (Tekes) and the Radiation and Nuclear Safety Authority (STUK).

Figure 13.1. Resources for nuclear energy R&D in Finland in 1999 (total €27 million)



A country using nuclear energy needs to possess a sufficient infrastructure to support the required education and research in this field, as well as the operating and supervisory organisations for the plants. Public sector nuclear safety research programmes must ensure that the knowledge needed for continuing safe and economic use of nuclear power is retained, that new expertise is developed, and that the country can participate in international projects. In fact, the Finnish organisations engaged in research in this sector are an important resource which the various ministries, STUK and the power companies have at their disposal.

The largest of the public research programmes has been FINNUS, which concentrated on nuclear reactor safety-related issues in existing power plants. The advanced LWR (ALWR) programme was set up to look at possible future nuclear power plant designs. The plant life management programme (XVO) was directed towards plant-specific ageing problems, with particular support from the power companies. JYT2001 was the public sector nuclear waste management programme.

These programmes were mainly conducted at the various research units of the Technical Research Centre of Finland (VTT). Universities also contributed. Similar programmes are continuing – in 2002 the national research programme for waste management (KYT) and the RKK project on plant ageing were started. KTM has also commissioned a long-term plan for the next national nuclear safety programme for the period 2003-06, called SAFIR (Safety of Nuclear Power Plants – Finnish National Research Programme).

The initial decision-in-principle on the final disposal facility for spent fuel was valid for the spent fuel from existing Finnish nuclear plants; an additional decision-in-principle covered extension of the facility for spent fuel from the planned new unit. The total amount of spent fuel that the facility will be able to hold is a maximum of 6 500 tonnes of uranium, beginning from 2020. This decision makes it possible for Posiva, the joint waste management company of the power companies, to focus on confirmatory bedrock investigations at Olkiluoto. An underground rock characterisation facility, ONKALO, will be constructed at the site. According to the plans, the construction of ONKALO will start in 2003-04, and investigations at final disposal depth will commence around 2006.

In the KYT national nuclear waste management programme (2002-05), strategic studies as well as long-term safety studies of spent fuel disposal will be conducted. Fundamental options (including partition and transmutation) and general safety principles in the fuel cycle and waste management, as well as the costs of managing different levels of waste, transportation and decommissioning, will be studied. Projects on the release of radionuclides from the repository, on their migration in bedrock and groundwater, on safety assessment methodologies and on biosphere behaviour will be included.

The research objectives of the FINNUS programme (1999-2002), aimed at ensuring integrity, safety and reliability, were classified under three themes: ageing, accidents and risks. The effects of ageing on nuclear power plants were studied intensively in order to evaluate the safe remaining lifetime of components and the efficiency of corrective measures. The programme mainly concentrated on studies of the effects of ageing on material properties, degradation mechanisms in metallic structures, structural integrity and in-service inspection. It also covered monitoring methods, including the new area of reinforced concrete structures.

The accident theme concerned the operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal-hydraulics and severe accidents were addressed by both computational and experimental studies. In the risk field, attention was paid to advanced risk analysis methods and their applicability, and to the evaluation of fire risks, safety critical applications of software-based technology, and human and organisational performance. VTT and Lappeenranta University of Technology (LTKK) conducted the FINNUS research.

In order to make the best use of limited public resources, the expertise of the power companies is exploited in the steering and reference groups of the public sector programmes. The results of public research are also at the disposal of the power companies. There were 11 research projects in FINNUS, with many natural connections between them. One objective of the programme was to strengthen these links and create new ones. Being the nuclear community of a small country was an advantage when creating such novel couplings between different research fields. The interdisciplinary approach of the projects extended the know-how of both the research scientists and the external experts participating in the work of the reference groups. The research programme also contributed to the education of new experts – six doctoral, two licentiate and 18 master's theses were completed.

The ALWR technology programme (1998-2003) concentrated on those reactors which could be built within 5-10 years. Participating and sponsoring organisations included VTT, Tekes, STUK, the power companies and the universities LTKK and HUT. In addition, FRAMATOME ANP, General Electric, Westinghouse Atom and the European Commission contributed. The objectives were to increase understanding of next-generation nuclear reactors so that Finnish organisations could evaluate their economics and safety on an independent and solid technical basis, and to gather experience and to develop tools for effective design and safety assessment. Research within the programme concentrated on the safety aspects. There was close co-operation between experimental research and modelling. New experimental capabilities were constructed, thermal-hydraulic codes were further developed, and wide practical experience was gained on the use of computational fluid dynamics tools for design and safety analyses. Most of the reactors studied are candidates for the new Finnish unit, and the tools and capabilities developed will be well used during its evaluation.

The research plan of the new programme SAFIR covers the period 2003-06, but it is based on safety challenges identified for a longer time span. The favourable decision-in-principle on a new nuclear power plant has also been taken into account in the plan. However, the safety challenges posed by the existing plants and by the new plant, as well as the ensuing research needs, do converge to a great extent. The construction of the new unit will increase the need for experts in the field in Finland. Together with the retirement of existing experts, this calls for more education and training, in which active research activities play a key role. This situation also poses a great challenge for long-term safety research.

The general plan of SAFIR aims to define important research needs related to the main safety challenges (such as ageing of the existing plants), to developments in the various areas of technology (e.g. fuel, safety functions, automation, IT) and to organisational changes. The research into these needs is the programme's main technical/scientific task. The programme will also aim to preserve know-how in those areas where no significant changes are occurring but in which dynamic research activities are an absolute precondition for safe use of nuclear power. The general plan lays the ground for drafting more detailed annual plans and the research proposals needed for these. The programme has the flexibility to include short-term specific projects, projects running through the entire lifetime of the programme and development work of very long duration.

All these programmes require national resources to be assembled and efficiently controlled. The research programmes must function as efficient conveyors of information to all organisations operating in the nuclear energy sector, and as open discussion forums for participation in international projects, allocation of resources and planning of new projects.

International co-operation has been vital in all the fields of research in Finland. In addition to publishing results in various international forums, Finnish research staff has contributed to working groups and networks, to the definition and solution international benchmarks, and to round-robin exercises. The most important contact organisations have been the NEA, the IAEA, the European Commission and Nordic Nuclear Safety Research (NKS).

Chapter 14

ADVANCED FUEL CYCLE FIVE-YEAR PROGRAMME

Of the issues that must be addressed to enable a future expansion in the use of nuclear energy, in the United States and elsewhere, none is more important or more challenging than that of dealing effectively with spent nuclear fuel. While nuclear plants produce far less waste by volume than any comparable energy-producing or industrial activity, the unique nature of spent fuel requires that long-term planning for the use of nuclear power include consideration of its eventual disposition. Spent fuel is radioactive for hundreds of thousands of years, which presents a wide range of social, political, regulatory and technical issues. With the once-through fuel cycle, the potential growth of nuclear power in the USA has been limited.

Since the late 1990s, the US Department of Energy (DOE), its laboratories, and university and industry partners have worked with the international research community to explore the potential of advanced nuclear technologies for dramatically reducing the difficulty of disposing of spent fuel from nuclear power plants, in terms of both volume and toxicity. The Advanced Fuel Cycle (AFC) programme mission is to enable the future of nuclear power by addressing the spent nuclear fuel problem. The development of advanced nuclear power is essential to address energy security, environmental and economic concerns, as the once-through fuel cycle is recognised as limiting the growth of nuclear power in the USA and other countries to address these concerns.

The primary goals of the AFC programme are to:

- Develop and implement advanced fuel cycle technologies to significantly reduce the long-term cost of geological disposal of commercial spent nuclear fuel.

- Develop methods to reclaim the energy value from highly toxic spent fuel, while providing for its destruction.

The AFC programme is a multi-phase project consisting of two components executed in parallel as part of an integrated research effort:

- *Series One*. This component addresses the medium-term issues associated with spent nuclear fuel, primarily by reducing the volume and heat generation of material requiring geologic disposal. This will optimise utilisation of the first US repository, reducing or eliminating the need for additional repositories. It includes creating proliferation-resistant processes and fuels to enable the destruction of significant quantities of plutonium in LWRs or high-temperature gas-cooled reactors (HTGRs) by the middle of the next decade.

- *Series Two*. This component addresses the longer-term issues associated with spent nuclear fuel. Specifically, this effort will develop fuel cycle technologies to greatly reduce the long-term radiotoxicity and heat load of high-level waste sent for geologic disposal through support for the development of Generation IV reactors and possible accelerator systems.

The Series One and Series Two activities are managed as part of an integrated programme. For example, waste treatment technologies emerging from Series One may prove to be invaluable front-end steps for the more advanced processes targeted in Series Two. Integration of these two components is essential to the success of the overall effort.

The main priority of the AFC programme is to provide timely resolution of issues related to spent nuclear fuel disposition, including reducing the cost of the proposed first US repository and delaying or eliminating the need for a second repository. This will be achieved through separating long-lived highly toxic elements, reducing high-level waste volumes and the toxicity of spent nuclear fuel and reducing long-term heat generation. A second priority is to address energy and economic security issues by enabling the proliferation-resistant recovery of the energy contained in spent fuel and supporting the future operation of Generation IV nuclear energy systems. Finally, the programme will address non-proliferation concerns by reducing long-term inventories of plutonium in spent fuel and thus reducing its proliferation threat.

The DOE's Office of Nuclear Energy, Science and Technology (NE) is responsible for leading the US Government's investment in nuclear science and technology. The AFC programme is closely linked to another NE programme, the Generation IV Nuclear Energy Systems programme. This is an international initiative to identify, develop and demonstrate one or more new nuclear energy systems offering advantages in the areas of economics, safety and reliability, and sustainability, with a deployment target of 2030. By integrating the AFC and Generation IV programmes through the use of systems analysis, NE has established a structure which will facilitate the co-ordination of both programmes in order to create a unified R&D effort. Within this structure, the AFC programme has been organised to maximise and leverage technical functional expertise, while enhancing communication between programme participants through systems analysis and technical integration.

The AFC and Generation IV programmes have an integrated management structure, sharing a common systems integration and analysis function. Roles and responsibilities for key AFC programme functions are shared among the headquarters organisations of NE (Technical Integration, Systems Integration and Analysis, and National Technical Directors) for each of the three elements of the AFC programme: fuels, separation and transmutation. Product teams are established, as needed, to address cross-cutting issues throughout the functional programme elements of both programmes.

The Series One component of the AFC programme addresses specific medium-term issues facing nuclear power in the USA. These issues are:

- Reducing high-level waste volumes.
- Increasing the capacity of the planned geologic repository.
- Reducing the technical need for a second repository.
- Reducing long-term inventories of plutonium in spent nuclear fuel.
- Enabling recovery of the energy contained in spent nuclear fuel.

To achieve these goals, operation of a spent fuel treatment facility and a proliferation-resistant fuel fabrication facility will be initiated by 2015 and 2018, respectively. The technical programme for the first five years is structured to analyse and develop options for technology selections in 2007, as well as provide support to the facility design activities. After the technology selections, the technical programme will provide more focused support to the final design, construction, start-up and initial operation of the facilities. Series One transmutation activities for commercial plutonium and other minor actinides are considered an implicit part of the fuel development activity, and consequently no Series One transmutation activities are indicated.

The Series Two component of the AFC programme addresses specific long-term issues facing nuclear power. These are:

- Reducing the toxicity and long-lived activity of spent nuclear fuel.
- Reducing the long-term heat generation of spent nuclear fuel.
- Providing a sustainable fuel source for nuclear energy.
- Supporting the future operation of Generation IV nuclear energy systems.

To achieve these objectives, it will be necessary to deploy advanced reactors that use sustainable, proliferation-resistant fuel cycles. This will require a Generation IV fuel fabrication facility to be in operation by 2022 to provide fuels for the performance testing and deployment of Generation IV reactor systems. In addition, under Series Two an approach will be developed for the transmutation of un-burned plutonium and minor actinides from the Series One spent fuel treatment facility in Generation IV reactors and, possibly, accelerator-driven systems (ADS).

One element of Series Two transmutation activities will focus on ADS. Many countries are considering such systems as a viable approach to transmutation, because they may be capable of destroying all long-lived radioactive isotopes without creating plutonium. A technology selection for transmutation is planned for 2007. Regardless of the path forward chosen at that point, the information derived from the work on accelerator systems should facilitate more efficient and economic designs in later transmutation work.

Chapter 15

JAPANESE ACCELERATOR-DRIVEN SYSTEMS PROGRAMME

Since plutonium is to be used as a nuclear fuel material under the Japanese nuclear energy policy, partition and transmutation (P&T) activities in Japan are oriented to the separation and transmutation of minor actinides and long-lived fission products.

The OMEGA programme for R&D in P&T is a joint activity of the Japan Atomic Energy Research Institute (JAERI), the Japan Nuclear Cycle Development Institute (JNC) and the Central Research Institute of Electric Power Industry (CRIEPI). Its progress was reviewed in 1999 by the Atomic Energy Commission's Advisory Committee on Nuclear Fuel Cycle Back-end Policy. A report was issued in March 2000 highlighting the necessity of R&D work on future system design and development of implementation scenarios for P&T, basic experiments to demonstrate the feasibility of the processes and engineering-scale experiments to obtain safety data.

The goal of the OMEGA programme is to reduce the radiotoxicity of high-level waste to the level of natural uranium, and so reduce the burden for geological disposal and hence public acceptance. The R&D areas covered by the OMEGA programme are:

- Physical and chemical properties of minor actinides and fission products.
- Partitioning of radioactive elements in high-level waste from reprocessing.
- Transmutation, including nuclear and fuel property data for minor actinides, system design studies, reactor fuel and accelerator target development, development of high-power accelerators for transmutation.

Under the OMEGA programme, JAERI continues to develop partitioning processes and nitride fuel technology, and carry out basic studies to support ADS development. The reasons for developing ADS are as follows:

- An ADS would provide the flexibility to include nuclear waste transmutation in scenarios involving future nuclear reactor designs which are currently being considered. For example, in scenarios with a fuel cycle involving UO_2 in LWRs, MOX fuel in LWRs and reduced-moderation water reactors (RMWRs) or FBRs, an ADS could transmute minor actinides from LWR fuel, and Am and Cm from RMWR/FBR fuel. In such a scheme, the ADS would co-exist with RMWRs/FBRs.
- An ADS could provide a dedicated minor actinide transmutation system, independent from the commercial fuel cycle. P&T with an ADS would reduce the burden on the commercial fuel cycle in terms of economy and safety. Minor actinides would be confined to one small P&T site, separate from the commercial fuel cycle.

- An ADS consists of a sub-critical system with external neutron sources using a high-intensity proton accelerator. There would be sufficient control and flexibility in the design and operation to transmute large amounts of minor actinides in contract with critical reactors. As a result, an ADS would provide a highly efficient and safe transmutation system.

R&D at JAERI for P&T technology has been carried out on the basis of the double-strata fuel cycle concept. The main R&D items for ADS are shown in Figure 15.1. Recent technical achievements in analysis of mass flow and costs for a double-strata fuel cycle, in lead-bismuth technology, in partitioning and in fuel processes are summarised below.

Figure 15.1. R&D programme for accelerator-driven systems

Accelerator
- High intensity, high reliability and high efficiency

Reactor Physics
- Reliability of calculated multiplicity, power distribution, neutron spectrum, burn-up reactivity swing, etc.
- Monitoring of sub-criticality.
- Estimation of transmutation efficiency.

Operation of Sub-critical System
- Control method for reactivity and thermal output.
- Transient at start-up/shutdown and beam trip.

Target, Material, Operation of Pb-Bi system
- R&D for new material for Pb-Bi (window, structure)
- Handling technology of Pb-Bi (thermal-hydraulics, Pu recovery)
- Transient at start-up/shutdown and beam trip.

Fuel
- Fabrication and reprocessing of MA nitride fuel with ^{15}N

To ensure the long-term sustainability of the commercial fuel cycle, different reactors and fuel cycles are being considered. One such scheme involves an ADS co-existing in a plutonium-utilising fuel cycle with MOX fuel in LWRs and FBRs. The impact of P&T on geological disposal in such a scheme was studied; the electricity cost would be increased by 8% through introducing the P&T cycle. The ADS considered here was 800 MWth, able to transmute the minor actinides and ^{129}I produced each year from about 10 PWRs of 3 400 MWth.

For lead-bismuth technology, static corrosion tests were performed after 3 000 hours at 450°C. Loop corrosion tests were conducted twice at 3 000 hours under the following conditions: temperature 450°C, temperature difference 50°C, electro-magnetic pumping power 5 litres/minute, velocity at test section 1 m/s.

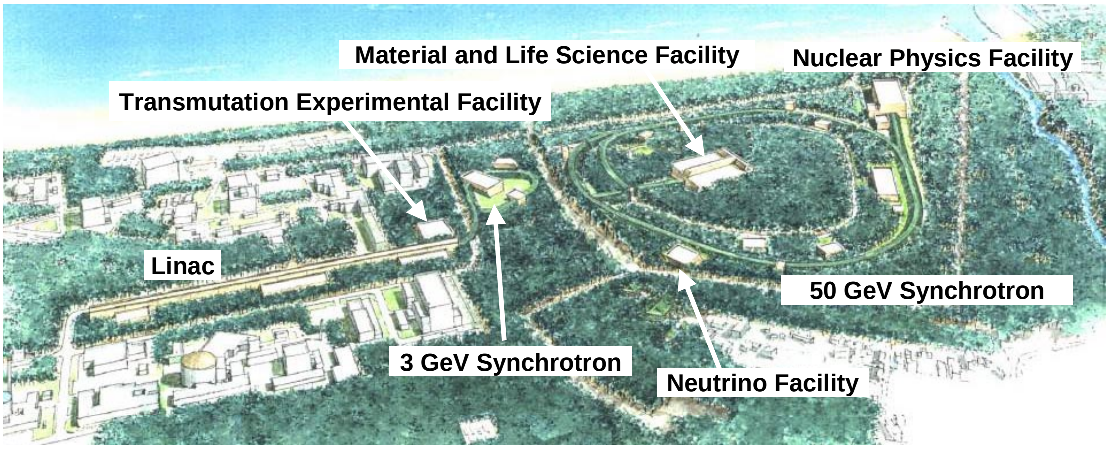
The four-group partitioning process which has been developed was tested with concentrated real high-level waste. The target elements Am and Cm were more than 99.98% separated, and Np more than 99.95%. A new extractant, TODGA, is being studied for the modification of the transuranics separation step.

Nitride fuel and pyroprocessing techniques are being developed. JAERI's ADS design uses (Pu,MA)N fuel diluted by inert matrices such as ZrN and TiN at initial loading. So far, (Pu,Zr)N and PuN+TiN pellets have been fabricated, and fuel pins containing the pellets are to be irradiated in the Japan Materials Testing Reactor (JMTR). Electro-refining of UN, NpN and PuN has been demonstrated. Similar work for AmN is planned from 2003 at the TRUHITEC facility at JAERI's Tokai site.

A P&T scenario with ADS transmutation is an innovative and flexible system for reducing high-level wastes, co-existing with plutonium-fuelled reactors such as MOX-LWRs, FBRs and resource-renewable BWRs (RBWRs). Following the basic R&D mentioned above, an experimental ADS with 80 MW thermal power is being considered for the late 2010s to demonstrate the engineering feasibility of the design. This will be operated with MOX fuel at first, gradually moving to minor actinide nitride fuel.

Prior to this, however, to study the basic characteristics of the ADS and to demonstrate its feasibility from the viewpoints of reactor physics and spallation target engineering, JAERI plans to build the Transmutation Experimental Facility (TEF) at its Tokai site in the framework of the High-intensity Proton Accelerator Project (see Figure 11.2). The budget for the construction of TEF has not yet been approved, but work is scheduled to start in 2006. The first phase of the whole project, the construction of the linear accelerator (linac), started in 2002.

Figure 15.2. Site plan of High-intensity Proton Accelerator Project



The TEF will consist of two buildings: the Transmutation Physics Experimental Facility (TEF-P) and the ADS Target Test Facility (TEF-T). TEF-P is a zero-power critical facility where a low-power proton beam accelerated up to 600 MeV is available to research the reactor physics and controllability of the ADS. TEF-T is a material irradiation facility which can accept a maximum 200 kW-600 MeV proton beam into the spallation target of Pb-Bi eutectic.

Some experimental research on the reactor physics aspects of sub-critical reactors has been carried out using existing facilities, such as MASURCA in France. There has, however, been no experiment to research and demonstrate a fast-neutron sub-critical system combined with a spallation source and a proton beam with appropriate energy. The main purpose of TEF-P is, therefore, to research the reactor physics of a sub-critical core driven by a spallation neutron source using a 600 MeV proton beam. The second purpose of TEF-P is to demonstrate the controllability of the sub-critical core; control of reactor power will be attempted by adjusting the power of the proton beam. The third purpose of TEF-P is to research the transmutation performance of the sub-critical core using certain amounts of minor actinides and long-lived fission products.

For the above-mentioned purposes, high thermal power is not necessary – a power level for critical experiments of about 100 W is optimal from the viewpoint of accessibility to the core. TEF-P is therefore designed with reference to JAERI's Fast Critical Assembly; a horizontal table-split type critical assembly with a rectangular lattice matrix.

TEF-T is a material irradiation facility with 200 kW proton beam of 600 MeV and a Pb-Bi eutectic target. To demonstrate the feasibility of the beam window, it will be necessary to coincide the proton beam density at the beam window with that of the future ADS. At TEF-T, the proton beam of 0.33 mA will be focused to 4 cm in diameter, resulting in a beam density of 0.026 mA/cm^2 on average; this is considered high enough for the demonstration. More than 10 DPA (displacement per atom) irradiation can be achieved per year. The neutron flux will exceed 10^{14} $n/cm^2/s$ at the centre of the target and 10^{13} $n/cm^2/s$ in the region of 30 cm in diameter and 30 cm in length where various materials can be irradiated by the fast neutrons.

The purpose of TEF-T is to demonstrate the feasibility of a high-power spallation target system and to research material compatibility in the Pb-Bi under irradiation. As well as the irradiated material test pieces, the irradiated structural material of the target vessel will be examined from the viewpoints of tensile strength, ductility, fatigue, etc. Parameters which can be varied for each irradiation include temperature, irradiation period, flow speed and oxygen concentration in the Pb-Bi. Valuable experience will be accumulated at TEF-T in operating such a system and handling the high-power spallation target.

Many countries are now interested in ADS as a possible option for dealing with back-end nuclear fuel cycle problems. As mentioned above, the TEF will be a unique facility dedicated to the development of an ADS. The experiments to be performed there will be extremely useful in realising an ADS, especially in making the step up to a demonstration ADS. JAERI hopes to proceed with this project as an international programme. At the present stage, it is discussing the specification and design of the facility with research organisations from other countries.

Chapter 16

EUROPEAN ADS PROGRAMME: DESIGNS AND PERSPECTIVES

In several European countries, fuel cycle strategy and closure of the back end are subject to national policy. There are various strategies under consideration for the future:

- UO_2 in LWRs + final disposal.
- UO_2 in LWRs + accelerator-driven system (ADS) + final disposal.
- UO_2 in LWRs + MOX in LWRs + ADS + final disposal.
- UO_2 in LWRs + MOX in LWRs + dirty MOX in FBRs + ADS + final disposal.

In nearly all scenarios, partitioning and transmutation (P&T) using an ADS is considered. This has led to the need for co-ordinated R&D on such systems.

Both critical and sub-critical reactors are potential candidates as dedicated transmutation systems. However, critical reactors, being heavily loaded with fuel containing large amounts of minor actinides (Am and Cm), pose safety problems caused by unfavourable reactivity coefficients and a small delayed neutron fraction. With regard to this latter problem, sub-criticality is particularly favourable, allowing the maximum load of minor actinides per unit while operating in a safe manner.

The European Technical Working Group (ETWG) on ADS under the chairmanship of Professor Carlo Rubbia has played a co-ordinating role at the European level for P&T and ADS development as a route for radioactive waste management. In its April 2001 report, the ETWG concluded that:

- P&T associated with an ADS could help the waste management problem.
- There was a need for a first step demonstration of an ADS at international level.
- There was a need for a co-ordinated R&D effort at European level, with strong support from the European Commission.

In response to these conclusions, the major European research centres, companies and universities active in the field of P&T and ADS development decided to create a thematic network called ADOPT (Advanced Options for P&T). The aim of ADOPT is to co-ordinate the various P&T projects under the European Commission's Fifth Framework Programme. Fifteen projects are co-ordinated under the ADOPT thematic network, representing a total European contribution of €28.6 million, or a share of about 50% of the total budget. Information on ADOPT can be obtained at www3.sckcen.be/adopt/.

Among these projects co-ordinated by ADOPT is the PDS-XADS (Preliminary Engineering Design Studies of the Experimental ADS) project. Its objectives are:

- To select the most promising technical concepts.
- To address the critical points of the whole ADS system.
- To identify the R&D needed in support of ADS development.
- To define the safety and licensing issues.
- To make a preliminary assessment of the cost of the installation.
- To consolidate the roadmap for development of the experimental ADS (XADS).

Taking into account that a fast neutron spectrum is the *a priori* solution for transmutation purposes, R&D efforts are focused on liquid-metal-cooled and gas-cooled ADS. The preliminary design studies are concentrated mainly on three concepts, namely:

- EA-80, a large Pb-Bi-cooled XADS proposed by Ansaldo (80 MWth core power, cooled by liquid Pb-Bi, driven by a 600 MeV × 10 mA proton beam current delivered by a linear accelerator or a cyclotron on a liquid Pb-Bi window or windowless spallation target).
- GC-XADS, a large gas-cooled XADS proposed by FRAMATOME ANP (100 MWth core power, cooled by He, driven by a 600 MeV × 10 mA proton beam current to be delivered by a linear accelerator on a liquid Pb-Bi window spallation target).
- MYRRHA, a small Pb-Bi-cooled XADS proposed by SCK•CEN (40 MWth core power, cooled by liquid Pb-Bi, driven by a 350 MeV × 5 mA proton beam current to be delivered by a cyclotron or a linear accelerator on a liquid Pb-Bi windowless spallation target).

For the Sixth Framework Programme, the European Commission created two new instruments to encourage the integration and reinforcement of large-scale European R&D, known as Integrated Project (IP) and Network of Excellence (NoE). ADOPT members expressed their wish to reinforce their activities in the field of P&T and ADS by creating an integrated project, IP-ADOPT.

The aim of this proposal is to mobilise European scientific and industrial expertise in nuclear fuel reprocessing, nuclear fuel development, nuclear reactor research and engineering design, and high-power proton accelerator R&D, to provide advanced options for high-level waste management, leading to a simplification of the conditions for geological disposal of such waste.

The IP-ADOPT project will allow the structuring and integration of European activities related to P&T and ADS development, for the following reasons:

- It will avoid the fragmentation of the present P&T and ADS community by focusing on the medium-term objective of realising an ADS demonstration facility, testing at a large scale the economic feasibility of the transmutation of minor actinides in a dedicated core.
- It will offer a stable research environment for this R&D community.
- It will help maintain national funding in this field at a reasonable level.
- It will support the objective of exploring new technologies which contribute to the solution of waste management problems, and will look to technologies that can be of use for next-generation reactors.

The R&D perspectives and requirements already proposed for IP-ADOPT are as follows:

- Partitioning:
  - Development of hydrometallurgical partitioning processes, possibly adapted to innovative fuels.
  - Demonstration of the feasibility of actinide recovery by pyrochemical processes, possibly with recycling of all actinides together.
- Advanced fuels:
  - Development of fuels and targets specifically devoted to transmutation, and achieving better understanding of their behaviour (by experimental irradiation and modelling).
  - Demonstration of fuel safety, licensing and reprocessing aspects.
- ADS design:
  - Accelerator development, in particular reliability qualification of prototypic components.
  - Proof of feasibility of both window and windowless spallation targets.
  - Development of Pb-Bi-cooled and/or gas-cooled sub-critical core designs, with testing of key components.
  - Establishment of the safety case for ADS.
  - Development of a siting and licensing approach for ADS.
  - Measurement and evaluation of nuclear data needed for achieving a reliable design.
- Materials and heavy liquid metal (HLM) coolant technology:
  - Qualification of promising structural materials under proton and neutron irradiation.
  - Exposure of such materials to Pb-Bi through the use of the MEGAPIE-Test, KALLA and CIRCE facilities.
  - Address issues such as: corrosion due to HLM; liquid metal embrittlement due to Pb-Bi; embrittlement due to irradiation in a mixed radiation (n,p) field; coatings for corrosion or mitigation of liquid metal erosion; establishing an engineering characterisation database.
  - Development of Pb-Bi technology as a core coolant or target material, including thermal-hydraulics data, and modelling and instrumentation of HLM.

CONCLUSIONS AND RECOMMENDATIONS

Following the presentations contained in Sections 2 and 3 of this report, the workshop participants divided into three discussion groups in order to produce conclusions and recommendations. These groups concentrated respectively on the topics of nuclear data, reactor physics and systems behaviour, and fuels, materials, coolants and chemistry. The recommendations put forward by each group are detailed in the chapters that follow.

Chapter 17

NUCLEAR DATA

17.1 Differential nuclear data

The NEA Nuclear Science Committee (NSC) Working Party on International Nuclear Data Evaluation Co-operation (WPEC) provides a general framework for international collaboration on differential nuclear data evaluation activities carried out by the participating nuclear data projects. Within this context, the discussion group on nuclear data underlined the importance of an efficient information flow between the different NSC working parties and expert groups. In particular, the group noted the importance of feedback from benchmark exercises on nuclear data issues being communicated to WPEC.

The discussion group put forward the following specific points and recommendations for consideration by the NSC and WPEC:

- The need to improve the High Priority Request List for nuclear data has been recognised and is being addressed by a WPEC sub-group. The objectives are:

  – To solicit an active review of contributions from requesters, so as to maintain a valid working tool for identifying the key measurement and evaluation requirements.

  – To have well-qualified requests, preferably based on sensitivity studies, supported by proper documentation giving the origin of and justification for the request, and with an assigned priority according to generally-agreed criteria.

  – To have rapid on-line access to the list, including simple extraction of top priority items, and allowing feedback and comments on individual requests.

  The recommendation from the discussion group to the NSC was to review the conclusions of the WPEC sub-group responsible for the High Priority Request List for nuclear data and to consider whether further actions should be proposed.

- Evaluation and processing codes should be widely and freely distributed within NEA member countries. Further developments should take place in a coherent and internationally co-ordinated manner.

- To facilitate the use of the latest nuclear data, WPEC should:

  – Consider a standard format for multi-group cross-section libraries for use in testing and benchmarking.

  – Encourage wider availability of application libraries and computer codes for transport calculations.

- A standard set of integral experimental benchmarks should be elaborated, covering a wide range of applications, as a common basis for validation studies.

- Evaluators should be encouraged to include uncertainty information in their files in order to meet the explicit demands from end users.

- WPEC should monitor the international situation regarding experimental facilities and teams active in the fields of measurement, evaluation and validation, and keep the NSC informed of developments.

17.2 Integral nuclear data

Integral nuclear data are essential for the validation of evaluated data libraries. In addition to the above recommendations to the NSC and WPEC on differential nuclear data, the discussion group made several suggestions for actions in relation to integral data. These were:

- The preservation of information on past experiments, ideally in the form of benchmarks (ICSBEP-type benchmarks), should be encouraged.

- Consideration should be given to making new benchmarks available in a common format (IRPhE format).

- Increased international collaboration should be encouraged for work at the few existing facilities for integral experiments still in operation.

Chapter 18

REACTOR PHYSICS AND SYSTEMS BEHAVIOUR

The second discussion group examined future challenges in the field of reactor physics and systems behaviour. Instead of simply listing possible activities that could be undertaken by the NSC, the group based its discussion on the five specific questions below, as proposed by the chairman of the group, Rakesh Chawla.

What are the fields of work (crucial issues) we need to address in the future?

The group agreed upon the following four items of highest priority:

- Fuel cycle issues, covering both conventional and advanced systems, and including:
  - High burn-up.
  - The physics and safety of advanced fast-spectrum systems.
  - Minor actinide recycling in LWRs (recognising the need to review the status of this topic first).
- Uncertainty analysis for nuclear power plant dynamics.
- Refined modelling for materials behaviour, neutronics, thermal-hydraulics and sensitivity analysis.
- The safety of non-electricity energy product systems (recognising the need to first review recommendations from earlier related workshops).

What are the inputs, interactions and synergies we need to seek?

It was felt by the group that individual NSC members should be more conscious of their responsibility regarding the flow of information in respect of, for example, the Generation IV International Forum (GIF), European Commission programmes, the Japanese ADS programme, etc. However, it was noted that the NEA Information Exchange meetings constituted a useful mechanism for disseminating and sharing information.

The anticipated role of the NEA Secretariat in GIF activity was welcomed and would be, if confirmed, very helpful in co-ordinating international efforts in the field of advanced reactor development.

Are the current NSC "modi operandi" the right ones?

The group generally agreed that the present mixture of working parties, expert groups and benchmark and database activities was largely correct and included the right activities. However, it was felt that there should be more emphasis on time-bound activities.

Are the present types of "deliverables" right?

The group noted that the different NSC activities resulted in different types of output or deliverables. It was felt that the deliverables were currently satisfactory, but needed in some cases to be enhanced.

A question on how the deliverables were exploited by the user community was raised during the general discussion at the end of the workshop.

Can we consider setting up sponsored projects and/or facilities? If so, of which type?

The group underlined the strong need to match key experimental needs with the possible use of existing and/or planned facilities to set up NEA-sponsored joint projects. Possible research facilities included:

- The Japanese HTTR for high-temperature materials testing.
- Facilities of the French CEA for integral experiments and irradiation tests.
- The LWR-PROTEUS facilities in Switzerland and/or the VENUS-REBUS facilities in Belgium for high burn-up physics validation.
- The TEF-P/TEF-T facility in Japan for transmutation research.

Chapter 19

FUELS, MATERIALS, COOLANTS AND CHEMISTRY

The third discussion group covered fuels, materials, coolants and chemistry. The group noted the objective of the NSC to facilitate the sharing of information and experience resulting from international as well as national nuclear energy programmes. In view of this, the group felt that the NSC should:

- Support projects and facilities essential for sustaining the development of nuclear systems.
- Publish summaries of the relevant national and international programmes.

The group pointed out the importance of materials-related issues for future nuclear energy systems, in line with several presentations during the workshop.

It was considered to be outside the scope of the group to list all the main R&D topics for ADS, SCWR, VHTR, FGR and LMR systems, as these issues were already covered by the Generation IV, WPPT and European Commission programmes. Despite this, the group recognised that atomistic defect modelling could help in cases where no experimental facilities were available to measure the required data [e.g. for high displacement per atom (dpa) at very high temperatures].

It was also noted that questions related to public acceptance of nuclear energy were dealt with in other NEA committees.

Based on the background discussion described above, the group recommended that the NSC organise the compilation of handbooks on:

- Lead-bismuth coolants (to include advances from current status).
- Advanced nitride, carbide, metallic and IMF fuels.

The group also recommended that the NSC establish two new expert groups on:

- Irradiation defect behaviour, starting from atomistic modelling.
- Materials science, for identifying basic research for innovative systems.

The group also pointed out the importance of providing support to experimental facilities in NEA member countries. The NSC was encouraged to organise and facilitate collaborative efforts using the following types of facilities in member countries:

- Fast flux test facilities.
- High-temperature irradiation facilities.
- Classical material test reactors.

The group did not make any special recommendations in the field of coolants and chemistry, as it was noted that the WPPT already had several ongoing activities in these areas. In addition, it was felt that many chemical problems related to current nuclear systems, including high burn-up fuel, could be solved by developments in engineering.

Annex A

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