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Research and Test Facilities Required in Nuclear Science and Technology

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NUCLEAR ENERGY AGENCY
Organisation for Economic Co-operation and Development

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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, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the 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.

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Foreword

In 2001, the NEA Nuclear Science Committee (NSC) initiated a study on *Research and Development Needs for Current and Future Nuclear Energy Systems*, which was published in 2003. The NSC further developed its work in this area in 2005 through the formation of an expert group whose designated task was the preparation of a report on *Research and Test Facilities Required in Nuclear Science and Technology*.

The expert group met in May and December 2005, May and October 2006 and April 2007. At these meetings, participants reviewed existing activities within the NEA as they related to the task of the group, as well as on similar activities in other organisations.

Following the initial expert group meeting, a database listing all relevant facilities was prepared using information from the earlier NSC report, from members of the expert group and other scientists in their countries, from relevant databases and from other reports such as the NEA report on *Support Facilities for Existing and Advanced Reactors* (SFEAR). The database was released for public access via the NEA website in February 2008 at *www.nea.fr/rtfdb/*.

Based on information collected for the database and on input from expert group members and a small number of external contacts, this report identifies the current portfolio of facilities (of an international dimension) that exist in NEA member countries and relates them to the R&D needs within the scope of the NSC. In addition, and as a particular focus of the expert group's activity, the report has sought to identify the research facilities required to satisfy future needs.

Acknowledgements

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ACP	Advanced spent fuel Conditioning Process
ACPF	Advanced spent fuel Conditioning Process Facility
ADOPT	Advanced Options for Partitioning and Transmutation
ADS	Accelerator-driven Systems
AFCI	Advanced Fuel Cycle Initiative
AFCL	Advanced Fuel Cycle Laboratory
AGF	Alpha Gamma Facility
ALISIA	Assessment of Liquid Salts for Innovative Applications
ALWR	Advanced Light Water Reactor
APS	Advanced Photon Source
ARTIST	Amide-based Radio-resources Treatment with Interim Storage of Transuranics
ATALANTE	Atelier alpha et laboratoires pour analyses, transuraniens et études de retraitement
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BARC	Bhabha Atomic Research Centre
BIP	Behaviour of Iodine Project
CANDU	Canadian Deuterium Uranium Reactor
CBP	Chaîne blindée procédé hot cell
CCVM	Code Validation Matrix
CEA	Commissariat à l’énergie atomique
CENDL	Chinese Evaluated Nuclear Data Library
CERCER	Ceramic-ceramic
CFPs	Coated Fuel Particles
CIAE	China Institute of Atomic Energy
CJD	Center Jadernykh Dannykh
CMR	Chemistry and Metallurgical Research Facility
CNRA	Committee on Nuclear Regulatory Activities
CPF	Chemical Processing Facility
CRIEPI	Central Research Institute of Electric Power Industry
CSEWG	Cross-section Evaluation Working Group
CSNI	Committee on the Safety of Nuclear Installations
CT	Computer Tomography
D&SM	Design and Safety Management
DAE	Department of Atomic Energy (India)
DCA	Deuterium Criticality Assembly
DCH	Direct Containment Heating
DDP	Dimitrovgrad Dry Process
DELTA	Development of Lead-bismuth Target Applications
DNB	Departure from Nucleate Boiling
DOE	Department of Energy (USA)

DOVITA	Dry Reprocessing, Oxide Fuel, Vibropac, Integral, Transmutation of Actinides
dpa	Displacements Per Atom
EAC	Environmentally Assisted Cracking
EDF	Électricité de France
EFF	European Fusion File
EFIT	European Facility for Industrial Transmutation
EGIS	Expert Group on the Implications of Radiological Protection Science
EISOFAR	European Innovative Sodium-cooled Fast Reactor
ELSY	European Lead-cooled System
ENDF/B	Evaluated Nuclear Data File
ENSDF	Evaluated Nuclear Structure Data File
EPMA	Electron Probe Micro Analysis
EPR	European Pressurised Reactor
ESF	European Science Foundation
ESRF	European Synchrotron Radiation Facility
ESS	European Spallation Source
ETD	European Transmutation Demonstration
ETDR	Experimental Technology Demonstration Reactor
EU	European Union
EURADOS	European Radiation Dosimetry Group
EXAFS	Extended X-ray Absorption Fine Structure
F/M	Ferritic/Martensitic
FaCT	Fast Reactor Cycle Technology Development
FBR	Fast Breeder Reactor
FBTR	Fast Breeder Test Reactor
FCF	Fuel Conditioning Facility
FCI	Fuel-coolant Interaction
FCMFC	Fuel, Core Materials and Fuel Cycle
FENDL	Fusion Evaluated Nuclear Data Library
FEUNMARR	Future European Union Needs in Material Research Reactors
FFTF	Fast Flux Test Facility
FGR	Fission Gas Release
FR	Fast Reactor
GACID	Global Actinide Cycle International Demonstration
GANEX	Group Actinide Extraction
GANIL	Grand accélérateur national d'ions lourds
GCFRs	Gas-cooled Fast Reactors
GCRs	Gas-cooled Reactors
GEDEPEON	Gestion de déchets et production d'énergie par options nouvelles
GELINA	Geel Electron Linear Accelerator
Gen. IV	Generation IV
GFRs	Gas-cooled Fast Reactors
GIF	Generation IV International Forum
GNEP	Global Nuclear Energy Partnership
GSI	Gesellschaft für Schwerionenforschung
GTAC	Graphite Technical Advisory Committee
GTSI	Glenn T. Seaborg Institute
GUINEVERE	Generator of Uninterrupted Intense Neutrons at the Lead VENUS Reactor
HANARO	Hi-flux Advanced Neutron Application Reactor
HBWR	Halden BWR

HCLWR	High Conversion Light Water Reactor
HEU	Highly Enriched Uranium
HFEF	Hot Fuel Examination Facility
HFIR	High Flux Isotope Reactor
HFR	High Flux Reactor
HLLW	High-level Liquid Waste
HLM	Heavy Liquid Metals
HLW	High-level Waste
HPLWR	High-performance Light Water Reactor
HPPA	High-power Proton Accelerators
HPRL	High-priority Request List
HTGRs	High-temperature Gas-cooled Reactors
HTR	High-temperature Reactor
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation-assisted Stress Corrosion Cracking
ICNC	International Conferences on Nuclear Criticality Safety
ICSBEP	International Criticality Safety Benchmark Evaluation Project
ICSM	L'Institut de chimie séparative de Marcoule
IEA	International Energy Agency
IFMIF	International Fusion Materials Irradiation Facility
IFPE	International Fuel Performance Experiments
IGCAR	Indira Gandhi Centre for Atomic Research
IGORR	International Group on Research Reactors
ILL	Institut Laue-Langevin
ILL-HFR	Institut Laue Langevin – High Flux Reactor
IMF	Inert Matrix Fuel
INET	Institute of Nuclear and New Energy Technology (Tsinghua University)
INFCIS	Integrated Nuclear Fuel Cycle Information Systems
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
IPCC	International Panel on Climate Change
IPPE	Institute of Physics and Power Engineering
IPSN	Institut de protection et de sûreté nucléaire
IRPhE	International Reactor Physics Benchmark Experiments
IRSN	L'Institut de radioprotection et de sûreté nucléaire
ISI&R	In-service Inspection and Repair
ITER	International Thermonuclear Experimental Reactor
IUPAP	International Union of Pure and Applied Physics
JAEA	Japan Atomic Energy Agency
JAEC	Japan Atomic Energy Commission
JANNUS	Joint Accelerators for Nanosciences and Nuclear Simulation/Jumelage d'accélérateurs pour les nano-sciences, le nucléaire et la simulation
JEFF	Joint Evaluated Fission and Fusion
JENDL	Japanese Evaluated Nuclear Data Library
JHR	Jules Horowitz Reactor
JMTR	Japan Materials Testing Reactor
JNFL	Japan Nuclear Fuel Limited
JRC	Joint Research Centre
JSNS	Japan Spallation Neutron Source
KAERI	Korea Atomic Energy Research Institute

KALIMER	Korea Advanced Liquid Metal Reactor
KART	Kumatori Accelerator-driven Reactor Test
KRI	Khlopin Radium Institute
KUCA	Kyoto University Critical Assembly
KURRI	Kyoto University Research Reactor Institute
LACANES	Lead-alloy-cooled Advanced Nuclear Energy Systems
LANL	Los Alamos National Laboratory
LANSCE	Los Alamos Neutron Science Center
LBE	Lead-bismuth Eutectic
LFR	Lead-cooled Fast Reactor
LIFE@PROTEUS	Large-scale Irradiated Fuel Experiments at PROTEUS
LLFP	Long-lived Fission Products
LLRN	Long-lived Radionuclides
LMC	Lead Mini-cell
Ln	Lanthanide
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LWR	Light Water Reactor
MA	Minor Actinides
MARCEL	Module avancé de radiolyse dans les cycles d’extraction-lavages
MDD	Modified Direct Denitration
METI	Ministry of Economy, Trade and Industry
MOX	Mixed-oxide
MSRs	Molten Salt Reactors
MTRs	Materials Test Reactors
MTS	Materials Test Station
MUSE	Multiplication avec une source externe
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications
NDC	Nuclear Development Committee
NEA	Nuclear Energy Agency
NEXT	New Extraction System for TRU Recovery
NGNP	Next Generation Nuclear Plant
NISA	Nuclear and Industrial Safety Agency
NNDC	National Nuclear Data Center
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRDC	Nuclear Reaction Data Centres Network
NSC	Nuclear Science Committee
NuCoC	Nuclear Centres of Competence
NUPEC	Nuclear Power Engineering Corporation
NuPECC	Nuclear Physics European Collaboration Committee
OCEAN	Oscillation en cœur d’échantillons d’absorbants neutroniques
ODS	Oxide Dispersion Strengthened
OECD	Organisation for Economic Co-operation and Development
ORELA	Oak Ridge Electron Linear Accelerator
ORNL	Oak Ridge National Laboratory
OSMOSE	OScillations dans MINERVE d’Isotopes dans des Spectres Eupraxiques
P&T	Partitioning and Transmutation
PA	Project Arrangement

PATEROS	Partitioning and Transmutation European Roadmap for Sustainable Nuclear Energy
PBMR	Pebble-bed Modular Reactor
PBWFR	Pb-Bi-cooled Direct Contact Boiling Water Fast Reactor
PCI	Pellet-clad Interaction
PCMI	Pellet-clad (mechanical interaction)
PEACER	Proliferation-resistant, Environment-friendly, Accident-tolerant, Continual and Economical Reactor
PFBR	Prototype Fast Breeder Reactor
PHWRs	Pressurised Heavy Water Reactors
PIE	Post-irradiation Examination
PSA	Probabilistic Safety Assessment
PSI	Paul Scherrer Institut
PWR	Pressurised Water Reactor
R&D	Research and Development
RAPHAEL	Reactor for Process Heat, Hydrogen and Electricity Generation
RCS	Reactor Coolant System
RIAR	Research Institute of Atomic Reactors
RIAs	Reactivity Insertion Accidents
RMWR	Reduced-moderation Light Water Reactor
RPI	Rensselaer Polytechnic Institute
RPV	Reactor Pressure Vessel
RRP	Rokkasho Reprocessing Plant
RSICC	Radiation Safety Information Computational Center
RTFDB	Research and Test Facilities Database
RWMC	Radioactive Waste Management Committee
SA	Severe Accidents
SATIF	Shielding Aspects of Accelerators, Targets and Irradiation Facilities
SCC	Stress Corrosion Cracking
SCW	Supercritical Water
SCWRs	Supercritical Water-cooled Reactors
SEM	Scanning Electron Microscope
SERENA	Steam Explosion Resolution for Nuclear Applications
SESAR	Senior Group of Experts on Nuclear Safety Research
SFCOMPO	Spent Fuel Isotopic Composition
SFEAR	Support Facilities for Existing and Advanced Reactors
SFRs	Sodium-cooled Fast Reactors
SINBAD	Shielding Integral Benchmark Archive Database
SLS	Swiss Light Source
SMINS	Structural Materials for Innovative Nuclear Systems
SNE-TP	Sustainable Nuclear Energy Technology Platform
SNF-TP	Sustainable Nuclear Fission Technology Platform
SNS	Spallation Neutron Source
SNU	Seoul National University
SPIRAL	Système de production d'ions radioactifs accélérés en ligne
SSRL	Stanford Synchrotron Radiation Laboratory
STACY	Static Experimental Critical Facility
T/H	Thermal-hydraulics
TBP	Tri-butyl Phosphate
TCA	Tank-type Critical Assembly

TDB	Thermochemical Database
TIT	Tokyo Institute of Technology
TODGA	Tridentate Diglycolamide
TOF	Time-of-flight
TP	Total Partitioning
TRACY	Transient Experiment Critical Facility
TRG	Technical Review Group
TRPO	Trialkyl Phosphine Oxides
TRU	Transuranic
TWGRR	Technical Working Group on Research Reactors
UNEX	Universal Extraction
VELLA	Virtual European Lead Laboratory
VHTR	Very High-temperature Reactor
VHTRC	Very High-temperature Reactor Critical Assembly
VVER	Vodo-Vodyanoi Energetichesky Reactor (Водо-водяной энергетический реактор)
WGFCS	Working Group on Fuel Cycle Safety
WGFS	Working Group on Fuel Safety
WGHOF	Working Group on Human and Organisational Factors
WGRISK	Working Group on Risk Assessment
WGRNR	Working Group on the Regulation of New Reactors
WNA	World Nuclear Association
WPEC	Working Party on International Nuclear Data Evaluation Co-operation
WPFC	Working Party on Scientific Issues of the Fuel Cycle
WPMM	Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems
WPNCS	Working Party on Nuclear Criticality Safety
WPRS	Working Party on Scientific Issues of Reactor Systems
XANES	X-ray Absorption Near Edge Structure
XAS	X-ray Absorption Spectroscopy
XRF	X-ray Fluorescence
XT-ADS	eXperimental Facility demonstrating the technical feasibility of Transmutation in an Accelerator-driven System

Executive summary

Activities in nuclear science using research and test facilities are key to maintaining scientific progress in the nuclear technology field. However, economic pressures are constraining the level of activity relating to continued support of existing reactors, development and planning for new construction, and work towards advanced reactors. In addition, the preservation of the knowledge base associated with integral data, which has been accumulated over the course of nuclear development to date, is becoming uncertain.

This report and an accompanying database is the outcome of the work of an Expert Group set up by the Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA) to extend earlier studies and a workshop on "Research and Development Needs for Current and Future Nuclear Systems"[1] it commissioned in 2002.

The Expert Group extended across a total membership of 20 drawn from 10 countries plus the IAEA, the EU and the OECD with some participants largely contributing by correspondence.

To facilitate its overall task, the Expert Group has developed a database of over 700 nuclear research and test facilities. An extensive verification operation was conducted in order to confirm the records with the facility operators. The database was then released for public use in the spring of 2008 under the name "RTFDB" for Research and Test Facilities Database. Access over the Internet is available at the URL: *www.nea.fr/rtfdb/*.

A user can search the database by country, facility type (reactor, accelerator, etc.), application (ADS, fuel research, etc.) or the organisations owning facilities.

The Expert Group believes that the database has provided a most useful tool during the present review and has also already become a valuable resource for the scientific community world wide. The Expert Group therefore encourages the NSC to continue to update and expand RTFDB in the future. This would assist the NEA in its function of regularly reviewing the situation of nuclear facilities, especially in those fields for which the risk of losses are high, in order to monitor and comment upon undesirable trends.

The Expert Group has classified its discussion of facilities (and this report) based on applications using the same grouping as in the RTFDB database in order to best identify and analyse the needs for such facilities.

The first topic described is nuclear data. The Expert Group has identified as being particularly important the availability of modern facilities able to provide results for materials of current technological interest and at the level of accuracy required for present applications. In addition, some of these facilities must be capable of handling active materials. In this light, the number of appropriate facilities is much more limited than a quick reading of the full list of facilities related to nuclear data measurement in RTFDB may give. It should also be borne in mind that the present consideration does not just cover cross-sections as there are also other "microscopic" data requirements to be met.

The shortage of facilities should be understood as extending to those for integral measurements; no new construction of such facilities has taken place for many years. Only a small number of critical facilities with sufficiently diverse supplies of simulation materials are still operational today. Yet, such facilities are essential to the validation of nuclear data and to neutron physics programmes.

The Expert Group believes that differential and integral measurement facilities will continue to be necessary for continued progress in nuclear data and to preserve compatibility with users' needs.

1. Detailed references to articles, conferences, meetings, etc., will be found in the main body of the report.

Therefore, maintenance, even upgrading, is required and new construction projects should be encouraged. Making facilities accessible internationally to many users should also be encouraged.

As well as addressing the issue of physical facilities the Expert Group also makes recommendations relating to the maintenance of: *i)* expertise; *ii)* infrastructures and samples. For the former, it is noted that the NEA Steering Committee for Nuclear Energy, in making its recent statement and recommendations regarding a government role in ensuring qualified human resources in the nuclear field, has served to emphasise the skills issue. The Expert Group believes that better integration between academia and other user communities would be of great advantage for the whole process of evolving evaluation and validation of databases to the appropriate standard required for nuclear energy applications.

In relation to infrastructure, it should be noted that requirements arising from: *i)* lifetime extension of presently operating reactors; *ii)* evolution of new reactors and their associated fuel cycle infrastructures; *iii)* the more stringent economical, environmental and safety burdens of the current era put challenging demands on the accuracy of the models and the underlying nuclear database. They will only be met if the present level of expertise and facilities is maintained. The provision of samples for measurement is an important infrastructure activity and the availability of facilities able to supply the demands of modern measurements is a crucial issue.

The Expert Group has identified: *i)* operational safety of nuclear power plants, *ii)* dealing with nuclear waste; *iii)* fusion, as cases in which new facilities (and expertise) may become necessary in the future.

In the field of reactor development the Expert Group concludes that, generally speaking, obtaining a perspective on the future needs relating to specific research fields and types of reactors is extremely difficult since variations in importance of particular reactor designs over time and in particular countries influences the view. However, the current resurgence of interest in novel reactor concepts is certainly widening the field of development.

As well as the direct requirements for specific research reactors and criticality assemblies related to particular reactor designs, the Expert Group concludes that there is also a need for versatile zero (or low) power reactors and subcriticality assemblies for basic reactor physics experiments and educational purposes. These are required in order to extend the knowledge of the skills base, a requirement which applies for any future nuclear energy developments regardless of reactor types adopted.

The Expert Group believes that some new research reactors will be required though demands can partly be met by continuing to operate existing research reactors as long as they meet international safety standards. In particular, the need for criticality facilities should be underlined.

The Expert Group also stresses that there is a continuing requirement for research reactors as sources of neutrons, in particular for high-intensity neutron sources as probes of materials. This arises in order to provide continuous irradiation or more representative conditions, alongside the resource evolving from the current trend for provision of large and/or multi-purpose accelerators. Both types of facility (accelerators and reactors) are seen as being required and are complementary in their abilities.

The Expert Group notes that several OECD countries have either started construction or have announced their intention to build new nuclear reactors and other nuclear fuel cycle facilities, while non-OECD countries like Russia, China and India have already launched programmes leading to implementation in this field. In particular, significant efforts have been deployed in order to build fast neutron reactors. However, no new fast spectrum power reactor is scheduled to come into operation before 2020 in OECD countries, which will likely hamper innovative R&D activities, especially in connection with fuel research.

The Expert Group concludes that the recent expansion of the Global Nuclear Energy Partnership (GNEP) is an indication of the desire for increased collaboration within the international nuclear power community in order to address near-term developments. With a longer-term perspective, the Generation IV (Gen. IV) initiative similarly brings together a number of countries in developing reactor designs for future application.

The Expert Group recommends that further federation of the financial, scientific and technical efforts of the OECD countries could optimise available resources. This would have the aim, for instance, of better usage of existing research reactors, or of building jointly-owned nuclear facilities, following a similar approach to that adopted by the Institut Laue-Langevin (ILL). The Expert Group notes the key role played by international institutions in the promotion of such co-operation between countries and recommends that existing synergies between NEA and IAEA activities in this matter might be explored further.

The Expert Group also wishes to encourage the exchange of researchers, of research plans, and of results, such as the collaboration between facilities in France and Belgium (EOLE and VENUS).

Equally the Expert Group concludes that there is a need to conserve the body of knowledge built up and, as far as is appropriate, to retain existing facilities operational because experience has shown that abandoned technologies may experience a revival (*e.g.* the HTR).

Within the context of building up the base of knowledge within younger researchers, the Expert Group commends the initiative of: *i)* France and Germany in creating and building up the Frédéric Joliot/Otto Hahn Summer School on Nuclear Reactors Physics, Fuels and Systems; *ii)* the formation of the World Nuclear University. Initiatives of this form are to be encouraged.

In relation to neutron applications (including neutron scattering) new neutron scattering sources appear to be coming on stream to keep pace with the projected losses due to source closures. However, there will be a concentration of resources in a fewer number of larger sources and there will be a shift in the centre of gravity from Europe, to North America and Japan, unless the ESS is built soon.

It is anticipated that neutron diffraction and small angle neutron scattering measurements of the structure and the defects therein will continue to play a role in testing and developing new engineering materials for nuclear technology. It is also to be expected that the burgeoning fields of strain scanning and texture analysis will grow in importance as they find a wider audience. Inelastic neutron scattering measurements will probably constitute a lower-level activity in this field, though it should retain an important role in measuring scattering kernels which will be used in Monte Carlo studies to study the behaviour of moderator performance.

An extensive use of neutron radiography techniques in fuel fabrication processes requires the adoption of standardised methods for non-destructive controls. Standardised procedures for qualification of beams, technicians and image treatments should be established, and neutron radiography facilities should operate in a co-ordinated way to accomplish this task.

Application of modern neutron radiography methods such as phase-contrast radiography could integrate neutron scattering and neutron radiography competences in a synergic action offering a wider applicability of neutron imaging techniques.

Accelerator-driven systems (ADS) and transmutation technologies are becoming important for the sustainable development of nuclear energy all over the world, but have technical challenges spread over a wide range of fields. Thus sharing experimental efforts in a systematic way is highly desirable, MEGAPIE being a good precursor for such international collaboration.

The Expert Group regards an international roadmap for ADS as being of importance.

It is considered necessary to build a dedicated accelerator in order to demonstrate its reliability, controllability, economy and safety. Such a demonstration accelerator would be coupled with a subcritical reactor as an experimental ADS. There is some feeling that a global programme (perhaps in a similar form to the ITER project in fusion energy development) is desirable.

A materials properties database for minor actinides and long-lived fission products is important in order to design the fuels for transmutation systems. Rarity of materials and licence restrictions on the amounts permitted in a facility make it difficult to measure the physical and chemical properties of these materials. Therefore, retention of hot cell laboratories is regarded as important by the Expert Group. Equally, means of procurement of samples for materials property measurements, and also for nuclear data measurements and reactor physics experiments are seen as vital.

A materials properties database for accelerator-related items such as windows and targets needs to be prepared covering a wide range of design conditions such as temperature, oxygen content and flow velocities of the lead-bismuth eutectic (LBE), beam density and irradiation period. At present, the status of material irradiation data is too poor to create a reliable design for a window-type target.

For the alternative windowless design a demonstration using a real megawatt class proton beam is considered necessary to prove the engineering feasibility, before connecting it to a subcritical reactor, as well as mock-up experiments without beams. Currently, the stable control of a target with a free surface might be difficult when a high-power proton beam is incident.

The Expert Group recommends that an international benchmark of experiments be organised to establish a global standard for LBE materials properties. Further, an integral test to verify the feasibility of oxygen control in a realistic reactor vessel would be necessary before constructing a large-scale LBE-cooled nuclear system.

The thermal-hydraulics of the LBE coolant should also be verified by experimental work, *e.g.* local erosion of the materials in the core by the high-speed LBE flow. Large-scale components such as heat exchangers and pumps also require development for LBE.

Experimental irradiation facilities are essential for fuel development and testing. The Expert Group therefore recommends that the lifetime of existing key irradiation facilities, such as the Halden Reactor and the Advanced Test Reactor should be extended; the Jules Horowitz Reactor (JHR) Co-ordination Action development is also of particular note. In terms of knowledge retention, the International Fuel Performance Experiments (IFPE) database should be maintained and extended.

The Expert Group emphasises that new facilities will be required for Gen. IV conditions. While some specific loops are already under development, testing or construction, the need for fast spectrum irradiation facilities should be stressed, as the stakes are high with regard to Gen. IV fast reactor fuel research; there is presently a shortage of facilities. In addition, it should be emphasised that fuel R&D is a long-term process. Many irradiation tests under representative conditions are necessary before a new fuel can be considered qualified for use.

Besides the fuel itself, related structural materials must be tested as well, again under the proper regime of neutron spectrum, fluence, temperature, coolant environment, etc.

Concerning TRISO coated fuel particles (CFPs) for high-temperature gas-cooled reactors (HTGRs), the Expert Group concurs with the recommendation in the report *Support Facilities for Existing and Advanced Reactors* (SFEAR) that international collaboration should be promoted due to the importance of fuel performance to HTGR safety, the long lead time and the cost of fuel testing. In addition, the Expert Group regards it as important to maintain existing test reactors, such as CABRI, NSRR and ATR, due to their ability to test HTGR fuels.

In relation to hot cells and post-irradiation examination (PIE), the Expert Group recommends that the long-term availability of hot cells for fuel examination be guaranteed. Continuous observation of global developments will be required in order to monitor and predict the utilisation of hot cells, particularly in the longer term.

The Expert Group concludes that for fuel cycle chemistry: *i)* a considerable chemical engineering effort will be required for scaling proposed partitioning processes to pilot scale and industrial prototype; *ii)* possession of R&D facilities which fulfil the minor actinide (MA) handling requirements is becoming a determining factor among partitioning and transmutation (P&T) oriented countries in view of the strict MA handling regulations and the construction costs; *iii)* irradiation facilities for studying the radiolysis resistance of ligands are essential for the development of new organic reagents and an actinide laboratory capable of handling significant amounts is also needed for alpha radiolysis studies; *iv)* laboratories for organic synthesis, analytical chemistry and structural chemistry are also important for the development of partitioning processes.

For materials the Expert Group concludes that facilities will continue to be required to cover the range of requirements for: *i)* materials irradiation; *ii)* modelling validation/materials characterisation; *iii)* materials testing.

The continuing availability of materials test reactors (MTRs) and the facilities that such reactors are able to provide is regarded by the Expert Group as an essential feature of the study of materials of interest to reactors and other branches of nuclear science. As noted earlier, the scope of the irradiation capabilities will need to increase as the demands from work on new reactor types evolves.

The availability of large facilities such as spallation sources and reactors for analysis of materials is equally deemed essential.

The safety section of this report has largely been derived in collaboration with the Committee on the Safety of Nuclear Installations (CSNI) effort that led to the SFEAR report. Therefore the conclusions and recommendations that follow are essentially those in the SFEAR report from which amplification should be sought.[2] However, the Expert Group notes the following:

i) Many large, expensive and unique facilities are projected to close over the next five years. CSNI efforts concerning facility preservation should focus on large facilities, whose loss would mean the loss of unique capability as well as the loss of substantial investment ("preservation" also includes maintaining expertise, knowledge, capabilities and personnel essential to infrastructure conservation).

ii) Both CSNI and the Committee on Nuclear Regulatory Activities (CNRA) should take steps to encourage industry co-operation by emphasising the responsibility of industry to develop sufficient data to support their applications, the benefits of co-operative research and the value of preserving critical research infrastructure.

iii) Because of the large numbers of hot cells and autoclaves, each country is recommended to monitor the status of these essential facilities and bring to CSNI's attention any concerns regarding loss of critical infrastructure.

iv) Certain safety issues have no large-scale facilities identified for the conduct of relevant research so the appropriate CSNI working groups should evaluate whether or not such facilities are needed to support resolution of these issues.

The SFEAR report recommendations are directed toward those actions that CSNI could take in the short term to prevent the loss of key facilities in imminent danger of closure. In the thermal-hydraulics area, a couple of unique and expensive facilities are in danger of being closed in the next one to two years. Further arguments and a preference for retention are given in the SFEAR report. In the severe accident area, most facilities supporting the resolution of issues in pre-core melt conditions, combustible gas control and coolability of over-heated cores for BWRs, PWRs, VVERs and ALWR are in danger in the short term. The SFEAR report recommends that three specific facilities should be preserved due to their replacement cost, high relative ranking and versatility. In the other technical areas (fuels, integrity of equipment and structures) no short-term CSNI actions were recommended. The SFEAR report recognises that implementation of the above recommendations are dependent upon interest and commitment of the host countries to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries.

In the longer term, it is recommended that CSNI adopt a strategy for the preservation of a research facility infrastructure, based upon preserving unique, versatile and hard-to-replace facilities. The strategy should include consideration of short- and long-term priorities, cost of preservation, competition for funds, and contingency plans in case of facility loss. The SFEAR report identifies factors recommended for use in deriving conclusions and recommendations.

A table of critical research facility infrastructure needs is given in the SFEAR report; those considered unique, hard-to-replace and having high relative importance in their technical area are identified. CSNI are recommended to monitor the status of these facilities in the longer term with a goal of taking action to meet the critical research infrastructure needs. For new reactors and technologies, it is recommended that CSNI take an active role in encouraging and organising co-operative research efforts, hence infrastructure preservation. Host country interest is regarded as an important factor in determining which facilities to preserve.

In the area of nuclear and radiochemistry research the Expert Group recommends that that integrated hot cell laboratories be retained to measure basic physical and chemical properties of actinide compounds. The group also notes that hot cells and glove boxes owned by universities are important tools for education. It therefore recommends a network such as the pooled facilities in ACTINET as an important approach for the effective sharing of facilities and to promote international collaboration.

The group believes synchrotron radiation facilities capable of measuring radioactive samples should be retained, *e.g.* the SSRL for measurement of plutonium samples. In addition there are future

2. The SFEAR remit was expressly limited to the safety issues, research needs and supporting research facilities associated with currently operating water-cooled reactors in NEA member countries plus high-temperature gas-cooled reactors (HTGRs). However, fast reactors were not part of the consideration of the SFEAR group.

requirements to measure properties of actinides and LLFP in spent fuel directly by means of X-ray absorption spectroscopy. Special beam lines like MARS at SOLEIL are needed for measuring highly radioactive samples.

In a final, miscellaneous section, the Expert Group considers issues relating to nuclear process heat for hydrogen production; the Expert Group concurs with the recommendation for further international collaboration in this field made during the 3rd Information Exchange Meeting on the Nuclear Production of Hydrogen in October 2005. It also endorses the need for consideration of safety issues to ensure that the chemical and nuclear facilities pose no risk to each other. The Expert Group notes the need for co-operation on matters such as: *i)* safety; *ii)* materials and chemical property measurement and verification; *iii)* materials development, including structural materials, membranes, and catalysts; *iv)* advanced fabrication techniques and the implied need for facilities to elucidate the corresponding information.

Finally, in a chapter on related NEA activities, the Expert Group provides a brief summary of the work of the other parts of the NEA for information on additional programmes, committees and work relevant to its task.

Chapter 1: Introduction

As a result of evolving economic situations in many OECD/NEA member countries, which have often included declining budgets in nuclear development activities, numerous problems are arising in maintaining the present level of scientific activity in the nuclear technology field. Examples include: continued support of existing reactors, development and planning for new build and work towards advanced reactors, together with the underpinning research and development activities using research and test facilities (and which form the focus of this report). Added to this, the preservation of the totality of the knowledge associated with integral data, accumulated over the course of nuclear development to date, is becoming uncertain.

In order to overcome these difficulties, it has become more and more important both to develop advanced nuclear technologies and also to form infrastructures within international frameworks. The NEA has promoted various activities of this sort as is exemplified below:

- The NEA Nuclear Science Committee (NSC) initiated a study on research and development (R&D) needs in nuclear science, held a workshop on "R&D Needs for Current and Future Nuclear Systems" in November 2002 in Paris and published a report on these topics (NEA, 2003).
- In parallel, the NSC continues to review existing integral reactor physics data within the International Reactor Physics Benchmark Experiments (IRPhE) Project (NEA, 2008r).
- The Nuclear Development Committee (NDC) is undertaking work on areas of similar interest: *i)* partitioning and transmutation of actinides and fission products has been the subject of several meetings; *ii)* experience with the use of plutonium on the industrial scale has been reviewed; *iii)* the role of and conditions for deploying small- and medium-sized reactors have been analysed.
- The Committee on the Safety of Nuclear Installations (CSNI) has reviewed the continued need for experimental R&D facilities in the area of safety and has recently published an expert group report entitled *Support Facilities for Existing and Advanced Reactors (SFEAR)* (NEA, 2007d).

The NSC concluded that further discussions on R&D needs were essential in order to profit from the synergy of these and similar studies. In addition, recommendations concerning the scientific research and associated facilities needed for the future development of nuclear energy, using new technology developed in member countries, should be made.

The NSC therefore tasked an Expert Group in 2005 to seek to anticipate realistic future needs for R&D facilities in nuclear science based upon the results of the NEA activities mentioned above and in close collaboration with other NEA standing technical committees. A specific aim of the study was to contribute to promoting international collaboration for the development of new nuclear technologies.

The detailed objectives of the study are specified in the Mandate of the Expert Group (see Appendix A) but, in brief, the group was charged with preparing a report on *Future Research and Test Facilities Needed in Nuclear Science*, focusing on:

- reviewing the status of research and test facilities world wide and clarifying the future needs of such facilities as corresponds to R&D needs in nuclear science and technology, collaborating with other technical standing committees, based on results from the NSC study as well as those of the NDC and CSNI on review of the status of research and test facilities;
- monitoring the NSC IRPhE activity on existing integral data of reactor characteristics and fuel cycle in order to identify the future needs of research facilities;
- establishing recommendations on future needs of research facilities in nuclear science for international collaboration.

The membership of the Expert Group is given in Appendix B. In assessing its tasks the Expert Group decided that it would be clearer if the title of the activity were slightly modified to: "Future Research and Test Facilities Needed in Nuclear Science *and Technology*".

Appendix A provides some amplification of the programme, but in outline the deliverables of the project were to:

- organise Expert Group meetings to review and exchange information on status of integral data and needs for research and test facilities for future R&D in the field of nuclear science and technology (the work to be undertaken in co-ordination with CSNI and NDC);
- establish a database of research and test facilities for R&D in the field of nuclear science and technology, and to clarify the status and needs of these facilities;
- produce a report on the status of integral data and the need of research and test facilities for future R&D in nuclear science and technology.

The Expert Group activity was considered timely for many reasons, including:

- Climate change issues are becoming more clearly understood and are leading to greater consideration of energy generation schemes with smaller carbon footprints than traditional fossil-fuelled power stations.
- As an example, the European Parliament said in a report in October 2007 that nuclear energy would be "indispensable if basic energy needs are to be met in Europe in the medium term" (European Parliament, 2007).
- On the other hand and at the same time there has been growing pressure and competition for the various fossil fuels resources demonstrated by significant increases in prices for these commodities.
- As a result there has been a growing interest in possible new nuclear build during the lifetime of the Expert Group.
- According to the International Atomic Energy Agency (IAEA) Power Reactor Information System website (IAEA, 2008f), there are 439 nuclear power plants in operation and 35 under construction around the world and a number of reactors that had been long-term shutdown have been restarted.
- In the OECD area nuclear energy represents nearly a quarter of the electricity supply (NEA, 2008o) while world wide it represents about 16% of electricity supply (WNA, 2007), an achievement that can be ascribed to an effective and safe performance of both nuclear power plants and the related fuel cycle facilities.
- For new build it should be noted that Generation III (Gen. III) and III+ designs tend to make more use of "passive" safety features and/or design simplifications and these require appropriate validation for the development of the necessary safety cases.
- Thus there are, for example, active plans for construction of nuclear power stations in, amongst others, China, Japan, USA, Russia, India, and South Korea totalling 87 GW(e) with a further 18 GW(e) elsewhere in the world as of September 2008 (WNA, 2008g). Added to this, the WNA lists a further 198 GW(e) as "proposed" with China again the highest contributor with 63 GW(e) and the USA with 26 GW(e). It should be noted that, for example, Brazil, Bulgaria, Indonesia, Mexico and South Africa appear in these two lists as having: *i)* "plans for construction"; *ii)* "reactors proposed". In addition, it is known that the UK is actively considering building a new generation of nuclear power stations and the Italian government has indicated it plans to resume building nuclear energy plants within five years.
- These developments indicate that security of supply issues for both fossil and nuclear fuels can be regarded as significant. It also confirms that, in the longer term, there will be a need for breeder reactors because uranium resources are not infinite. In this framework and also associated with the desire for waste minimisation the Global Nuclear Energy Partnership (GNEP, 2008) and Generation IV (Gen. IV) (GIF, 2007) activities are moving forward, with both programmes involving international collaboration. These also indicate moves to reactor designs and fuel cycles not in current use and thus encompass new needs for research.

- Also related to fuel cycle interests are the currently active research areas of partitioning and transmutation (P&T) together with the potential for use of accelerator-driven systems (ADS).
- Linked to fast reactor technology there is the restart of MONJU (JAEA, 2008) and the planning towards new fast reactors to come on stream – with Gen. IV designs as a longer-term goal.
- In the shorter term, there is also significant effort being devoted to life extension of existing plants and even up-rating beyond current maximum output levels for both economic and resource conservation reasons. These demands have implications for better understanding of, for example, graphite in gas-cooled reactors and radiation damage of pressure vessels in water reactors.
- Beyond the electricity production sphere, there is the ever-expanding utilisation of nuclear techniques such as neutron scattering and neutron radiography in a wide range of technological fields through application of the neutron's interesting scattering and magnetic moment properties. New facilities are coming on line and others are being contemplated.
- Equally, and associated with activities like Gen. IV, continuing interest in a hydrogen-based economy exists, with the potential for advanced nuclear reactors to provide the energy required to drive the hydrogen production.

As a result, the work of the Expert Group has concentrated on both nuclear energy (within which the future needs for research relating to both current and future systems is included) and the applications of nuclear science, as indicated above. The aim has been to assess how the currently available research facilities could cope with future needs or whether new facilities will be needed in the foreseeable future. Equally, some existing facilities may be at risk and the review has attempted to identify areas in which this is the case.

This report is the result of the deliberations of the Expert Group (EG). First, it describes the creation of a database of research facilities by the EG which currently lists more than 700 entries from laboratories around the world (but with a focus on those in OECD countries). The database was released for public use in the spring of 2008; further details are given in Chapter 2.

The report also contains a detailed consideration of the areas regarded by the Expert Group as particularly requiring attention. These are presented in a series of sub-sections of Chapter 3:

- nuclear data;
- reactor development;
- neutron applications (including neutron scattering);
- ADS and transmutation systems;
- fuel;
- materials;
- safety;
- nuclear and radiochemistry research;

A miscellaneous heading is also used, which covers issues such as nuclear process heat for hydrogen production. Each section has its own conclusions and recommendations.

Clearly, it is impossible to claim completeness; the scope of the NSC request was broad and our membership and time available was, of necessity, limited. The report is therefore built around the background and expertise of the experts who volunteered their time, while attempting to be as comprehensive as possible.

Equally, since the remit of the Expert Group's considerations (and thus this report) is focused on the interests of the NEA Nuclear Science Committee, there has been no consideration of facilities entirely devoted to issues such as fusion or waste disposal. In the same way, because the mandate was related to *facilities*, systematic consideration of *techniques of measurement* has not been undertaken, although reference to measurement methods is made throughout the report.

Chapter 4 provides a convenient guide to NEA activities which are specifically related to the present study. Therefore, there is discussion of activities such as the CSNI SFEAR report (NEA, 2007d), the IRPhE (NEA, 2008r), the International Criticality Safety Benchmark Evaluation Project (ICSBEP) (INL, 2008) and the Shielding INtegral Benchmark Archive Database (SINBAD) (NEA, 2008pp) databases, but the chapter should not be regarded as a totally comprehensive review of all of the work of NEA.

Finally, Chapter 5 brings together a number of conclusions and a series of recommendations. It should be noted that the Expert Group draws its membership from a wide variety of backgrounds and as a result there has naturally been some disparity of views; however, the conclusions describe a consensus viewpoint inasmuch as possible.

Chapter 2: Review of status and needs of facilities in nuclear science: Creation of a database

The mandate for the current project called for the establishment of a database of research and test facilities for R&D in the field of nuclear science and technology (see Appendix A). This chapter describes the creation and development of that database and its current representation in a web-based format. The process of confirming with facility owners that they are content for their facility to be included in the database is also described. Chapter 3, which follows, describes the results of the review implicit in constructing the database and the analysis of the needs of the nuclear science community derived from the database and other sources of information.

2.1 Construction of the database

A start to data collection was made with a simple Excel spreadsheet format as this was a format easily readable by members of the Expert Group. However, this was later converted into a formal database format for wider access through the NEA Internet site. For the purposes of the remainder of this report the present web-based database is referred to as the Research and Test Facilities Database (RTFDB).

An initial "template" defined certain items of information to be collected for each facility and the first input to the database was information supplied on the facilities mentioned in the previous NSC report, *Research and Development Needs for Current and Future Nuclear Energy Systems* (NEA, 2003).

Further entries were added from information derived from a number of sources:

- from members of the Expert Group themselves, plus information solicited by members of the Expert Group from other scientists in their countries;
- from the Nuclear Centres of Competence Database (NuCoC) database (JRC, 2008);
- from the NEA/CSNI Senior Group of Experts on Nuclear Safety Research (SESAR), *Report on Support Facilities for Existing and Advanced Reactors* (SFEAR report) (NEA, 2007d). As many as possible of the facilities listed in the report have been entered into the database, but see Chapters 3.7 and 4.1 for a more complete discussion of the SFEAR report and its contents;
- from the IAEA databases on:
  - fast reactors (IAEA, 2008a);
  - ADS systems (IAEA, 2008);
- from the Nuclear Physics European Collaboration Committee Handbook (NuPECC, 2004);
- from the NEA report on lead-bismuth eutectics (LBE) (NEA, 2007);
- from the NEA Nuclear Science Committee 3rd Information Exchange Meeting on the Nuclear Production of Hydrogen (NEA, 2006);
- from the members of the NEA Nuclear Science Committee.

In addition, data have been obtained through a review of the Internet for information contained in the websites of research institutes. For research reactors, in particular, a check has been made for further information available on the web where names and locations of reactors have been established using the IAEA Research Reactors Database (IAEA, 2008e).

Other resources reviewed were:

- The European Radiation Dosimetry Group (EURADOS) Irradiation Facilities website (EURADOS, 2008) has information on facilities in the European Union (EU) directly linked to its interests in dosimetry, radiation protection, radiobiology, radiation therapy and medical diagnosis.
- The International Union of Pure and Applied Physics (IUPAP), which has recently published *A Worldwide Overview of Research Facilities in Nuclear Physics* (IUPAP, 2006).

While some of the facilities listed in these other resources have been included in the RTFDB database constructed by the Expert Group, it should be understood that this does not apply to all of them because some stray beyond the interests of the NEA Nuclear Science Committee and hence the remit of the Expert Group's considerations. For example no facilities entirely devoted to fusion or waste disposal have been included.

Equally, it should not be interpreted that all facilities of the types listed in the database are necessarily present – a major consideration has been to attempt to cover OECD countries as opposed to being all inclusive world wide.

2.2 The web-based version of the database

During 2006 a subsidiary project commenced development of a web-based version of the database that could be accessed over the Internet and, following a verification process, it was released for public use in the spring of 2008 (NEA, 2008ss) with over 700 facilities listed.

The following description applies to the public access to the database; however, there is also a mechanism for access to a working area that allows a reviewer (*e.g.* a staff member who is the contact point for a particular facility) to interrogate the full information held on that facility. Section 2.3 describes the checking of the database and the information which is displayed for records where the verification process is incomplete.

Following the "Guest" login at the URL *www.nea.fr/rtfdb/*, a user is taken to the "Top" level screen from which they have access to: *i)* the Search facility; *ii)* the User Manual (in .pdf format); *iii)* a "Directory" of keywords which permits an alternative way to access the database information.

2.2.1 The "Search" facility

Figure 1 shows the "Search" screen from which it can be seen that selections can be made of various parameters such as the country, facility type (reactor, accelerator, etc.), application (ADS, fuel research, etc.) or the organisations owning facilities. More information on the facility type and application keywords is given in Section 2.2.2.

A search is performed by typing keywords into the "Select Word(s)" section for a full text search in all database fields, or by selecting keywords from the "Select Condition(s)" section where pull-down lists are available in the six selection areas. It is possible to make selections from both the "Select Word(s)" and "Select Condition(s)" sections and, logically, these selections are "AND" combined. A brief set of instructions is included in the "Help" section of the Search page, but more detailed setting information, with examples, is described in the User Manual which is available on line after logging into the RTFDB.

Figure 2 shows a sample output for a search, in this case for "Country: Japan" and "Application: ADS".

Clicking on "Details" for one of these entries leads to a screen showing the information available for that entry as shown in the example in Figure 3.

Figure 1: RTFDB search facility

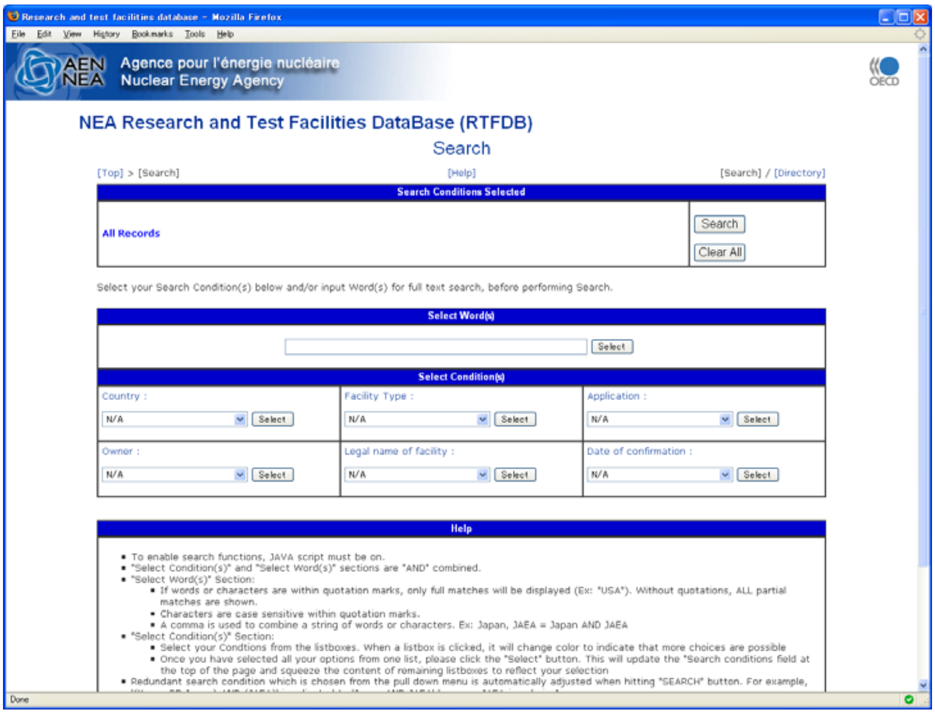


Figure 2: RTFDB search facility – search results

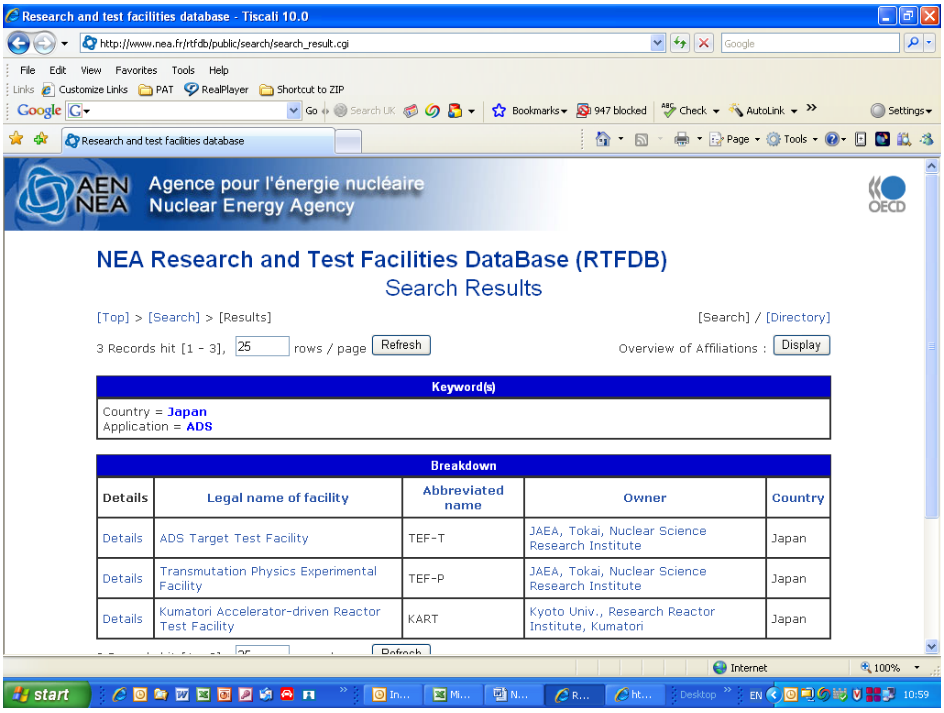


Figure 3: RTFDB search facility – search results


AEN NEA Agence pour l'énergie nucléaire Nuclear Energy Agency

OECD

NEA Research and Test Facilities DataBase (RTFDB)

Record Details

[Top] > [Search] > [Results] > [Details] [Search] / [Directory]

Record Details	
Record No.	367
Legal name of facility	ADS Target Test Facility
Abbreviated name	TEF-T
Facility Type	Laboratory Facility
Application	ADS
Current purpose	Material irradiation and feasibility demonstration for spallation target for ADS
Status	Conceptual Design Phase
NEA Programme Cross Reference	
Other Cross Reference	
Owner	JAEA, Tokai, Nuclear Science Research Institute
Homepage	http://j-parc.jp/Transmutation/en/ads.html
Division	Nuclear Science and Engineering Directorate
Country	Japan
Address of facility	2-4 Shirakata Shirane, Tokai-mura, Naka-gun, Ibaraki-gen 319-1195, Japan
Keywords	ADS; Transmutation; MA; LLFP; Lead-bismuth; Spallation target; Proton; Irradiation; Neutron
Company Use Only or National or International	National
Local Language Legal name of facility	ADSターゲット試験施設
Local Language Abbreviated name	TEF-T

One particular task has been to try to identify a reasonable Internet website reference for each facility. This is the "Homepage" entry in the output. In some cases the facilities do have their own web address, but in many cases they do not; equally there is no commonality over there being both original language and English URLs available. A priority order was therefore adopted:

- if nothing other than original language was available, that was used;
- if both original language and English URLs exist and there are easy links to English from original language, then the former is quoted;
- if both exist but there is not an easy link from the original language to the English version URL, then the English version URL is quoted;
- if no sensible direct web address exists, one that at least refers to the facility has been used – for example a .pdf document or a paper which is available on the web.

2.2.2 The "Directory" facility

The following keywords are accessible by clicking on the "Directory" button on the "Top" level screen.

- Country
- Facility Type
- Application
- Owner
- NEA Programme Cross Reference

The first four are the same as in the "Select Condition(s)" section of the "Search" facility. By default, selecting the "Directory" facility will produce a list of countries and clicking on a country name will produce a list of all the facilities for that country contained in RTFDB. A listing of the "Details" of any of those facilities can be obtained in the same way as described in the previous section. Equally, selecting "Owner" leads to a list of organisations which have their facilities listed in RTFDB.

In order to simplify searching of the database "Facility Type" and "Application" keywords have been assigned to all the facilities. These are relatively broad brush allocations and sometimes multiple allocations are used where a facility spans more than one type.

The "Facility Type" category attempts to describe the *physical form* of the facility and uses the following seven terms:

- Accelerator
- Irradiation Facility
- Laboratory Facility
- Non-reactor Based Instrument
- Radioactive Material Handling Facility
- Reactor Instrument
- Reactor, Critical Assembly or Subcritical Assembly

Selecting the "Facility Type" Directory provides some further information on the types allocated to each of these categories, *e.g.* "Glove Box, Hot Cells, Shielded Cave Facility, Shielded Facility" all appear under "Radioactive Material Handling Facility".

In contrast, the "Application" directory lists ten categories which describe the *use* to which the facility is put.

- ADS
- Accelerator Based Application
- Fuel Research
- Materials Research
- Neutron Application
- Nuclear Data Measurement
- Nuclear Heat Application
- Nuclear Safety
- Nuclear and Radiochemistry Research
- Reactor Development

Where a facility is known to have been or is being used for a particular NEA programme, the details for that facility include a hyperlink to the information on that NEA programme. Hence, the final category in the directory listing: "NEA Programme Cross Reference" provides a quick route to identifying those facilities which are listed as being related to the various programmes mentioned.

2.3 Checking of the database

Because the database is accessible by the public, the Expert Group regarded it as important that the information contained therein should be checked and agreed with the facility owners. As a result, prior to the release of RTFDB to the public in the spring of 2008, the database was sent to a number of representatives who were asked to contact the individual facilities and confirm with them that the entries were acceptable. The following contacts were used for this activity:

i) an Expert Group member for those countries where there was representation on the EG;

ii) a member of the NSC for the other countries not covered by the EG membership;

iii) direct contacts identified by the NEA for the remaining countries.

When an entry has been confirmed, the date of the check was entered in the database and is visible to users.

As at the point of release to the public in the spring of 2008, checking has been recorded for about 80% of the facilities. For those where checking has not yet been recorded, a more limited amount of information is displayed and its Record Status is set to "RECORD NOT YET CHECKED". These unchecked records show only the general information below, which has been assigned initially by the Expert Group:

- Legal Name of Facility
- Abbreviated Name
- NEA Programme Cross Reference
- Other Cross Reference
- Homepage
- Country

Other provisional information (*e.g.* Owner, Application) is held in a working area of RTFDB and *cannot* be accessed by public users.

Offers of further entries to the database or confirmation of the information held would be welcomed as it is anticipated that the resource will be of continuing and evolving interest to the nuclear science community.

Finally, it should be clearly understood that the existence of a facility against a particular category should *not* be interpreted as being sufficient. The Expert Group is firmly of the view that the ability to be able to compare and contrast is most important, in the same way that the development of Monte Carlo codes has been aided by having more than one code in existence. Diversity of provision is an important feature that can assist in clarifying, for example, systematic effects that can be important in certain measurements.

2.4 Conclusions and recommendations – RTFDB

The RTFDB database has been successfully created and published on the Internet. The Expert Group believes that it has been a most useful tool during the present review. The EG also believes it already forms a valuable resource for the scientific community world wide and therefore encourages the NSC to continue to update and expand the RTFDB in the future.

Chapter 3: Review of status and needs of facilities in nuclear science: Results

In seeking to develop a logical structure for this review, the Expert Group has considered the classification of the facilities in the RTFDB database. In order best to identify and analyse the needs for such facilities, it has been decided to order the discussion based on applications, or groups of applications, rather than on particular facility types. Thus applications consider topics such as production of nuclear data, reactor development and ADS. As a result, consideration of particular physical facility types, such as accelerators, reactors and glove boxes, will be found distributed in the sections below as the applications for which they are used are discussed.

For consistency, in the RTFDB database described in Chapter 2, the information on "Application" has been listed against the same categories as have been used in the subsections below. However, the RTFDB also lists the physical facility type and this ought to fulfil the aim of the Expert Group of providing a means of cross reference between applications and facility types in the RTFDB.

3.1 Nuclear data

The RTFDB database includes many facilities devoted to nuclear data measurements. These facilities are of several different types such as accelerators, reactors, etc., and they lead to the production of experimental nuclear data, which are later used as inputs to models, theories or codes, generally after an analysis and evaluation process. The evaluated data (or subsets of those) are of interest to nuclear physics databases, astrophysics models, neutron physics codes, nuclear energy and fuel cycle studies, radiation shielding studies, medical applications, etc.

It is worth mentioning that, although some facilities were designed and built originally for the special purpose of measurement of nuclear data, at many installations only a small fraction of the total beam time is presently devoted to this purpose; in other words, nuclear data applications are now having to compete for: *i)* a smaller number of facilities, and *ii)* limited beam time.

In 2001 the NSC initiated a study on R&D needs in nuclear science and on 6-8 November 2002 a Workshop on R&D Needs for Current and Future Energy Systems was held in Paris (NEA, 2003). The availability of facilities to measure nuclear data – either basic (microscopic) data on nuclear reactions, or integral data – was underlined to be crucial for the different projects maintaining evaluated nuclear data files, co-ordinated by the Working Party on Evaluation Co-operation (WPEC) of the NSC (NEA, 2008D) and a short survey on facilities for nuclear data measurements in different countries was included in the report (NEA, 2003). It was noted that the number of facilities still operational for such measurements had dwindled to reach critically low levels.

Experimental data are kept in the EXFOR format, in the eponymous database (NEA, 2008h). Before being further used, these data need to be submitted to a QA procedure. In this evaluation process the experimental data are complemented with model calculations for reactions not accessible experimentally, in order to arrive at a complete database. Consistency of the data is also ensured (*e.g.* summation of partial cross-sections versus total cross-section measurements, sum of isotopic data versus elemental data, etc.). Very importantly, the performance of the database is tested by comparison with the results of dedicated benchmark experiments. The detailed requirements for new measurements are collected in a High Priority Request List (HPRL) (NEA, 2008w) which is organised by WPEC (NEA, 2008D) and maintained by the NEA Data Bank. (More discussion on the data libraries available internationally is provided in Section 3.1.2.)

Different types of facilities for nuclear data measurements may be distinguished:

- The first category consists of facilities for measurement of microscopic (or "differential") nuclear data. Examples for such measurements are neutron cross-section measurements of specific nuclear reactions as a function of the incident neutron energy, decay data, fission yields, fission neutron multiplicities and spectra, and isotopic half-lives. Depending on the specific purpose, the requirements on the facility and on the equipment are very different. All the facilities in this category contribute to establishing the experimental basis for the development of model codes and adjustment of model parameters (evaluation process). They also provide information for the basic physics understanding of nuclear reaction mechanisms and nuclear structure, *e.g.* for the Evaluated Nuclear Structure Data File (ENSDF) (BNL, 2008).
- A second, equally important, category consists of facilities for measurement of integral (or semi-integral) data. Examples are measurements of isotopic reaction rates in critical assemblies using activation detectors or fission chambers, averaged over the incident neutron spectrum, hence the qualifier "integral". Such measurements are often considered as data for an *a posteriori* validation of evaluated files derived from nuclear models and differential data alone, but they can equally be seen as providing a different, complementary form of experimental information useful to a global data evaluation process. As different nuclides and reactions generally contribute to a given integral measurement, a crucial aspect in the global analysis is to account properly for the individual contributions. This is done by means of sensitivity calculations. Such sensitivity calculations also serve to guide the setting of priorities for new experimental programmes.

Both types of measurements are useful to evaluators in practice, since the desired information cannot be obtained from one type of experiment alone. Integral data are essential for pinpointing problems, for validation purposes, and to improve the accuracy of the evaluated data library. Specific, well-designed integral experiments provide very accurate spectrum-average information, which evaluators can use in combination with differential measurements in order to infer improved fits (reduced uncertainties), and hence improved evaluated data files.

The WPEC is itself essential in the co-ordination of the whole data evaluation process at an international level, providing the framework for concerted actions and monitoring and maintaining the High Priority Request List for Nuclear Data.

It is worth noting that the improvements in the nuclear data used as inputs to neutronics codes have assisted the considerable progress that has been achieved in the use of these codes over the last two decades alongside the use of more sophisticated models and algorithms, validation and benchmarking activities, and improved computer power.

In validation, the process tries to separate different phenomena as much as possible, and to validate each modelling step individually. Thus, more diverse and analytical experiments are required these days which are aimed at characterising each individual phenomenon. This is in contrast to the single global experiments or mock-ups commonly performed in the past in which different phenomena were coupled in a complicated fashion. Today, sensitivity studies are used to identify error trends in the data or models and to suggest improvements to models and codes.

As a result of this multi-year effort it can be said that modern neutronics codes (and their accompanying nuclear data) are capable of reliable and accurate predictions of conventional light water reactor (LWR) and fast reactor (FR) core neutronic performance. As a consequence, more reliance is now being placed on code predictions for investigating changes in sub-assembly or core design and even in innovative reactor studies. It is therefore essential that reliable estimates of uncertainties be provided with the code predictions so that the user can decide whether the risks associated with the proposed changes are acceptable or not. Clearly the quality of neutronics code predictions depends upon the quality of the input nuclear data. In spite of considerable progress in this area, significant biases and uncertainties do remain in the basic evaluated nuclear data files. As a result, uncertainties can grow rapidly when the code is used outside its domain of experimental validity, a point sometimes not adequately appreciated by users.

3.1.1 Facilities for measurements of nuclear data

Concerning the requirements for the measurement of microscopic data, it must be remembered that a large variety of different nuclear reactions and many different specific nuclear properties have to be considered in order to meet the needs of the applications. Neutron-induced reactions are certainly of highest priority, but reactions induced by light-charged particles and photonuclear reactions are most important for some applications. The requests for completion of the nuclear database with data on such reactions lead automatically to the need for incorporation in the evaluation process of other facilities, which had not been in close contact to the fission-oriented community in the past. A recent example can be seen in the efforts towards the development of accelerator-driven systems for the transmutation of long-lived nuclear waste, which profited from the capabilities at the GSI laboratory for relativistic heavy ions in Darmstadt, in providing accurate yields of spallation products using the technique of inverse reactions. (Further discussion on transmutation can be found in Section 3.4.)

To a large extent, the same comments can be made about experimental reactors and critical facilities capable of providing integral measurements. Many such facilities have been shutdown since the 1980s and have not been replaced by new ones.

As noted earlier, except for a few facilities built explicitly for nuclear data measurements, most experiments are presently carried out at facilities used for a broad range of applications, not necessarily related to nuclear fission or fusion. In fact, many facilities are only used for a relatively small fraction of their beam time for measurements of nuclear data for nuclear energy applications. The much-reduced public acceptance of nuclear energy in the late 1980s and in the 1990s led inevitably to a reduced interest of young scientists to work in this field. There is a risk of loss of expertise for the future, which could *not* be recovered quickly and easily when needed.

On the other hand, the physics problems addressed in the modelling of nuclear reactions and the development of advanced new measurement techniques are demanding and interesting, providing an excellent basis for scientific education. One of the major challenges in the near future will be to integrate more effectively scientific education at universities with the requirements of industry, for example through the support of the use of the available facilities by external users. A closer interlink with academia and other user communities would be of great advantage for the whole process of establishing an evaluated and well-validated database for nuclear energy applications.

The following categories of facilities may be distinguished:

- *Reactors and critical assemblies:* Critical assemblies are research facilities which can sustain a fission chain reaction and which operate at low power and near room temperature. Examples include VENUS (SCK•CEN Mol) (Baeten, 2008), EOLE, MASURCA and MINERVE (CEA Cadarache) (Fougeras, 2005, 2007), PROTEUS (PSI) (PSI, 2008a) and FCA (JAEA Tokai) (JAEA, 2008b). MINERVE, for example, uses a highly accurate sample oscillation technique. Further details on these facilities are provided in Section 3.2, *Reactor Development*. The core and fuel are readily accessible and can be easily modified, repositioned and instrumented for investigating various configurations, *e.g.* by measuring neutron spectra. In a (power) reactor, the core configurations are generally much more complex and there is considerably less flexibility for introducing measurement devices and performing “clean” physics measurements; on the other hand, the much higher neutron flux level makes it possible to investigate important phenomena such as fuel burn-out, isotopic build-up, reactivity loss with burn-up, etc., which are not accessible in a zero-power critical facility. The availability of both types of facilities is therefore essential for obtaining the integral nuclear data which are needed.

  Power reactors and critical facilities are also discussed in Section 3.2, in connection with reactor development in general. Indeed, these facilities are also required to obtain experimental evidence for benchmarking methods, validating specific sub-assembly concepts, core configurations, design options, safety-related parameters, etc., in representative conditions (such as temperature, neutron and gamma flux, and coolant chemistry), together with the corresponding calculation procedures.

  Measurements made in critical facilities include critical mass, spectral indices, reaction rate traverses, reactivity effects of various types (substitutions, voids, and sample oscillations), and gamma heating. In principle, these can be repeated for a large variety of neutron spectra.

In power reactors, post-irradiation examination of fuel rods or separated isotope samples provide extremely valuable data on neutron capture reactions.

Concerning isotopic data, the stable neutron flux at some nuclear reactors is convenient for measuring reactions in an energy range where they are important for nuclear energy applications. A well established measurement technique is neutron activation analysis, either using off-line analysis of the activated products, or prompt-gamma spectroscopy at an external beam line. The method is widely used for trace element analysis for many different applications. High flux reactors [such as at the Institut Laue-Langevin (ILL, 2008)] are valuable tools for investigating, in particular, fission yields using on-line mass spectrometry, cross-sections of short-lived isotopes that are bred *in situ* (Marie, 2006) and for neutron capture gamma ray spectroscopy. At present, LOHENGRIN (Armbruster, 1976) is the only operating fission fragment separator for the precise measurement of fission product yields. Since the neutron spectrum in the in-pile position is purely thermal, only thermal neutron-induced fission can be studied. Unfortunately, no recoil separator exists world wide where fast neutron-induced fission product yields could be measured with similar precision.

High-precision cross-section measurements at thermal energies provide, in many cases, an important constraint for assessment of measurements at higher energies.

- *Neutron time-of-flight facilities:* Due to the discrete energies of intermediate states populated in neutron-induced reactions, neutron cross-sections exhibit a pronounced resonance structure and diminishing level spacing with increasing excitation energy. This makes it impossible to extrapolate the nearly 1/v behaviour of the cross-section at thermal energies or to model the detailed resonance structure, except for some average values. Whenever a detailed knowledge of the resonance structure is required (*e.g.* for self-shielding calculations) an experimental determination of the cross-section is essential, and this can only be achieved at facilities providing the necessary good energy resolution for resolving the individual resonances.

  The only practical possibility for such high resolution cross-section measurements is the time-of-flight method using pulsed neutron beams. There are few such facilities left today. They are located at Oak Ridge, USA [Oak Ridge Electron Linear Accelerator (ORELA)] (ORNL, 2008a), Geel, Belgium [Geel Electron Linear Accelerator (GELINA)] (IRMM, 2008) and Kyoto University Research Reactor Institute (KURRI, 2003) and use pulsed electron accelerators and photo-neutron conversion targets. Other time-of-flight (TOF) facilities of this type have recently been installed at Rensselaer Polytechnic Institute (RPI, 2008) and at the Pohang Laboratory in Korea (Kim, 2002, 2003). The energy range extends typically from sub-thermal energies up to about 20 MeV, thus covering the main energy range of interest for nuclear energy production. Time-of-flight facilities using high-energy proton beams and spallation targets have been installed at Los Alamos National Laboratory, USA (LANL, 2008a) and at the European Organisation for Nuclear Research in Geneva, Switzerland (CERN, 2008). At these facilities, the useful neutron energy may extend to several hundred MeV, depending on the energy of the primary protons (800 MeV at LANL, 20 GeV at n_TOF).

  TOF facilities using photoneutron converters profit from an excellent time resolution, but the neutrons are emitted quasi-isotropically, thus limiting the useful flux for experiments, especially at large distances where the best energy resolution is achieved. Spallation targets at high-energy proton beams have the advantage of much higher, focused neutron beams, but less time resolution and higher background contributions from high-energy secondary particles. A peculiarity of the n_TOF facility at CERN is the very high instantaneous flux concentrated in well separated pulses (pulse separation 2.4 s) which makes this facility especially suited for measurements on small radioactive samples. Total-absorption (*i.e.* transmission) measurements are, however, not possible due to the high instantaneous flux. Once again, this example illustrates the fact that different facilities are generally needed to obtain the necessary data with the required accuracy.

- *Charged-particle accelerators:* RTFDB contains a large number of charged-particle accelerators in a wide energy range. At low energies up to a few MeV electrostatic accelerators such as Van de Graaff accelerators, Tandems or Singletrons are frequently used, while at higher energies several cyclotrons are operational.

Besides the direct measurement of reactions induced by charged particles, such facilities are often used as sources of quasi-monoenergetic neutrons using suitable neutron converters.

– *Charged-particle induced reactions*: Compared to the database for neutron-induced reactions, the database for nuclear reactions induced by charged particles is much less complete. In the past, most requests for experimental data on reactions induced by light charged particles came from medical applications, dosimetry and for shielding purposes. New efforts in the field of fusion energy may increase the demand in the near future.
– *Quasi-monoenergetic neutron-induced reactions*: Typical applications are measurements in the so-called average energy domain, where the spacing of the resonances is less than the resonance widths. At these energies, the cross-sections exhibit a smooth energy dependence. For example, excitation curves for many neutron-induced reactions, such as (n, α), (n,p), can be measured conveniently with quasi-monoenergetic neutron beams at a few energy points and interpolated using appropriate nuclear models. Often, these reactions result in radioactive products and the cross-sections can be measured using the activation technique.

Due to the reaction thresholds, such excitation curves extend very often to high energies beyond the energy range accessible with electrostatic accelerators. In such cases, measurement capabilities at cyclotrons are necessary. Unfortunately, in the recent past such experiments at cyclotrons have become scarce and there is a lack of accurate measurements at energies beyond 20 MeV for a better adjustment of the model parameters for many reaction channels with high threshold energies.

Pulsing of the primary beams allows the investigation of time dependent processes such as the population of precursors for delayed neutron emission after fission, and decay of isomer states.

An experimental technique which has recently generated renewed interest (Cramer, 1970; Escher, 2005; Jurado, 2007) is the measurement of neutron-induced reactions not accessible directly (*e.g.* due to the short half life of the target) using transfer reactions leading to the same intermediate nucleus. This so-called surrogate method is an important technique complementing the efforts for direct measurements on highly radioactive targets, especially minor actinides.

Looking to the future, the ambitious FAIR/ELISE experimental project at Gesellschaft für Schwerionenforschung (GSI, 2008), will be dedicated to fission fragments and emitted neutron measurements and could potentially cover equivalent incident neutron energies of, say, 5 to 35 MeV. Another contribution will come from the SPIRAL2/NFS[3] project at the Grand Accélérateur National d'Ions Lourds (GANIL, 2004), covering the 1 to 15 MeV incident neutron energy range, although it is not yet clear how the isotopic separation will be done there.

A recurrent problem with nuclear data measurements is the difficulty in obtaining appropriate isotopic samples of (pure) raw materials, to be used as targets for the incident neutrons. This problem is exacerbated in the case of radioactive samples, minor actinides, some fission products, etc. This is an important point, as it demonstrates that having a good facility is not entirely sufficient in order to generate measured data; target materials are also needed. Thus specialised chemistry laboratories, equipment and staff are also required. For transmission measurements, the problem is even more acute as the masses of samples required are fairly high.

3.1.2 Nuclear data files and centres

The results from the measurements of nuclear parameters using the kinds of facility described above are crucial to the projects which *evaluate* nuclear data. Since it is through these evaluated nuclear data libraries that most users make use of the outputs of the experimental facilities it is worth making a few points about these libraries here. They are maintained at several data centres in OECD as well as non-OECD countries, and for ease of use all the libraries obey the same standard format. Much of the work is done in collaboration between contributing countries and evaluation efforts are co-ordinated by WPEC (NEA, 2008D), which organises annual meetings of major evaluation projects for an exchange of information on nuclear data evaluations, measurements, model calculations, and data evaluation and validation. The IAEA also organises annual technical co-ordination meetings of nuclear reaction data centres, and NEA participates in these.

3. SPIRAL: Système de production d'ions radioactifs accélérés en ligne.

In Europe, the Joint Evaluated Fission and Fusion (JEFF) project of evaluated nuclear data, (NEA, 2008s), regularly updates a state-of-the-art general-purpose file for routine applications in the various areas of science and technology and libraries for special applications. The JEFF project co-operates closely with other evaluation projects: in Japan [Japanese Evaluated Nuclear Data Library (JENDL)] (JAEA, 2008d), the United States [Evaluated Nuclear Data File (ENDF/B) managed by the Cross-section Evaluation Working Group (CSEWG)] (NNDC, 2008), and non-OECD member countries [BROND in Russia (IPPE, 2008), Chinese Evaluated Nuclear Data Library (CENDL) in China (CNDC, 1991)]. Through the participation of the Nuclear Data Services of the IAEA (2008c) co-operation with non-OECD member countries is ensured [*e.g.* with the Fusion Evaluated Nuclear Data Library (FENDL) (IAEA, 2004)].

Of course, crucial to the availability of nuclear data to the wider community, are the Nuclear Data Centres which share and distribute the data derived from the experimental work undertaken on the physical facilities. Databases such as EXFOR (NEA, 2008h), ENSDF (BNL, 2008) are essential to make the results available to all. Principal in this role are the NEA Data Bank (NEA, 2008g), the National Nuclear Data Center (NNDC) at Brookhaven (NNDC, 2008a), Institute of Physics and Power Engineering (IPPE) at Obninsk (IPPE, 2008a), and the IAEA Nuclear Data Section (IAEA, 2008c) which are the "core" nuclear data centres in the Nuclear Reaction Data Centres Network (NRDC), which is a worldwide co-operation under the auspices of the IAEA (2008d). Other specialised nuclear data centres in countries such as Japan, China, Korea and Hungary complement the core centres with their responsibility for data of a specialised type or application; for details see (IAEA, 2008d).

3.1.3 Recent trends in nuclear data facilities

In the late 1990s a report was prepared by Rowlands and Bioux (1996) which reviewed the status of measurement facilities and concluded that both facilities and expertise in measurement and evaluation were at a critically low level. There is relatively little that has happened in the interim to reverse this trend.

A recent paper by Salvatores (2006) presented at the Workshop on Nuclear Physics and Related Computational Science R&D for Advanced Fuel Cycle, reviewed the current state of nuclear data availability, including particular discussion on covariance data, for the specific advanced fuel cycle topic. In summarising, Salvatores concluded that the tight interconnection of basic sciences, applied physics, engineering and industry that existed in the early days of the applications of nuclear technology has to be reconstructed to meet the new requirements and challenges of recent fields such as advanced fuel cycles. His view was that the boundary conditions to start this endeavour look favourable.

In terms of actual facilities, there is perhaps the perception in Europe that the worsening of the situation reported by the previous project (NEA, 2003) and by Rowlands and Bioux (1996) has stabilised. However, the availability of trained and skilled personnel for:

- the maintenance and operation of the facilities;
- experimental measurement;
- evaluation;

is still problematic.

[Regarding appropriate provision of qualified human resources, see the recent OECD/NEA Press Release (NEA, 2007a).]

One concern is that nuclear data activities only represent a small percentage of the work on some facilities and, with other pressures, the availability for nuclear data measurement can be assigned low priority, or even be lost.

Equally, the number of people around the world involved in evaluating nuclear data for energy applications is very small and the amount of direct industry funding of such activities is also low.

In the coming years, it can be expected that demands for new and improved data will come from three major areas linked to nuclear energy production:

- *Operational safety of nuclear power plants*. Considerations for lifetime extension of presently operating Gen. II reactors placed challenging demands on the accuracy of the models and the underlying nuclear database. This also applies to the Gen. III reactors which are under development or construction. The co-ordinated efforts of the nuclear data community in the past decades resulted in well-qualified databases which give a solid basis for such calculations. There is continuous effort to improve the performance of the evaluated databases, to determine uncertainties (currently largely missing in the files) and to unveil deficiencies while seeking for solutions. The facilities which are presently operating emerged from these efforts and the data needs can be handled, provided that the present level of expertise and facilities is maintained.

- *The nuclear waste issue*. The public acceptance of nuclear energy is strongly linked to the solution of the question of how to handle nuclear waste. Considerable efforts have been devoted to the partitioning and transmutation of high-level radioactive waste. Current activities involve the design of accelerator-driven systems for transmutation and the development of schemes for minor actinide burning in fast reactors, such as foreseen in several Gen. IV systems (see Section 3.4). These developments led to new demands on data for certain reaction channels and in energy ranges which have been less important in the past. Tackling the experimental challenges has commenced and measurements in these energy ranges are ongoing, largely within an extended international co-operation. Priorities for further efforts in these directions and for development of new experimental facilities, if needed, will very much depend on the outcome of ongoing development studies for new systems and operation experience with prototype test facilities. However, past experience tells us that it is wise not to allow everything to be needs-driven. Much progress was made in the past from generic progress in measurement techniques, theory and model developments, and basic research, a comment that applies to other applications of nuclear data and also to other aspects of nuclear research and use of facilities.

- *Fusion energy*. While studies which are strictly related just to fusion are not the subject of this present Expert Group task, this report and its related database, it is worth making the following comments since in the area of nuclear data there is significant overlap with nuclear technology related issues. The fusion community has been very much involved in the nuclear data evaluation process for many years, contributing to the improvement of relevant databases [*e.g.* the European Fusion File (EFF) which has been merged into the earlier JEF database to form the Joint Evaluated Fission and Fusion File (JEFF) maintained by the NEA Data Bank (NEA, 2008s), or the Fusion Evaluated Nuclear Data Library (FENDL) (IAEA, 2004) maintained by the IAEA-NDS (IAEA, 2008c)]. Because of the need in the fusion community for accurate data primarily in the 14-MeV energy range, these efforts particularly contributed to the improvement of the activation data files. Construction of the International Thermonuclear Experimental Reactor (ITER, 2008) and plans for building the International Fusion Materials Irradiation Facility (IFMIF) (ENEA, 2008) will lead to an increase in needs for nuclear data up to 50 MeV, an energy range where the databases are much less complete than at energies below 20 MeV. Facilities for measurements with quasi-monoenergetic neutrons in this energy range are scarce and it would require major efforts to provide the necessary manpower and re-install suitable equipment for such measurements.

It is worth noting at this stage that some of the facilities of interest to the current Expert Group activity are also of importance to the Nuclear Physics European Collaboration Committee (NuPECC);[4] see their website for the current "roadmap" (NuPECC, 2005). It must be emphasised, however, that the focus and priorities of NuPECC do not necessarily match those of the NEA Nuclear Science Committee.

3.1.4 Conclusion and recommendations – nuclear data

A quick reading of the list of facilities related to nuclear data measurement such as is contained in the RTFDB may give an erroneous impression of a "luxurious" infrastructure for experiments. What is

4. NuPECC is an Associated Committee of the European Science Foundation (ESF). The objective of NuPECC is to strengthen European Collaboration in nuclear science through the promotion of nuclear physics and its trans-disciplinary use and application in collaborative ventures between research groups within Europe and particularly those from countries linked to the ESF.

important is the availability of modern facilities able to provide results for materials of current technological interest and at the level of accuracy required for present applications. It is also important that some of these facilities be capable of handling active materials. Viewed in this light, the number of appropriate facilities is much more limited.

A pair of general recommendations relate to the maintenance of:

- expertise;
- infrastructures and samples.

Section 3.1.3 above attempted to point out cases in which new facilities (and expertise) may become necessary in the future.

In relation to expertise, it is worth noting that the NEA Steering Committee for Nuclear Energy, in making its recent statement and recommendations regarding a government role in ensuring qualified human resources in the nuclear field, has served to emphasise the skills issue (NEA, 2007a).

We regard better integration between academia and other user communities as being of great advantage for the whole process of evolving evaluation and validation of databases to the appropriate standard required for nuclear energy applications. The modelling of nuclear reactions and the development of advanced new measurement techniques are challenging and interesting and thus provide an excellent basis for scientific education, while the industry and other end users have definitive requirements for broad range and specific data allied together with reliable estimates of uncertainty. Better use should be made of this symbiosis.

In relation to infrastructure, it should be noted that the situation for facilities for integral measurements is similar to the situation for facilities dedicated to microscopic measurements. (NB *It should be borne in mind that the present consideration does not just cover cross-sections because there are also other "microscopic" data required*.)

The facilities which are presently operating emerged from the efforts which lead to the earlier and current generations of reactors. Equally, the well-qualified databases which gave a solid basis for calculations arose from co-ordinated efforts of the nuclear data community in the past decades. Requirements arising from: *i)* lifetime extension of presently operating reactors; *ii)* evolution of new reactors and their associated fuel cycle infrastructures together with *iii)* the more stringent economical, environmental and safety burdens of the current era put challenging demands on the accuracy of the models and the underlying nuclear database. They will only be met if the present level of expertise and facilities is maintained.

The availability of facilities able to supply the samples required for modern measurements is an important additional requirement alongside the availability of the actual measurement facilities themselves.

3.2 Reactor development

Nuclear power plants continue to seek improved performance through up-rating of power output, the use of higher burn-up fuel, shorter refuelling and maintenance outages, and longer operating cycles. As a result the importance of reactor physics issues continues to be significant, potentially increasingly as new designs such as the Gen. IV systems are being considered.

Reactor configurations are becoming more heterogeneous both in composition and in the distribution of power throughout the core. This makes prediction more difficult when assessing core behaviour and determining safety parameters, such as reactivity coefficients, that dictate transient behaviour. Thus it is evident that experimental validation of neutronics methods continues to be required. In addition, the modern use of advanced computational methods (*e.g.* 3-D neutronics) to refine safety analyses and safety margins has emphasised the need for more detailed reactor physics data and also the experimental confirmation of analytical methods. In addition, thermal-hydraulic and neutronic codes are now being coupled in order to address issues such as boron dilution and anticipated transient without scram (ATWS) or the analysis of pressurised heavy water reactors (PHWRs) and other pressure tube reactors. These further developments have their requirements for benchmarking data as well.

The NEA Committee on the Safety of Nuclear Installations (CSNI) "Support Facilities for Existing and Advanced Reactors (SFEAR)", Report of the Senior Group of Experts on Nuclear Safety Research (SESAR), has also been undertaking a review recently in the reactor safety area and the status of key facilities (NEA, 2007d). In contrast, the current Expert Group project covers facilities of interest to nuclear science and, while there is a mutual interest in reactor physics issues, overall there is a different spread of issues and facilities discussed in the two reports, reflecting the particular interests of the CSNI and the NSC.

3.2.1 Reactor development – the near term

Work in the area of reactor physics of particular importance for the continued development of nuclear power in the nearer term obviously concerns current reactor designs: LWRs, gas-cooled (thermal) reactors, and PHWRs, plus those designs already in development such as the pebble-bed modular reactor (PBMR) and fast reactors.

For these types, typical interests are:

- Reactor core and fuel cycle physics issues at high and very high burn-up and for enrichments higher than currently used in LWRs.
- Moderator and coolant void coefficient in PHWRs and advanced PHWRs (*e.g.* heavy water moderator temperature, density, and poison concentration effects on the safety analysis).
- Physics related to plutonium management and mixed-oxide (MOX) usage in the medium term (before Gen. IV systems are deployed); this applies both to the use of weapons-grade plutonium in LWRs, PHWRs and VVERs[5] as well as to the use of pressurised water reactor (PWR) recycled plutonium. Advanced fuel cycles involving plutonium are also being studied for use in PHWRs.
- Criticality prediction for storage of new and spent fuel particularly for fuels of higher uranium enrichment (*i.e.* > 5%) and different composition (MOX). Experimental data to verify analytical tools will be needed.
- Issues relating to high-temperature reactors including pebble-bed designs. It is worth noting that research for pebble-bed designs extends to consideration of the movement of the pebbles and hence the need to couple to fluid dynamics (use of diffusion theory to tackle this problem is not successful near boundaries). The NEA has a programme "PBMR Coupled Neutronics/ Thermal-hydraulics Transients Benchmark – The PBMR-400 Core Design" (NEA, 2008mm) though the majority of the comparisons are between codes rather than with experiments. See also the comments in Section 3.7.1 on the "RAPHAEL" (ReActor for Process heat, Hydrogen and Electricity generation) Integrated Project associated with the very high-temperature reactor (VHTR) which is designed to supply both electricity and heat for industrial applications.
- Issues relating to current fast reactors; this includes activities like burning of transuranics which is an area of study in both Europe and in relation to GNEP (2008) (further discussion on fuel developments and actinide burning are given in Section 3.5). Examples in Japan include work at JOYO (JAEA, 2008e) on cross-section measurement, core management systems and burn-ups.
- Effects of radiation on reactor internals and vessel at very high burn-up and extended plant lifetime [*e.g.* neutron flux and spectra on the reactor pressure vessel (RPV) internal structures and RPV wall are critical for determining material embrittlement, component lifetime and the potential for RPV failure due to thermal shock while pressurised. This issue also applies to the ageing of pressure tubes in PHWRs]. Further discussion of materials issues will be found in Section 3.6.
- Associated with, but perhaps broader in overall scope than the previous topic, is the whole area of materials testing in reactors and hence the continuing availability of materials test reactors (MTRs) and the facilities that such reactors are able to provide.
- Minor actinide recycling in LWRs.
- Reactor measurements relating to shielding.

5. Vodo-Vodyanoi Energetichesky Reactor (Водо-водяной энергетический реактор) (VVER).

3.2.2 Reactor development – the longer term

Looking to reactor designs being considered in the longer term, such as the six Gen. IV candidate designs (GIF, 2007), the status and current needs for information are:

- *Gas-cooled fast reactors (GFRs)*.[6] This design can put some reliance on the technology already used and under development for high-temperature reactors (HTRs) but as GFR core designs are very different from HTRs, they demand specific R&D so that the extrapolation beyond that foreseen for thermal HTRs can be undertaken. Specific areas under current consideration focus on:

  - *Fuel, other core materials and specific fuel cycle (FCMFC) processes*. This project concentrates on the development of an innovative fuel with high fissile-atom density, able to sustain high temperature, flux and burn-up, together with excellent fission product retention alongside the development of fuel cycle technologies aiming to permit integral recycling of actinides.

  - *Design and safety management (D&SM)*. This project aims at developing a coherent system design (fuel, reactor and cycle options) with a self-generating core and an attractive level of power density, together with a robust safety approach (including the development and validation of computational tools needed for the design and analysis of operating transients within the design basis and beyond). It also promotes the construction and operation of a first experimental reactor – the experimental technology demonstration reactor (ETDR) – which is needed in the performance phase to qualify key technologies (GIF, 2008).

- *Sodium-cooled fast reactors (SFRs)*. The United States is considering the SFR as an advanced recycling reactor under GNEP, designed to consume minor actinides with a breeding ratio considerably less than one. France, on the other hand, is studying innovative SFRs characterised by a positive internal breeding gain and improved safety characteristics. In Japan, future developments will be decided after the Monju restart, though information on the outline plans can be seen in the Japan Atomic Energy Commission (JAEC) "Framework for Nuclear Energy Policy" (JAEC, 2005) and in the Japan Atomic Energy Agency (JAEA) project "Fast Reactor Cycle Technology Development" (FaCT Project) which seeks to materialise innovative technologies based on the R&D results obtained in Monju, etc. (Okazaki, 2007).

  The 6th EURATOM Framework Programme (2002-2006) has supported research on the SFR system through the Specific Support Action EISOFAR (Roadmap for a European Innovative SOdium-cooled FAst Reactor) which was started on 1/02/2007 for a duration of one year with a total budget of €471 139 (EC contribution: €249 021). EISOFAR has led to the preparation of an important Collaborative Project (several millions of Euros) which might be adopted under FP7 (2007-2011).

  On the other hand, EURATOM is supporting the Co-ordination Action Sustainable Nuclear Fission Technology Platform (SNF-TP) (CORDIS, 2008c) which gathers the European nuclear industry, electricity producers and the research community. SNF-TP was launched under FP6 on 1/10/2006 for two years with a total budget of €795 305 (EC contribution: €600 000). Among its tasks, SNF-TP is considering SFR and other fast systems with closed fuel cycles.

  SNF-TP constitutes a key step towards the Sustainable Nuclear Energy Technology Platform (SNE-TP) (CEA, 2008i) which was launched on 21 September 2007. SNE-TP gathers the same stakeholders as SNF-TP. The Vision report of SNE-TP envisages starting the construction of a prototype sodium-cooled fast reactor in the range of 250 to 600 MW(e) in 2012 with this prototype being put into operation in 2020. It would be built through a research-industry partnership, together with a fuel fabrication pilot plant for an estimated overall project cost of about €2 000 000 000.

6. GFR is the acronym used within the Gen. IV project. The gas-cooled fast reactor (GCFR) is a European Commission project supported under the 6th Framework Programme and provides a EURATOM contribution to the Generation IV International Forum "Gas-cooled Fast Reactor (GFR)".

- *Supercritical water-cooled reactors (SCWRs)*. The following critical-path R&D projects have been identified:
  - Define a reference design that meets the Gen. IV requirements plus the design and construction of an in-reactor fuel test loop to qualify the reference fuel design.
  - Significant gaps exist in the heat transfer and safety database for the SCWR. Data needed at prototypical SCWR conditions will be produced.
  - Materials – the main objective is to select key materials for use both in and out of core for the pressure tube and pressure vessel designs.
  - Chemistry – part of the work will require the definition of a reference water chemistry, based on materials' compatibility and the radiolysis behaviour at supercritical conditions.
- *Very high-temperature gas reactors*. As indicated in recent reviews (Khalil, 2004; MacDonald, 2004), many – if not most – of the issues that have to be addressed relate to materials and fuel performance properties, *e.g.*:
  - high-temperature materials;
  - fuel performance and reliability;
  - hydrogen production technologies;
  - safe coupling of reactor and H_2 production facilities;
  - waste generation.

  Computational methods development and validation in the areas of thermal-hydraulics, thermal mechanics, core physics, and chemical transport are further major additional activities. Benchmark tests and code-to-code comparison supported by HTTR (30 MW) tests or by past HTR data (*e.g.* AVR, Fort St. Vrain) will provide some validation. A broad range of normal and abnormal operating analyses will be required. Elimination of unnecessary design conservatisms and improved construction cost estimates should be facilitated by improved computational methods.

  A guide to the relative expenditure predicted for VHTR development is given in a DOE paper (NERAC/GIF, 2002) where fuels and materials and balance of plant account for over 65% of the costs while safety and reactor systems account for 12% and 3% respectively.

  (Progress on the remaining two Gen. IV designs is not at the same stage of development. While "System Arrangements" have been signed by several GIF[7] members for the four designs above, for these last two systems, collaborative R&D is pursued on a provisional basis by interested members.)
- *Lead-cooled fast reactors*. The main ongoing activities by GIF members are:
  - The development of the European Lead-cooled System (ELSY) (Cinotti, 2006); a 600 MW(e) pool-type reactor cooled by pure lead. Under development since September 2006, and sponsored by the 6th Framework Programme of EURATOM, the ELSY project is being undertaken by a consortium of 17 organisations from Europe plus 2 from the Republic of Korea.
  - The development of SSTAR in the United States (Sienicki, 2005), a 20 MW(e) natural circulation pool-type reactor concept with a small shippable reactor vessel.
  - The development of advanced materials for lead-bismuth eutectic (LBE) applications and thermal-hydraulic studies on LBE systems in Japan. For further details on LBE-related research, see Section 3.6.1.2.
- *Molten salt reactors (MSRs)*. Recent studies have confirmed the potential of thermal and fast molten salt reactors for breeding and waste minimisation, while viability analyses have highlighted the assets of liquid salts as a coolant for heat transport at high temperature. As a result, the MSR Provisional System Steering Committee has modified the original R&D orientations and objectives in the original Gen. IV Technology Roadmap (NERAC/GIF, 2002).

7. GIF = Generation IV International Forum.

The current System Research Plan emphasises the role of liquid salt chemistry in the demonstration of viability, with such essential R&D issues as: *i)* fuel salt, coolant, fission product and tritium behaviour; *ii)* compatibility with structural materials for fuel and coolant circuits, as well as fuel processing materials development; *iii)* on-line fuel processing; *iv)* maintenance, instrumentation and controls development; *v)* safety issues, including interaction of liquid salts with sodium, water, air. In addition, the development of adequate simulation tools coupling neutronics, thermal-hydraulics and chemistry together with basic models is a high priority task. Experimental (analytical and integral) infrastructures (*e.g.* liquid salt loops) are needed at mid-term.

The Assessment of LIquid Salts for Innovative Applications (ALISIA) project (FZD, 2007a) forms a major part of the EURATOM contribution to GIF activities on the MSR and liquid salt applications.

[NB *A survey of the EURATOM involvement in research and training in innovative fuels and materials relating to Gen. IV was given in a paper to the 2007 High-performance Light Water Reactor Information Exchange Meeting at Cadarache, France (Van Goethem, 2007)*.]

3.2.3 Support facilities

Support facilities for providing the data required for resolving these various issues continue to be essential. Data collected from past experiments carried out on now dismantled or still existing facilities are not sufficient to cover the need of the evolutionary and next generation power systems. Specific new experiments are required, many of which can indeed be covered by existing facilities, provided they are maintained and refurbished. However, new support facilities would need to be constructed if a strong justification in terms of cost/benefit for assessing new safety and operational issues is provided, or for replacing outdated ones.

The experimental facilities, research reactors and tests in power reactors need to cover the measurement of the following parameters in critical and subcritical configurations:

- neutron multiplication and k-effective;
- buckling and extrapolation length;
- spectral characteristics;
- reactivity effects;
- reactivity coefficients;
- kinetics measurements;
- reaction rate distributions;
- power distributions;
- nuclide composition.

[NB *The use of research reactors for safety-related studies is discussed in Sections 3.5 (Fuel) and 3.7 (Safety). In addition there will be a need for test facilities for criticality-safety studies of advanced fuel cycles; see Section 3.5. MA-bearing fuels will pose difficult conditions*.]

Interpretation of reactor physics experiments for the purpose of improved understanding of system behaviour, of assessing the predictive power of models used and introducing refinements for best estimates requires two components:

i) the data describing the basic underlying phenomena of the macroscopic system behaviour;

ii) computer codes to predict the results from the interplay of the large number of different basic events; *i.e.* the macroscopic or integral effects.

Therefore, it is essential that, in addition to the integral facilities, facilities providing newly required or improved basic data are maintained. This relates, in particular, to the nuclear physics facilities described in Section 3.1.

As for computational models and codes, they have to cover:

- core physics;
- coupled neutronics/thermal-hydraulics;
- radiation shielding;
- criticality safety;
- physics of the fuel cycle;
- materials activation;
- decay heating;
- energy deposition.

The necessary basis for providing integral experimental data for model development and validation must be available and maintained and, indeed, expanded in order to meet the new requirements from advanced reactor designs.

The NSC together with the NEA Data Bank, in collaboration with the member countries and other specialised institutions have developed internationally-shared databases with evaluated and qualified experimental data in addition to a large set of computer codes covering the different needs in nuclear applications modelling. The databases cover:

- basic nuclear data (NEA, 2008g, 2008s), and chemical thermodynamics data (NEA, 2008uu);
- criticality experiments [ICSBEP (INL, 2008)];
- radiation shielding and dosimetry experiments [SINBAD (NEA, 2008pp)];
- reactor core and lattice experiments [IRPhE (NEA, 2008r)];
- data from coupled neutronics/thermal-hydraulics experiments and reactor operation (NEA, 2008k);
- fuel behaviour experiments [International Fuel Performance Experiments (IFPE) (NEA, 2008oo)].

Basic data needs for innovative systems in particular for transmutation of waste products (the minor actinides such as ^{238}Pu, ^{242}Pu, ^{241}Am and ^{242m}Am and fission products) are extensive and have to be quantified and prioritised. Other data are also required, such as: *i)* improved capture cross-sections of certain absorbers (hafnium, erbium and gadolinium), *ii)* improved scattering cross-sections of oxygen; *iii)* better knowledge of yields of fission product isotopes from the fission of most heavy isotopes; *iv)* decay schemes and energy yields of radioactive isotopes. In general, cross-section measurements, with higher than current resolution and covering the energy range from thermal energies to several MeV, are required for a number of important isotopes. For the purpose of providing guidance for those planning measurements, a high priority nuclear data request list for industrial applications has been established and is maintained by NEA (2008w).

The data evaluated and maintained within these databases are in the public domain. The aim is to contribute to and share reactor physics model and method improvement within the international community. This is achieved via specific projects or through international benchmark exercises. Other important data are proprietary, have commercial value or are accessible only through specific arrangements. These databases contain not only valuable data but document the development of measurement techniques and interpretation methodologies. Analysis of the content and quality of these databases provides a means to identify the coverage within the current knowledge base and, possibly, to identify further needs and thus to justify new experiments to fill existing gaps for advanced reactors. The public domain data are not comprehensive enough for all aspects of the current and future needs.

Data from some closed down facilities have been preserved in the public domain, see the IRPhE and SINBAD projects (NEA, 2008r, 2008pp). In particular, practically all reactor shielding facilities have now been dismantled, but the knowledge acquired has recently been transferred to a large extent to the databases and to the methods in the computer codes. However, validation of new codes for reactor dosimetry and shielding must now rely on the available evaluated experiments in the databases.

Section 4.4 expands on the work on such databases and the NSC recommends that the methods and QA procedures used for those be adopted for documenting current and future experiments with the proviso that the QA procedures should be consistent with needs and not needlessly cumbersome.

Evaluation of the accuracy of methods and codes is the objective of verification, validation and qualification studies. Measurements made in critical facilities, and irradiation measurements in reactors, play an essential role in the qualification studies. The interpretation of experiments is a driving force for the continuous improvement of computational methods and nuclear data.

3.2.4 Reactors, critical and subcritical assemblies

The RTFDB database has been prepared in the knowledge of the existence of the IAEA Research Reactors Database (IAEA, 2008e). It has not been the intention of the present project to cover exhaustively all of the reactors listed on the IAEA database; rather the purpose is to refer to facilities that provide guidance on the trends in the availability of research reactors. A further source of information on research reactors is the International Group on Research Reactors (IGORR, 2008).

In the light of the discussion above, this section analyses the needs for future research and test facilities within the reactor development field. Then, on that basis, Section 3.2.5 provides some specific recommendations.

3.2.4.1 Analysis of needs

Research reactors

"Research reactors" is a generic terminology which groups a number of different types of facilities; these can notably be dedicated to the development of new generations of nuclear plants, but also to the production of radionuclides for medical purposes, to material science experiments, to basic research, to safety benchmarks, to training, etc. In addition to all the different and numerous purposes which they can be assigned, nuclear research reactors constitute unique and necessary infrastructures aimed at supporting the industrial nuclear electricity generation capacity and its further development [see, for example (IAEA, 2004; WNA, 2008f)].

With 20% of the operating fleet world wide, the Russian Federation is the country currently possessing "the greatest park of research reactors" (Gabaraev, 2006).

It should be noted from the outset of this analysis that, in parallel with national programmes, international collaboration has, for many years, been viewed as vital and has already led to the realisation of a number of major projects. One significant example of this form of collaboration is the Institut Laue Langevin-High Flux Reactor (ILL-HFR), which was originally co-funded by Germany and France in 1967, with the UK joining as a third associate member country in 1973 [ILL, 2008]. Today ILL gathers nine additional European countries which have signed "Scientific Membership" agreements: Spain, Switzerland, Austria, Italy, the Czech Republic, Sweden, Hungary, Belgium and Poland.

However, many research reactors were put into operation in the 1960s and are thus clearly ageing. Some of them have already shut down, and a substantial number is awaiting the same fate. As an example, 245 reactors were reported as operating in 2007 (IAEA, 2007b), two-thirds of which were older than 30 years, while 272 research reactors were operating in 2004 (IAEA, 2003a). Within the European member states of the OECD, R2 – a 50 MW(th) reactor in Sweden – was shut down in 2005 (Studsvik AB, 2005). In France OSIRIS, a French 70 MW(th) reactor which has been in operation since 1966, is expected to be shut down by 2010 and Phénix is due to shut down in 2009. Within the same 1950s-1960s time frame, the following facilities also commenced operation; they continue to operate today within current and individual licensing periods which are generally of the period of up to about 10 years.

- BR2 (Belgium), 100 MW(th), in operation since 1961;
- BRR (Hungary), 10 MW(th), in operation since 1959;
- Halden (Norway), 19 MW(th), in operation since 1960;
- HFR (Netherlands), 45 MW(th), in operation since 1963;
- LVR15 (Czech Republic), 10 MW(th), in operation since 1957.

In China, HWRR and SPR are "facing the ageing problems" and will be "out of service successively in the near future" (Yuan, 2007).

However, alongside the observation that some research reactors have been shut down or are about to close, there are developments which are new and encouraging.

- Reactors recently coming into operation:
  - OPAL in Australia (first criticality 12 August 2006) (ANSTO, 2008).
- In construction:
  - The Jules Horowitz Reactor (JHR) (CEA, 2008c; EC, 2008a); France launched the JHR project in 1998. It involves French companies like EDF, AREVA, but also European partners, and it is supported by the European Commission. The first step of this shared implementation was performed in the frame of a co-funded EURATOM Framework Programme FP5 Project "Future European Union Needs in MAterial Research Reactors" (FEUNMARR) (CORDIS, 2002) which led to the elaboration of a joint conclusion: "*There is clearly a need as long as nuclear power provides a significant part of the mix of energy production sources…Given the age of current MTRs (Material Test Reactors), there is a strategic need to renew MTRs in Europe; at least one new MTR shall be in operation in about a decade from now*" (EC, 2008a). Further details of the JHR project are given in Section 3.2.4.2.
  - In China: CARR (start-up expected during 2008) (ENP, 2003; Shen, 2007; Yuan, 2007) and Chinese Experimental Fast Reactor (CEFR) (scheduled for criticality in 2009) (WNA, 2008a).
  - The PIK reactor in Russia (start-up expected 2009-2011) (Gabaraev, 2006).
  - One sad development is the halting of the commissioning of the MAPLE reactors in Canada; they were undergoing commissioning tests and relicensing, but this was aborted in May 2008 (WNN, 2008).
- In the planning stage: (more details are given in Section 3.2.4.3):
  - Japan Materials Testing Reactor (JMTR) upgrade in Japan (JAEA, 2008f);
  - PALLAS, Netherlands (NRG, n.d.; PALLAS, 2008);
  - Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA), Belgium (SCK•CEN, 2007);
  - an innovative nuclear prototype reactor, probably a Sodium-cooled Fast Reactor (SFR), France;
  - advanced recycling reactors (previously referred to as advanced burner reactors or advanced burner test reactors) in the GNEP programme, USA.
- In the United States, the Advanced Test Reactor (ATR) at the Idaho National Laboratory was designated by the Department of Energy as a National Scientific User Facility in April 2007 (INL, 2007), which will provide more nuclear energy researchers access to this facility and supporting post-irradiation examination capabilities. It is also expected that this designation will result in capability enhancements at the ATR over the next several years. At the same date, the capabilities of the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL, 2008) were significantly expanded through the successful installation of a cold neutron source.

In this context it is pertinent to emphasise some qualities of fast neutron research reactors as well as their current status and future expectations. They possess complementary features to thermal systems, notably related to their high neutron fluxes[8] and to their neutron energy spectrum. Primarily, however, they constitute necessary knowledge generating pathways before further scaling-up and industrial development of fast neutron reactors, which are believed to be 60 to 80 times more efficient in energy production from uranium feedstock than thermal neutron reactors and which could also transmute highly radioactive long-lived minor actinides (*i.e.* Am, Np and Cm) and possibly highly radioactive long-lived fission products such as technetium and iodine. Thus their importance for

8. Fluxes of $4.4 \times 1\,015$ n.cm^{-2}.s^{-1} can be achieved with Phénix.

qualifying new materials and fuels for new fast reactor systems (*e.g.* innovative SFRs) needs to be emphasised alongside the need for associated fuel testing facilities and hot labs. Further information on this last topic is given in the discussion on fuel in Section 3.5.

At present time, four fast neutron research reactors (all SFRs) are operated world wide: Phénix (France) (CEA, 2008e), JOYO (Japan) (JAEA, 2008e), BOR-60 (Russia) (RIAR, 2008) and Fast Breeder Test Reactor (FBTR) (India) (IGCAR, 2008a). In China, the 20 MW(e) CEFR first divergence is scheduled for 2009.

One industrial plant, BN-600 [600 MW(e)], is currently operated in Russia. BN-800 [800M W(e)], whose re-budgeting and construction restarted in 2006, should be commissioned in 2012 (Ivanov, 2006). In addition, India is constructing the PFBR [1 200 MW(th), 500 MW(e)], which should be put into operation by 2010 (IGCAR, n.d.; WNA, 2008b).

In addition to France, three other OECD countries envisage building SFRs, but they would likely not be operational much before 2020.

- The GNEP programme in the United States is considering a SFR concept for use as an advanced recycling reactor to consume minor actinides, with operation possible in the 2020-2025 time frame (GNEP, 2008).
- Korea is presently developing the KALIMER-600 (Korea Advanced LIquid MEtal Reactor) (KAERI, 2008, WNA, 2008c); this is a 600 MW(e) SFR loaded with metallic fuel U-TRU-Zr.[9] Under GNEP a partnership on SFR was agreed with Korea in 2006.
- In Japan, Mitsubishi Heavy Industry has been selected by the government to develop and construct a SFR by 2025, followed by a commercial reactor by 2050 (WNN, 2007).

In the period before these new reactors are built and started up, Phénix will have been definitively shut down in 2009. As a consequence, after this date, JOYO [140 MW(th)] (JAEA, 2008e) and MONJU [280 MW(e)] – which was planned to be re-started in 2008 (JAEA, 2008) – will constitute the only available fast neutron reactors in the OECD area until the newer reactors become available. This confirms the need for the new or updated facilities and emphasises the importance of these plans being brought to fruition.

Other facilities

In the context of the expected near-term nuclear renaissance, and in the light of preparing the nuclear technology of the future, in particular in the frame of Gen. IV, new R&D facilities covering the whole fuel cycle will be needed.

Inter alia, innovative research facilities dedicated to the exploration of advanced reprocessing processes precluding the sole separation of plutonium (for non-proliferation purposes), and pilot shielded and automated facilities plants for the fabrication of new types of: *i)* fuel (*e.g.* carbides, nitrides or metallic form and possibly loaded with minor actinides) and *ii)* cladding are essential steps prior to any further scaling-up.

In this context, there are ambitions for the ATelier Alpha et Laboratoires pour ANalyses, Transuraniens et Etudes de retraitement (ATALANTE) facility, France (CEA, 2008), to achieve “GANEX” (Group ActiNide EXtraction process) over the period 2008-2012 in order to avoid the sole separation of plutonium. The next step would be the construction of an international laboratory at La Hague, France, to be operational by 2015-2020, which would be followed by an industrial scaling-up facility expected by 2040.

Today, most of the fuel loaded in nuclear power plants consists of uranium oxide and, to a limited extent, of MOX. When group extraction has become a reality, fuels will very likely be made of uranium, plutonium and minor actinides, in order to address proliferation concerns and to minimise the amount of high-level long-lived waste. Because of the inclusion of the latter, these new fuels will be very radioactive and will require dedicated shielded and automated fabrication plants.

The above-mentioned laboratory will be designed to fabricate minor actinides-loaded fuel that will be shipped by 2025 to the Japanese fast neutron reactor MONJU for irradiation tests in the framework of the Global Actinide Cycle International Demonstration (GACID), a common project between CEA, JAEA and the DOE (Carré, 2007).

9. TRU = transuranic.

Learning from the past

While these comments refer to recent and future developments, it should be kept in mind that good results can still be obtained with older reactors. However, it also has to be recognised that current requirements on measurement accuracy and in the type of measurements place additional burdens on contemporary work that may only be achieved with newer equipment. Equally, it is recognised that while one can draw from older measurements [such as recorded in the IRPhE project (NEA, 2008r) and where there is an attempt to see if the results of older experiments can be brought to the point of being acceptable according to modern standards], the content in older work is finite. In essence, more information is required nowadays. Of course, there is currently a greater ability to perform simulations which can sometimes avoid the need for further experimentation. However, as is demonstrated by the cases for new facilities, requirements remain that can only be met by new, tailored facilities.

3.2.4.2 Current status of reactors, critical and subcritical assemblies

In reviewing the reactor experimental field, it is plain that some reactors are of general application while others are, for instance, oriented towards specific uses such as neutron scattering. In the latter category, it can be noted that all the facilities, including reactors, which were considered in the United States Office of Science and Technology Policy Interagency Working Group on Neutron Science (US IWGNC, 2002) are included in the RTFDB database.

In relation to fast reactors, the IAEA has compiled a database which provides information on existing and planned fast reactor plants (IAEA, 2008a). It is also available as TECDOC-1531 (IAEA, 2007) which supersedes the earlier TECDOC-866 from 1996. The database contains detailed information on liquid metal fast reactors – specifically their plant parameters and design details. A large number of parameters; design data and relevant graphic material are accessible.

Further information on the current status of a number of research reactors and critical assemblies has been supplied. The following should not be viewed as a comprehensive listing of facilities that are available around the world; rather it is intended to provide a view restricted to those facilities where the future is changing or which are of particular significance for reactor development.

Belgium

VENUS (Baeten, 2007). VENUS is a zero-power critical facility built in the 1960s at SCK•CEN. It has been in use ever since and has been used extensively for code validation for a wide variety of topics including, amongst others: reactor pressure surveillance, Pu recycling in LWRs and burn-up determinations. It has been decided to transform this facility into an ADS system where the core will consist of enriched uranium (30%) in a lead matrix and driven by a continuous-wave deuterium accelerator. The internal neutron source will be provided by a deuterium or tritium target. This programme, called GUINEVERE, will run until 2013.

France

- MASURCA (Fougeras, 2005) is a “zero” power reactor (5 kW) dedicated to the studies of fast neutron reactors and the development of measurement techniques. In 2006, the loading of a “gas” core to validate the new neutronic control system of the facility marked the beginning of an important renovation plan which, in 2013, will provide an upgraded facility able to meet new challenges. In the meantime, while MASURCA is undergoing a major overhaul, there will be no experimental programme for several years.
- The EOLE (Fougeras, 2007) critical facility is a very low power experimental reactor devoted to the neutronic study of moderated lattices, in particular pressurised water reactors (PWR) and boiling water reactors (BWR). Recent elements of the experimental programme are:
  - FUBILA, from 2005 to 2006, being the continuation of the 100% MOX study in the high burn-up BWR which was started with the earlier BASALA programme;
  - FLUOLE, which aimed by the end of 2006 to give an experimental database for the 1 300 MW(e) PWR vessel fluence calculation;
  - PERLE, in 2007, for the study of a heavy reflector option for the European Pressurised Reactor (EPR).

- The experimental reactor MINERVE (Fougeras, 2007) is devoted to neutronics studies of lattices of different reactor types. A unique and highly valuable feature of this facility is the availability of a sample oscillation device. The measured reactivity variations are induced by the periodic oscillation of small material or fuel samples. MINERVE is also used for training purposes.
  - The OCEAN programme (Oscillation en Coeur d'Echantillons d'Absorbants Neutroniques) will take place between 2005 and 2008 in various neutron spectra. In the framework of the lengthening of the fuel cycle and of the increasing utilisation of MOX fuel, it answers the need for more accurate knowledge for neutron absorbers: ^{155}Gd, ^{157}Gd, ^{nat}Gd, ^{177}Hf, ^{178}Hf, ^{179}Hf, ^{180}Hf, ^{166}Er, ^{167}Er, ^{168}Er, ^{170}Er, ^{160}Dy, ^{161}Dy, ^{162}Dy, ^{163}Dy, ^{164}Dy, ^{151}Eu, ^{153}Eu and ^{nat}Eu.
  - The OSMOSE programme (OScillations dans Minerve d'isOtopes dans des Spectres Eupraxiques) started in 2005 and will end in 2010. It will permit validation, over a large range of neutron spectra, of the absorption cross-sections of the minor actinides: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm.
- As noted above, France launched the Jules Horowitz Reactor (JHR) project in 1998 (CEA, 2008c; EC, 2008a). The JHR consortium agreement was signed on 19 March 2007 by 8 partners: CEN•SCK, CEA, EDF, AREVA, NRI, CIEMAT (including a pool of Spanish industries and public bodies), VTT, and the EU. JHR will provide high thermal neutron flux (up to 5.5 10^{14} n cm^{-2} s^{-1}) for fuel studies and high fast neutron flux (up to 10^{15} n cm^{-2} s^{-1} > 0.1 MeV) to simulate, especially, material ageing. The predicted commissioning date is 2014.

For a recent review of developments in the French research reactors, see Rive (2007).

Italy

- TRIGA RC-1 (ENEA, 2007) is a Triga Mark II reactor upgraded to 1 MW in 1967. Recent experimental programmes included the conclusion of the ADS experimental campaign in 2005 and the introduction in 2006 of a full 1 MW core for neutron radiography, medical isotope production, irradiation on demand and training.

Japan

- RTFDB lists several criticality assemblies and experimental reactors in Japan. Of these, the Deuterium Criticality Assembly (DCA) and the Very High Temperature Reactor Critical Assembly (VHTRC) of JAEA are already shut down. However, other criticality assemblies are widely used, not only for research and development of new fuel systems, but also for nuclear safety research. The following are of particular significance to reactor physics development.
- The Fast Critical Assembly (FCA) (JAEA, 2008b) has been used for several mock-up experiments of Japanese fast reactors such as JOYO (JAEA, 2008e) and MONJU (JAEA, 2008). In addition it has been used for other technical developments for fast reactors such as for an advanced FBR core and a moderator-added FBR core, as well as tests conducted with a gas expansion module.

  FCA has been used for verification of nuclear data and reactor physics methodologies for fast reactor design; in particular, cross-section data of minor actinides (MA) and effective delayed neutron yield (β_{eff}) have been measured (Sakurai, 2002).

  Because of the flexibility of its core configuration and fuel materials, FCA has been used for research and development of other types of reactors including thermal systems. For example, reactor physics experiments for the High Conversion Light Water Reactor (HCLWR), Reduced-Moderation Light Water Reactor (RMWR), and the 4S (Super, Safe, Small and Simple) Reactor, plus a basic experimental study of an ADS system were all carried out in FCA.
- The Tank Type Critical Assembly (TCA) (JAEA, 2008k) was initially built in order to research the reactor physics of light water reactors including the Japan Power Demonstration Reactor (JPDR).[10] TCA has since been used for a variety of reactor physics experiments. In the fast phase of its utilisation, it was used for development and verification of reactor physics experiments and neutronics parameters of reactor cores of JPDR and the Nuclear Ship MUTSU (JAEA, 1998).

10. Fully dismantled in 1996.

After that, reactor physics experiments relating to plutonium utilisation in thermal reactors were carried out. Subsequently, TCA has been used for criticality safety research including physics of subcritical system and measurement of subcriticality, and reactivity measurement of MA isotopes have been conducted in recent years.

Because TCA is a simple and well-characterised criticality assembly, it has been used to educate students and staff from other institutes. More than one hundred people participate in the TCA education programme each year. The future programme of TCA is under discussion.

- STACY (JAEA, 2008j) and TRACY (JAEA, 2008l). Nuclear criticality safety is one of the most important issues in safety evaluation of fuel cycle facilities. In Japan, a nuclear fuel cycle utilising spent fuel reprocessing is the main strategy of the national energy programme. The first commercial reprocessing plant, Rokkasho Reprocessing Plant (RRP) of Japan Nuclear Fuel Limited (JNFL, 2008), has been constructed in the Aomori prefecture. In order to validate calculational code systems and data libraries by obtaining experimental data of solution systems of uranium fuel in static as well as transient status, the respective facilities the Static Experiment Critical Facility (STACY) and the Transient Experimental Critical Facility (TRACY) have been operated.

  Since the initial criticality of STACY, fundamental criticality data of 10% and 6% enriched uranyl nitrate solutions as well as criticality properties for complicated systems such as multiple core systems have been obtained. The recent experimental activities are with a heterogeneous core (an array of fuel rods) with 6% enriched uranyl nitrate solution with fission product isotopes such as Cs, Nd, Sm, Gd and Eu in order to simulate the dissolver of a reprocessing plant. These data are used for burn-up credit analysis. Some of the data taken in STACY was evaluated in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) (INL, 2008).

  In TRACY, transient characteristics have been studied using 10% enriched uranyl nitrate solutions. Power profile data were obtained in various reactivity addition tests, and basic data such as the number of fissions in a criticality accident of low-enriched uranyl nitrate solution system were obtained. The knowledge accumulated in the TRACY experiments and the code system validated through analysis of these experiments were employed in the analysis of the criticality accident in Japan which occurred in 1999. Recent experiments in TRACY have been devoted to the study of the mechanism of void generation by radiation in solutions in criticality accidents.

- KUCA (KURRI, 2002). Experience obtained by operating actual nuclear reactors and criticality assemblies which are flexible tools for validating innovative concepts and methodologies of reactor physics has important meaning not only for research and development but also for education. The operation of criticality assemblies in universities is therefore a unique and important role.

  The Kyoto University Critical Assembly (KUCA) is a multi-core type critical assembly. It is a facility for the study of reactor physics which is available for the joint use of researchers from any university in Japan. KUCA has been used for the study of the nuclear characteristics of the Kyoto University High Flux Reactor, physics of coupled cores, critical experiments of thorium fuel, critical experiments using medium-enriched uranium fuel, criticality safety issues and reactor physics studies for high conversion light water reactors.

  – One of current aspects in the reactor physics experiments of KUCA is a study of erbium isotopes as a burnable poison of light water reactor fuel.

  – Another is the Kumatori Accelerator-driven Reactor Test project (KART) (KURRI, 2004). In this project, KUCA is connected with the Fixed Field Alternating Gradient (FFAG) accelerator, and several studies on the medical application of accelerators, materials, chemistry and physics as well as ADS will be conducted.

- JOYO (JAEA, 2008e) is the Japanese Fast Breeder Reactor operated by JAEA. It has been used for obtaining experimental data for reactor physics and technical development of fuel design and fabrication, plant behaviour, removal of decay heat and determination of defect fuels.

  The initial core of JOYO was upgraded to the MK-II version in 1982 and to MK-III in 2003 in order to archive higher irradiation performance and capacity. Recently, JOYO has been used for obtaining information on cross-section data of neutron-induced reactions of minor actinides (MA). Because JOYO is an experimental reactor and an irradiation experimental facility is co-located, experiments treating higher radioactive materials are possible.

- MONJU (JAEA, 2008) first achieved criticality in April 1994. A significant amount of modification work was completed in May 2007, based on the lessons learned from the sodium leakage accident in 1995. The reactor was planned to be restarted in 2008, initially by conducting a series of core confirmation tests and subsequent power raising tests over a two-year period (JAEA, 2008a).
- JMTR (JAEA, 2008f). See Section 3.2.4.3.

Russian Federation

- NIIAR has two renovation projects (Gabaraev, 2006): the currently under way MIR.M1 (Grachyov, 2005; Smirnov, 2000) and BOR-60 for which the design and engineering documentation is being prepared (RIAR, 2008).

  The BN-800 (Ivanov, 2006) fast neutron reactor being built by OKBM at Beloyarsk is designed to supersede the BN-600 unit and utilise MOX fuel with both reactor-grade and weapons-grade plutonium (WNA, 2008d). Following some earlier setbacks, re-budgeting for construction is approved and the programme was re-launched in 2006 with commissioning expected by 2012.

Switzerland

- PROTEUS (PSI, 2008a) is a highly versatile critical facility at the Paul Scherrer Institute (PSI), which since the 1960s has been used to provide integral data of relevance to a wide variety of advanced reactor types. These have ranged from the gas-cooled fast reactor (GCFR), through the high conversion light water reactor (HPLWR), to the pebble-bed HTR.

  The high degree of flexibility with regard to the neutron spectrum, which can be investigated in the central PROTEUS test zone, is largely due to the multi-zone character of the facility. Thus, in most of the experimental programmes carried out, the central test zone has been driven critical by outer thermal driver regions (with graphite and D_2O moderators), a dry natural uranium metal buffer zone separating these from the central test region.

  The most recently completed experimental programme, LWR-PROTEUS, represents a close collaboration between PSI and swiss*nuclear*, the Association of Nuclear Utilities in Switzerland. During the various phases of the programme, detailed investigations have been conducted on actual full-length fuel assemblies from the nuclear power plants.

  It is planned to refurbish and expand the PROTEUS research reactor in order to realise the new programme, LIFE@PROTEUS (Large-scale Irradiated Fuel Experiments at PROTEUS). Apart from modernisation of its instrumentation and control system, the facility's capabilities will be extended to handle larger radioactive inventories in the form of full-length spent fuel rods. For this purpose, the central region of the multi-zone reactor will have a large water tank in order to enable underwater manipulation of the irradiated fuel.

United States

- The United States still has a number of research reactors operating at universities across the country providing valuable research and education opportunities. The major operating test and research reactors operated by the US Department of Energy are the Advanced Test Reactor (ATR, 250 MW) at the Idaho National Laboratory (INL, 2007) and the High Flux Isotope Reactor (HFIR, 85 MW) at the Oak Ridge National Laboratory (ORNL, 2008). With the shutdown of the Experimental Breeder Reactor-II and the Fast Flux Test Facility in the 1990s, fast flux irradiation capabilities within the United States are somewhat limited.

Other reactors indicating reactor development activities, which are listed in RTFDB, are in Brazil, Canada, China, India [where the 1250 MW(th) Prototype Fast Breeder Reactor (PFBR) is due for operation around 2010 (IGCAR, n.d.; WNA, 2008b)], Netherlands, and Slovenia.

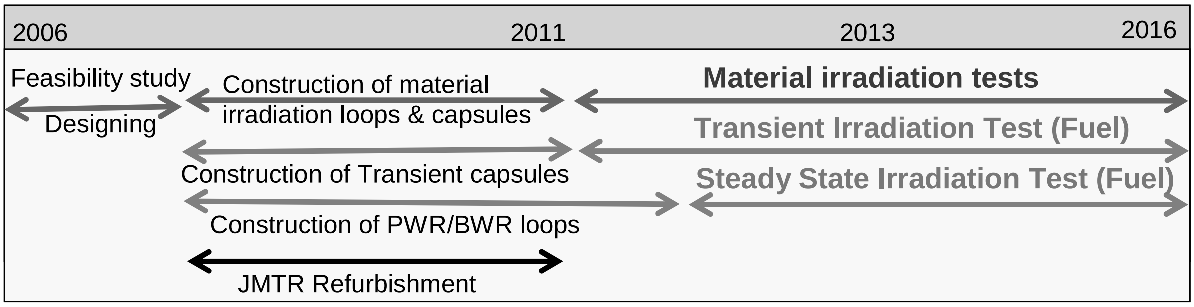
3.2.4.3 Reactor, critical and subcritical assembly developments in the planning stage

The following facilities are in the planning phase; such concepts may naturally change over time.

JMTR Upgrade, Japan

The Japan Materials Testing Reactor (JMTR) had been running since 1968 until August 2006 (JAEA, 2008f). The reactor had been used for power ramp tests of BWR fuels and for reactor material irradiation tests on blanket materials for fusion, irradiation-assisted stress corrosion cracking (IASCC) and irradiation embrittlement. JMTR is currently shutdown for refurbishment of the reactor and upgrading of the irradiation facilities. Re-start of the reactor is expected in 2011. The JMTR refurbishment plan has been discussed in several JAEA and governmental committees in recent years to have the plan authorised. Preparation for the JMTR refurbishment started in April 2007 to permit the completion of the renewal of the reactor control system, cooling system and electric power supplies within a four-year period. The anticipated preliminary schedule for the refurbishment, irradiation facility upgrade and subsequent tests is shown in Figure 4.

Figure 4: Preliminary schedule for refurbishment and upgrading of JMTR



A feasibility study on the material and fuel irradiation studies in JMTR and preliminary work on the design of the facilities were conducted in the Japanese FY 2006, sponsored by the Nuclear and Industrial Safety Agency (NISA)/Ministry of Economy, Trade and Industry (METI). The study is being extended to prepare for materials tests on stress corrosion cracking (SCC), corrosion, fracture toughness and irradiation growth of nuclear materials under radiation fields. The schematic configuration of the materials irradiation loops for in-pile SCC and corrosion tests is illustrated in Figure 5. The influence of radiation on the crack growth rates and corrosion will be investigated under simulated LWR water conditions including the chemistry.

A feasibility analysis of the fuel irradiation study under transient and steady-state conditions in JMTR has been made. The basic design of the transient test facility and the water loops for simulated high-duty uses of newly developed fuels has been done. Fabrication and installation of the transient test facility is expected to start in 2008. The transient tests will start with power ramp tests of BWR fuels in a boiling capsule under natural convection cooling in 2011. The facility is similar to the boiling capsule (BOCA), which had been successfully used for the power ramp tests of BWR UO_2 fuels in JMTR. A schematic configuration of the BOCA capsule is illustrated in Figure 6. The tests could be extended to cover forced convection conditions and boiling transient conditions using newly designed capsules. The new transient tests would provide failure criteria of UO_2 and MOX fuels with modified design and materials at high burn-ups under abnormal transient conditions. In addition to the transient test facility, upgraded water loops for fuel bundle irradiations were designed and proposed to conduct steady-state irradiation of newly developed fuels for high-duty uses under a well controlled environment simulating the LWRs. The loop would be able to simulate the new water chemistry, *e.g.* noble metal and hydrogen addition, and also provide higher burn-up UO_2 and MOX fuels for further testing. The loop irradiation tests would provide on-line measurement and post-test data on fuel behaviour, such as cladding corrosion at high burn-ups and cladding lift-off due to high rod internal pressures. Facilities for radioisotope production, such as ^{99m}Tc, and neutron doping of silicon semiconductors are also planned to be installed in JMTR.

Figure 5: Schematic of the material irradiation loops in JMTR

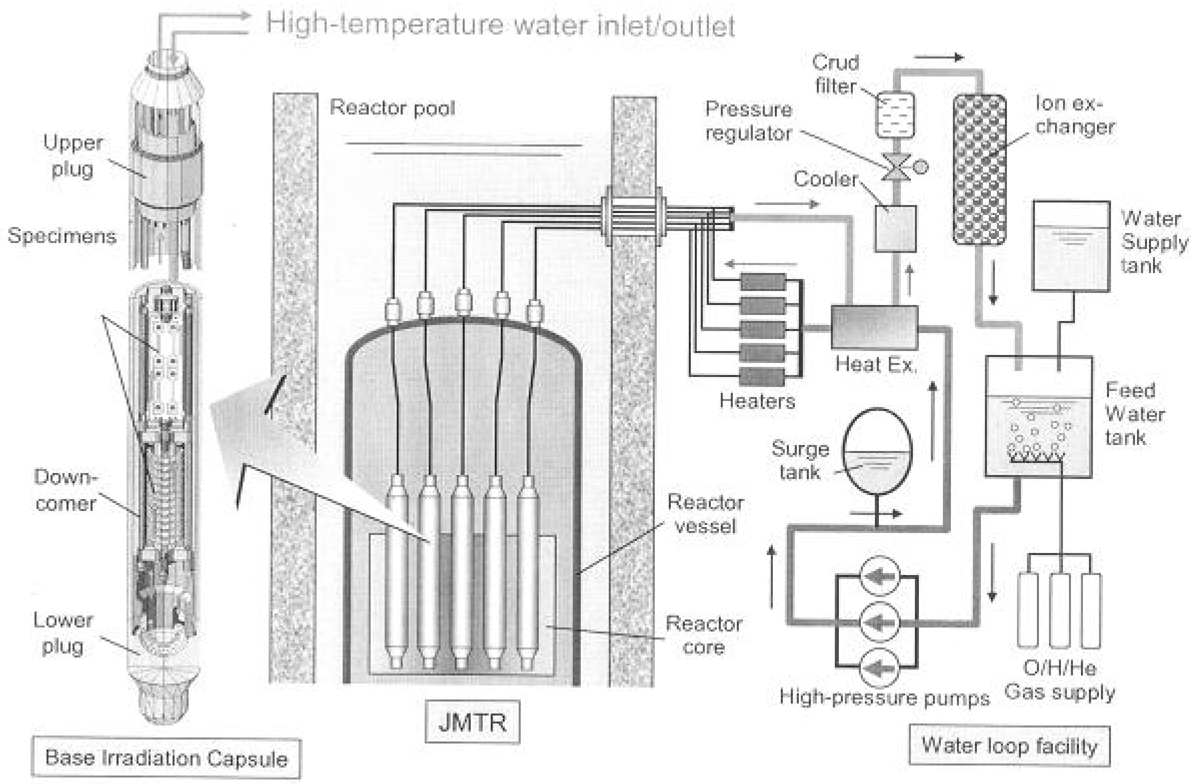
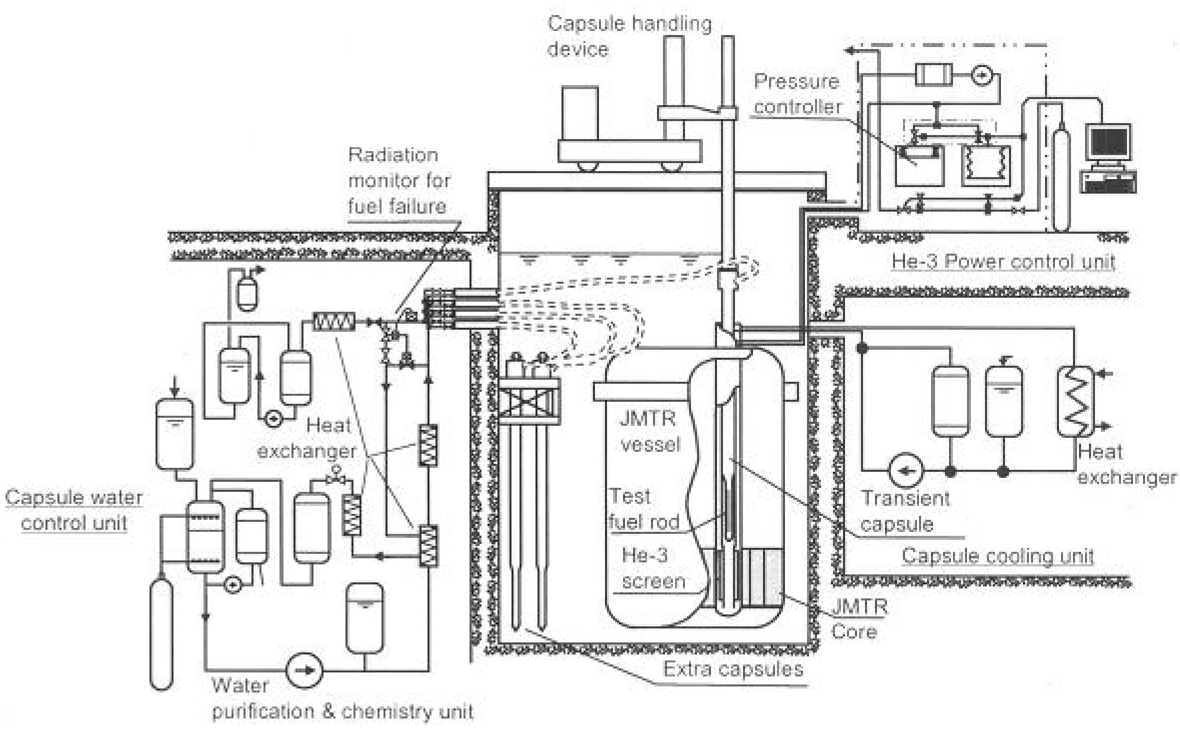


Figure 6: Schematic of the fuel transient test capsule



PALLAS, Petten, Netherlands

NRG, in co-operation with Mallinckrodt Medical B.V. and the Joint Research Centre of the European Commission, has started a project that will lead to the construction and operation of a new reactor by 2015 (NRG, n.d.; PALLAS, 2008) of a “tank-in-pool” type. The project is focusing on the technical and financial options, on the preparation of the licensing procedures and on obtaining the necessary social and political support.

MYRRHA, Belgium

Because of its relevance to accelerator-driven systems, the discussion of MYRRHA (SCK•CEN, 2007) will be found in Section 3.4. In brief, MYRRHA is a small Pb-Bi-cooled XADS: 40 MW(th) core power, driven by a 600 MeV $\times$ 3 mA proton beam current to be delivered by a linear accelerator on a liquid Pb-Bi windowless spallation target. (A 350 MeV cyclotron option had been discussed previously but is now outdated since no cyclotron can provide the required beam stability.)

An innovative nuclear prototype reactor, France

On January 2006 former French President Chirac announced the construction of this prototype Gen. IV reactor, which is likely to be a sodium-cooled fast reactor, and which would commence operation in 2020.

Advanced recycling reactors

In the frame of GNEP, the USA has proposed the development and implementation of advanced recycling reactors[11] to consume transuranic elements separated from spent fuel while producing power. Under GNEP, an advanced recycling reactor could be operating between 2020 and 2025 (US DOE, 2006a).

3.2.4.4 Future reactors, GNEP, Gen. IV, etc.

The review earlier in this chapter largely concentrated on facilities which are in existence, soon to be implemented or for which planning is under way. However, there are a number of initiatives that have the potential for development of new and advanced reactor systems. It is clear that programmes such as the medium-term GNEP (2008) and long-term Gen. IV (GIF, 2008a) programmes are already leading to work on a number of future reactors and other facilities. GNEP grew recently to a consortium of 21 countries and is focusing on:

- the need to deal with waste materials in a responsible manner;
- the costs involved with developing the necessary infrastructure;
- the need to develop and deploy technologies that will increase the efficiency of the fuel cycle;;
- the risks posed by the potential for proliferation of nuclear materials and sensitive technologies.

The six Gen. IV systems selected by the Generation IV International Forum (GIF) for further study were mentioned earlier in this section. Further up-to-date information on the project is best obtained from the GIF website (GIF, 2008a), which is maintained by the Nuclear Energy Agency.

3.2.5 Conclusions and recommendations – reactor development

This section brings together some recommendations based on the earlier analysis of the current and expected near future situation together with some views of the longer term. The section is divided into five parts:

- identifying the needs;
- building new facilities;
- extending reactor life;

11. “Advanced recycling reactors” were previously referred to as “advanced burner reactors” or advanced burner test reactors”.

- enhancing international co-operation;
- keeping the know-how;

These items will now be discussed in further detail.

Identifying the needs

Generally speaking, obtaining a perspective on the future needs relating to specific research fields and types of reactors is extremely difficult; variations in importance of particular reactor designs over time and in particular countries influence the view. In addition, the current resurgence of interest in novel reactor concepts is widening the field of development. For example, gas-cooled reactors (GCRs) were regarded as being of interest several decades ago; however, all but a few countries abandoned GCR development programmes thereafter. More recently, with the current development of the PBMR and also within the Generation IV International Forum, interest in GCR technology has returned again to prominence. Equally, Pu utilisation is a key interest in Japan, so reactors which can use Pu and MOX are important there. However, other countries are not so active in this area.

As well as the direct requirements for specific research reactors and criticality assemblies related to particular reactor designs, there is also a need for zero (or low) power reactors and even subcriticality assemblies for basic reactor physics experiments and educational purposes. This is despite the extensive development of reactor core simulators; it is evident that the actual experience of use of a reactor core is the most effective way to gain an understanding of the behaviour of a nuclear reactor. This requirement to extend the knowledge of the skills base applies for any nuclear energy developments in the future regardless of reactor types adopted. On this basis, it is clear that criticality assemblies should be versatile (multi-purpose) in order to be able to react to changing requirements.

The requirement for research reactors as a source of neutrons must also be stressed. Current basic research requires high intensity neutron sources as probes of materials. While large and/or multi-purpose accelerators feature in current trends for such facilities [*e.g.* IFMIF (ENEA, 2008) and Joint Accelerators for Nanosciences and Nuclear Simulation/Jumelage d'Accélérateurs pour les Nano-sciences, le NUcléaire et la Simulation (JANNUS) (Serruys, 2007)], conventional research reactors have advantages such as the ability to provide continuous irradiation. Thus both types of facility (accelerators and reactors) are required and are complementary in their abilities.

Building new facilities

Several OECD countries have started construction or have announced their intention to build new nuclear reactors and other nuclear fuel cycle facilities. On the other hand, non-OECD countries like Russia, China and India have already launched active programmes leading to concrete implementation in this field. In particular, significant efforts have been deployed in order to build fast neutron reactors in those countries, albeit not of the Gen. IV type.

In this context, OECD member countries could possibly provide a new impetus to the promotion of the relevant R&D in order to encourage innovation in the nuclear industry and to retain leadership in nuclear technologies. Clearly needing to be within the scope of the available national budgets, this momentum could assist the further promotion of international collaboration. Possible areas are: the construction of a (perhaps even jointly-owned) fast neutron reactor and/or of a laboratory dedicated to grouped extraction and to the fabrication of minor actinide-loaded fuel (see below under "*Enhancing international co-operation*").

Extending reactor life

Operating research reactors should be kept – with the proviso that they meet best international safety standards – in order to ensure that current and future research activities can be carried out. In particular, the need for criticality facilities could be underlined, as they are used for reactor physics and criticality safety studies. Nevertheless, some existing facilities have already been shutdown. An example of extending the life of an existing reactor is the upgrade of JMTR by JAEA (2008f); see the earlier discussion in Section 3.2.4.3.

Enhancing international co-operation

The recent expansion of the GNEP partnership to 21 member countries (GNEP, 2008) is an indication of the desire for collaboration within the international nuclear power community in order to address near term developments. Similarly, former French President Chirac (while announcing on 5 January 2006 that France intended to construct an innovative nuclear prototype reactor to commence operation in 2020) underlined that industrial and international partners who would like to become involved would obviously be welcomed. With a longer-term perspective, the Gen. IV initiative (GIF, 2008a) already brings together a number of countries through the Generation IV International Forum with the aim of making progress in reactor designs for future application.

Following the same line of thought, further federation of the financial, scientific and technical efforts of the OECD countries could optimise available resources. This would have the aim, for instance, of better usage of existing research reactors, or of building broadly accessible (or even jointly owned) nuclear facilities, following a similar approach to that adopted by the Institut Laue-Langevin consortium of countries (ILL, 2008) [or, in the fusion field, ITER (2008)].

International institutions have a key role to play in the promotion of such co-operation between countries and existing synergies between NEA and IAEA activities[12] in this matter might be explored further.

Also, the exchange of researchers, of research plans, and of results should be encouraged. Ongoing examples, such as the collaboration between facilities in France and Belgium [*e.g.* EOLE (Fougeras, 2007) and VENUS (Baeten, 2008)], could be followed.

Keeping the know-how

Present nuclear research-related facilities are operated by very competent and experienced scientists and technical staff. In the current context of a “nuclear renaissance”, it is recommended that this human resource and expertise be preserved, which thus requires the appropriate recruitment and retention of younger staff to replace those reaching retirement.

Examples from the past have shown that technologies that were abandoned may experience a revival; one such case is HTR, which is now one of the six systems selected within the frame of Gen. IV. This demonstrates the need to maintain the operational capability of existing facilities, as far as is appropriate, and to conserve the body of knowledge built up.

Also in relation to knowledge retention, it was noted at the beginning of the current section (3.2) that the work (which is further expanded upon in Section 4.4) on databases of older experiments such as IRPhE (NEA, 2008r) has led to the NSC recommendation that the methods and QA procedures used for those be adopted for documenting current and future experiments. The current Expert Group activity confirms this recommendation with the proviso that the QA procedures should be consistent with the needs and not needlessly cumbersome; clearly if the demands of the QA process are an undue burden, there is a risk of discouraging contributions.

There is also a wealth of information that exists in the results from past experiments on irradiated MOX, carbide, nitride and metal fuels, as well as operational reactor feedback, sodium technology, etc., and these resources should be conserved. Obviously, not all the information is in the open environment, but the information exists and some is available; see, for instance, the review paper by Crawford, *et al.* (Crawford, 2007).

Within the context of building up the base of knowledge within younger researchers, the initiative of France and Germany in creating and building up the Frédéric Joliot/Otto Hahn Summer School on Nuclear Reactors Physics, Fuels and Systems (CEA/FZK, 2008) has, over the last dozen years, been instrumental in both bringing together significant numbers of young researchers and exposing them to up-to-the-minute issues in nuclear power development. With a rather different and, perhaps, more political and leadership orientation the World Nuclear University has also been operating since 2003 [WNU, 2008]. These initiatives are to be encouraged.

12. For instance, Sub-programme D.2 “Research Reactors” within the IAEA’s Programme D. Nuclear Science and with the new IAEA Technical Working Group on Research Reactors (TWGRR).

3.3 Neutron applications

There are two applications of neutrons that are undertaken on reactors and spallation neutron sources which, because of their widespread and important applications, bear separate consideration. The following sections therefore consider neutron scattering and neutron radiography.

3.3.1 Neutron scattering[13]

Neutron sources

Neutrons are used to study the structure of matter on length scales from the atomic to several centimetres, and are used by the nuclear industry and academics for applications ranging from the validation of nuclear codes through to study of materials important to the nuclear industry, such as: nuclear fuels, fuel cladding, pressure vessel steels and welds; moderator materials, as well as potential storage materials such as glasses and minerals.

There are approximately 270 neutron research reactors distributed over 56 countries, and Table 1 provides information on a selection. These operate with different types of fuel: many of the most powerful use highly-enriched uranium (HEU), in which the proportion of ^{235}U is greater than 20%. However, in recent decades there has been increasing pressure to reduce the number of reactors that use this fuel because of heightened concerns about the risk of nuclear materials diversion; in parallel, there have been research programmes to develop forms of lower enrichment fuels that provide comparable performance in many of the reactors originally designed to operate with HEU.

Most of the neutron research facilities are based on fission reactors, many of which were built 40 to 50 years ago; however in recent years the power and reliability of accelerators has increased to such an extent that many governments have now chosen to build spallation neutron sources for neutron scattering research. A complete list of the world's spallation neutron sources is given in Table 2.

Table 1: Selection of reactor-based neutron sources world wide that run user facilities (ENP, 2003)

Reactor	Location	Operation since	Thermal power (MW)	Max. unperturbed neutron flux (in-pile) ($n.cm^{-2}.s^{-1}$)	Number of neutron-scattering instruments	Number of radiography/ tomography beam lines
			Europe			
HFR	ILL, Grenoble, F	1971	57	1.5×10^{15}	37	1
FRM II	TUM, Garching, D	2004	20	8×10^{14}	20	2
HFR	JRC, Petten, NL	1961	45	5×10^{14}	4	2
Orphée	LLB, Gif-sur-Yvette, F	1980	14	3×10^{14}	25	1
BRR	BNC, Budapest, HU	1959	10	2.5×10^{14}	6	2
BER II	BENSC,* Berlin, D	1992	10	1.2×10^{14}	20	3
IBR-2	FLNP, JINR, Dubna, RUS	1984	1 500 (in pulse) 2 (continuous)	1×10^{16} (in pulse)	11	0
			USA/Canada			
ATR[†]	INL, USA	1969	250 MW	1×10^{15}	None	0
HFIR	ORNL, USA	1966	85	2×10^{15}	10	0
NCNR	NIST, Gaithersburg, USA	1969	20	4×10^{14}	19	1
NRU	AECL, Chalk River, Canada	1957	125	3×10^{14}	7	0
			Japan			
JRR-3M	JAEA, Tokai-mura, Japan	1962	20	2×10^{14}	24	3
			Australia			
OPAL	ANSTO, Australia	2006	20	4×10^{14}	9	0

* Previously HMI – renamed "The Helmholtz Centre Berlin for Materials and Energy" in June 2008.

† Although not normally used for neutron scattering experiments, ATR is now designated a national Scientific User Facility in the United States.

13. The assistance of S.M. Bennington (Rutherford Appleton Laboratory, UK), A. Harrison and U. Köster (Institut Laue-Langevin, France) in preparing this section is gratefully acknowledged.

Table 2: List of the accelerator-based neutron sources world wide that run user facilities
(ENP, 2003; CCLRC, 2005)

Facility	Locations	Source type	Other information
KENS	KEK, Japan		Closed, 2007.
SNS	Oak Ridge, USA	1.4 MW pulsed	Started user operation in 2007.
ISIS	Rutherford Appleton Laboratory, UK	0.16 MW pulsed (0.24 MW by 2009)	Currently the world's leading pulsed neutron source. The second target station will start operation in 2008.
IPNS	Argonne, USA		Closed, 2008.
LANSCE	LANL, USA	0.8 MW continuous 0.1 MW pulsed	
J-PARC	Tokai, Japan	0.6 MW pulsed	Due to start operation 2008.
SINQ	PSI, Switzerland	0.75 MW continuous	Continuous spallation neutron source.
CSNS	Dongguang, China	0.25 MW pulsed	Due to start operation in 2010.
ESS	Europe	1-5 MW	Not yet funded.

Techniques[14]

- *Nuclear science*: Beam lines exist at LANSCE (LANL, 2008a) for nuclear science and the measurement of neutron cross-sections from the meV to MeV range, and fundamental nuclear science machines are under construction at the Spallation Neutron Source at Oak Ridge (SNS, 2008b) and are planned at J-PARC in Japan.
- *Imaging*: Neutron radiography provides a non-destructive probe of the internal structure of materials in which there is a contrast between different regions on account of different mean scattering cross-sections; further details are given in Section 3.3.2. Many reactor sources provide such facilities, while among spallation sources excellent facilities exist at SINQ (PSI, 2008b), there are some at LANSCE, and there is a proposal for a machine at ISIS (2008).
- *Diffraction*: Diffraction is used to study the atomic structure, to obtain a phase analysis, look at internal strain, material texture, lattice defects, etc. All established facilities have several diffractometers covering different wavelengths and resolutions.
- *Small angle neutron scattering*: This is used for studying fission gas and helium bubble formation and the formation of nano-domains or cluster formation. It can also be used to study the decoration of hydrogen on dislocations, or the porosity of rocks and glasses for storage applications. All facilities have a provision for small angle scattering.
- *Strain scanning*: By looking at small changes in diffraction patterns it is possible to map out the strain or texture deep inside components with sub-millimetre resolution. This can be used to validate welding codes to study map the strain fields around crack tips. There are several dedicated facilities able to do this sited at ISIS, SNS, LANSCE and in the future Japan Spallation Neutron Source (JSNS). Reactor sources commonly provide dedicated facilities for such measurements – indeed the upgraded HFIR at Oak Ridge National Laboratory includes a residual stress mapping instrument in the initial suite of instruments, while ILL gave a high priority to this class of instrument in its own upgrade programme.
- *Activation analysis*: It is possible to identify different elements and isotopes by measuring the energies of prompt gamma photons emitted on neutron absorption. Even the newest and most powerful spallation neutron sources have time-averaged fluxes that are lower than high power research reactors and so applications that rely on a high time-integrated flux such as activation analysis (and studies of irradiation damage) are confined to fission-based sources. Most, if not all, reactors have the ability to perform activation analysis, but facilities also exist at SINQ (PSI, 2008b) and LANCSE (LANL, 2008a).
- *Neutron irradiation*: Such work is very common at reactors, with good examples provided by research programmes at Petten in the Netherlands, NRU at Chalk River in Canada, HFIR at Oak

14. For information, some details of other capabilities alongside those strictly for neutron scattering are indicated.

Ridge National Laboratory in the USA, and the Munich Research Reactor in Germany. Although SINQ has some provision, the time-averaged flux at most pulsed sources is too low to be competitive with reactors. A dedicated facility at Los Alamos, called the Materials Test Station (MTS) (LANL, 2008; Cappiello, 2006), designed to mimic the conditions that fast reactor fuels and materials experience, is due to complete the conceptual design in 2008. It will achieve fluxes in excess of those experienced in-reactor by allowing samples very close to the neutron target. It should be noted that facilities that enable irradiation to be conducted at non-ambient temperatures are rare, despite a continuing need to perform such work.

- *Inelastic neutron scattering*: This is used to study the excitations and motion of atoms and molecules in condensed phases, illuminating atomic bonding, the measurement of density of states, and the diffusion rates of gasses and water in materials. It is often used to create low-energy kernels for neutron transport codes.
- *Elemental transmutation* (OECD, 1999; 2000): New facilities such as the J-PARC facility, due to come on stream in 2008, and LANL's MTS (LANL, 2008; Cappiello, 2006) are expected to be available relatively soon. Reactor sources commonly include facilities to produce isotopes for medical research, or to implant impurities into host materials – for example to produce doped semiconductors. For further discussion on transmutation related to nuclear waste, see Section 3.4.

Facilities

In the near future it is likely that neutrons will continue to play their part in materials development and testing and the validation of neutron Monte Carlo transport and engineering codes. The recent resurgence of interest in nuclear power is likely to increase the demand for this kind of work and, although this will be tensioned against the other uses of neutrons, there is no serious shortage of facilities in the short term. However, the majority of the research reactors in the OECD area were built in the 1950s and 60s and some have already closed, including: Brookhaven (USA), Risø (Denmark) and Jülich (Germany). Decisions pertaining to the closure or refurbishment of many others will have to be made as they come to the end of their working lives before 2015 (Richter, 1998). However, by this time the FRM-II reactor at Munich (TUM, 2008) will be running with a full instrument suite and the OPAL reactor in Australia will be operational. The world's leading research reactor at the ILL was refurbished in the early 1990s and there are no technical reasons why it should not run until at least 2030. Further, the ILL has just launched the next phase of its renewal programme (ILL, 2008a) which has already seen an increase in detected flux by a factor of 15 since 2000 as a consequence of new or refurbished instruments and guides.

Two of the pioneering spallation neutron sources, IPNS at Argonne in the USA and the KENS facility at KEK in Japan have also closed recently, replaced by more powerful national sources: the SNS at Oak Ridge in USA and the JSNS at the J-PARC proton accelerator facility in Japan. There are doubts that the SNS will reach its design power of 1.4 MW quickly because of problems with its mercury target, but it is likely to reach at least 0.5 MW by 2010 and probably full design power by 2015; by which time, if all goes according to plan, construction of its second target station should have begun. [NB *Information on this is contained in an ILL report (2008a)*]. The JSNS planned ramp-up in power is much more cautious, but by 2010 it is likely to have exceeded the current power of ISIS (0.16 MW) but with half the number of instruments and should then reach full power by 2015.

While the SNS and JSNS are building up to full power, plans for new sources in Europe are currently less clear. There has been discussion about a European Spallation Source (ESS) (ENP, 2003a) for several years; a new process to decide between different bids competing to host it was launched in 2007, with the aim of coming to a decision on a timescale of the order of a year. At present, proposed plans are for a 5 MW long pulse source. In parallel with such planning, existing national sources are also being strengthened in Europe: in addition to developments noted above at FRM-II and the ILL, the ISIS facility is currently upgrading its accelerator to 0.24 MW and, as of September 2008, is starting the test on its second target station.

Currently new sources appear to be coming on stream to keep pace with the projected losses due to source closures. However, there will be a concentration of resources in a smaller number of larger sources and a shift in the centre of gravity from Europe to North America and Japan unless the ESS is built soon and some existing sources in Europe continue to be supported strongly (Richter, 1998).

The challenges facing the nuclear industry have evolved very little; the measurement of neutron cross-section and yields to validate neutron transport codes, investigation of the many materials problems that are associated with the high temperatures and radiation environments found in reactors and the safe long-term storage of waste are among the issues that have not yet been "solved". There are however, some new challenges: *i)* a new set of materials issues around fast ion ablation and radiation damage of plasma facing components, as well as problems associated with tritium breeding has been raised by the funding of ITER and the recent development of fast ignition in the inertial fusion community; *ii)* the proposed use of reactors to produce hydrogen through high temperature steam reformation will mean materials will also have to be found that can survive in a highly corrosive nuclear environment; and *iii)* transmutation poses many difficult materials and nuclear issues that are only just beginning to be addressed.

The following note records some of the current and recent developments:[15]

- *LANSCE, Los Alamos*: The Los Alamos Neutron Science Center at the Los Alamos National Laboratory (LANL) is based around a 1 MW, 800 MeV linear accelerator that is used to drive several facilities (Lisowski, 2006). Some of the facilities use the protons directly from the linear accelerator and some use short pulses from a compression ring. The centre is currently undergoing a series of enhancements including: the construction of an ultra-cold neutron facility and a refurbishment programme that will start in 2008 to improve the reliability of the accelerator. Particular facilities at LANSCE include:
  - *Isotope Production Facility*: This takes protons from the accelerator soon after the initial drift tube at 100 MeV for the production of medical isotopes.
  - *Materials Test Station (MTS)*: This will be a high-power 0.8 MW facility that uses the 800 MeV protons directly from the linear accelerator to optimise and test the next generation of materials and fuels needed to develop advanced fission systems (LANL, 2008; Cappiello, 2006). The conceptual design of the MTS project should be completed this year.
  - *Weapons Neutron Research Facility (WNR)*: This has two spallation targets producing neutrons from 100 keV to 800 MeV and is fed from the proton storage ring. It is used for basic and applied nuclear science, including: cross-sections for neutron-induced reactions in actinides, fission cross-sections and yields, fundamental nuclear physics and neutron resonance radiography. It is also used for the accelerated testing of semiconductor devices for industry.
  - *Lujan Neutron Scattering Center*: This pulsed spallation source uses the 0.1 MW proton beam from the storage ring to produce neutrons in the meV to keV range. There are 12 beam lines covering a wide range of neutron techniques including beam lines for transmission radiography, neutron nuclear science, materials and engineering and condensed matter physics and chemistry.
  - *Ultracold Neutron Facility*: This is now operational and will be used for fundamental neutron physics.
- *SNS, Oak Ridge*: The Spallation Neutron Source (ORNL, 2008b) is a third-generation 1.4 MW spallation neutron source based around a 1 GeV proton accelerator and accumulation ring. At full power it will be the most powerful spallation neutron source in the world. The facility started operating a user programme in 2007 and currently has 18 instruments out of a potential total of 24. The instruments span a full range of neutron techniques including: diffraction and inelastic scattering machines, small angle scattering, an engineering beam line for residual stress and microstructure measurements, and a beam line dedicated to fundamental neutron physics.
- *ISIS*: ISIS (2008) is a spallation neutron source currently running at 0.16 MW, which will rise to 0.24 MW during 2008. Currently it has 18 instruments on its first target stations and a further 7 due to come on-line on its second target station in 2008. The facility has been running a successful user programme since 1987 and its instruments cover the whole range of scattering techniques, including: diffraction, inelastic, small angle neutron scattering and an engineering beam line for residual stress and microstructural measurements.

15. For information, some details of other capabilities alongside those strictly for neutron scattering are indicated.

- *J-PARC*: The spallation neutron source at the J-PARC proton accelerator facility (J-PARC, 2008a) is due to start commissioning its accelerators in 2008. It consists of a 600 MeV linac that feeds 3 GeV and 50 GeV synchrotrons. The 50 GeV synchrotron will be used for nuclear and particle physics and the 3 GeV one will feed a neutron spallation source and a transmutation facility. Particular facilities at J-PARC include:
  - *Materials and Life Sciences Facility*: This should receive its first neutrons in 2008 and will be starting its user programme later in the year. Five instruments are currently under construction and another five are being designed. The suite will contain a nuclear physics instrument for the measurement of nuclear capture cross-sections as well as the usual suite of small angle scattering machines, diffractometers and spectrometers. Ultimately it is hoped that the accelerator will reach 1 MW but at present there is only funding for a slow ramp to 0.6 MW.
  - *Accelerator-driven Transmutation Facility*: This will be composed of two experimental facilities: a physics facility TEF-P and a test target TEF-T. TEF-P is low power to study the dynamics of a critical target and TEF-T will have a 200 kW lead-bismuth target and will be used to test engineering components for transmutation.

3.3.2 Neutron radiography

Because neutrons are neutral and deeply penetrating, neutron radiography is a non-destructive evaluation technique, which possesses certain unique features distinguishing it from photon radiography. The interaction of neutrons with matter is governed by nuclear, rather than electrical characteristics of the medium, and is thus complementary to X-ray radiography where the reaction is with the diffused cloud of electrons. Neutron radiography uses its nuclear-related aspect to give evidence of light atoms (*e.g.* hydrogen) in the presence of heavier ones.

Neutron radiography has many industrial applications, for example, in the motor industry for the inspection of running engines. The main application remains, however, in nuclear materials inspection for examination of irradiated and non-irradiated nuclear fuel and reactor components. Key examples include the detection of cracks, or the revelation of hydrogenous materials such as water or synthetic polymers and plastics inside a host that is highly absorbing of optical light or even X-rays; for more detailed information, see Section 3.3.2.1 below.

New detector developments including real-time dynamic imaging techniques using equipment like charge-coupled device (CCD) cameras make neutrons an even more versatile instrument for investigation of materials. Some topical developments can be found in recent conference proceedings (ISNR, 2008b; IAEA, 2006a).

Those facilities known to have neutron radiography as one of their activities have been flagged in RTFDB, while a convenient graphical display of "Neutron Radiography Sites of the World" is to be available soon on the International Society for Neutron Radiology website (ISNR, 2008). To summarise, however, it can be said that many reactor sources provide such facilities, while among spallation sources excellent facilities exist at SINQ, there are some at LANSCE, and there is a proposal for a machine at ISIS.

A review of European facilities for neutron radiography was presented by E.H. Lehmann (2000), at the 15th World Conference on Non-destructive Testing, in Rome in October 2000.

More recently, at the 8th World Conference on Neutron Radiography (WCNR-8) held in Gaithersburg in October 2006 (NIST, 2006), a request was made to update an approximately ten-year old database of neutron radiography facilities held by ANL. [NB *The conference was immediately followed by a Workshop: "IAN 2006" (Imaging and Neutrons 2006) at SNS, Oak Ridge (ORNL, 2006a).*] The 9th World Conference is due to be held in 2010 (ISNR, 2008a).

Neutron radiography limits in detecting materials with low adsorbing properties can be overcome with the phase-contrast radiography technique: the image formation is obtained by the phase variations transformed in intensity variations in a neutron beam presenting a high lateral coherence due to the presence of the investigating sample (Kardjilov, 2004). The particular beam geometry reduces the neutron intensity on the sample, increasing the acquisition time, limiting the application of that technique to intense neutron sources. An extensive adoption of that technique could integrate the neutron scattering and neutron radiography competences in a synergic action offering the widest applicability of neutron imaging techniques.

3.3.2.1 Non-destructive analysis of nuclear fuel by means of neutron radiography

A very high degree of quality assurance is required for nuclear fuel in order:

- to ensure that any failures in manufacture are discovered before the fuel is put into use;
- to ensure the lowest possible failure rate while in use in a reactor, despite the stringent conditions of temperature, pressure, radiation level and power density;
- to provide the longest possible period of storage after irradiation without leakage of the radioactive material from within.

When added to current interests in obtaining the maximum efficiency in the operation of a particular power plant through such actions as prolongation of the cycle between outages (for reactors that do not have on-load refuelling) and operation with as flat and as high a power distribution as possible but at the same time not exceeding peak power conditions on any particular components of the fuelled core, it can be seen that ensuring the quality of the fuel being produced is key to success.

Because of their poor penetration of uranium and other fissionable fuel materials and the limited ability to penetrate cladding materials such as zircaloy and stainless steel, X-rays have limited use for non-destructive examination. They can be used if high-energy photons (> 1 MeV) are available but only with fuel prior to use in a reactor since, once irradiated, the fission product and other activation product gamma rays make discrimination of interrogating photons difficult. This has led to neutron techniques having particular interest in the quality assurance and PIE of nuclear fuels. Additional benefits of the use of neutrons can be found in:

- being able to detect hydrogen with a high level of sensitivity since hydrogen can, for example, be responsible for cladding embrittlement (damage can occur when high concentrations are accumulated in the outer surfaces of clad materials);
- being able to detect burnable poison materials, *i.e.* boron, lithium, gadolinium and dysprosium.

Recent application of digital imaging methods, combined with tomographic analysis, now permits 3-D reconstruction of a particular object. E.H. Lehmann, *et al.* (2003) show a good example of a sphere-type fuel element from the high-temperature reactor (HTR) programme. The 60-mm diameter graphite sphere contained ~8 500 TRISO fuel particles 0.5-mm in diameter, which were separated with image processing tools from measurements at the NEUTRA radiography facility at the SINQ spallation neutron source at PSI. The same paper also shows some examples of the detection of hydrogen in cladding. Further discussion of recent developments can be seen, for example, in (Lehmann, 2007).

It is worth noting that most spallation sources are not licensed nuclear sites and do not have facilities to handle highly active materials; recent legislation makes it increasingly difficult to handle even moderately active samples. The only exceptions to this are LANSCE in the USA and SINQ in Switzerland, where there is some provision.

3.3.3 Conclusions and recommendations – neutron applications

Currently new neutron scattering sources appear to be coming on stream to keep pace with the projected losses due to source closures. However, there will be a concentration of resources in fewer, larger sources and, unless the ESS is built soon (Richter, 1998), there will inevitably be a shift in the centre of gravity from Europe to North America and Japan.

It is anticipated that neutron diffraction and small angle neutron scattering measurements of the structure and the defects therein will continue to play a role in testing and developing new engineering materials for nuclear technology. It is also to be expected that the burgeoning fields of strain scanning and texture analysis will grow in importance as they find a wider audience. Inelastic neutron scattering measurements will probably constitute a lower level activity in this field, though they should retain an important role in measuring scattering kernels which will be used in Monte Carlo studies to study the behaviour of moderator performance.

An extensive use of neutron radiography techniques in fuel fabrication processes requires the adoption of standardised methods for non-destructive controls. Neutron radiography facilities should operate in a co-ordinated way in conjunction with standard authorities (ASTM, ISO, etc.) in order to adopt procedures for qualification of beams, technicians and image treatments.

Application of modern neutron radiography methods such as phase-contrast radiography could integrate neutron scattering and neutron radiography competences in a synergic action offering a wider applicability of neutron imaging techniques.

3.4 ADS and transmutation systems

Transmutation of long-lived nuclides contained in spent nuclear fuels is one of the key candidate technologies for sustainable utilisation of nuclear energy. Nuclides to be transmuted are minor actinides (MA) such as ^{237}Np, ^{241}Am and ^{243}Am, and long-lived fission products (LLFP) such as ^{99}Tc and ^{129}I. Plutonium may also be included depending on the nuclear fuel cycle policy of individual countries. The basic concept for transmutation of such nuclides is to irradiate them with neutrons in nuclear reactors and to induce their fission and capture reactions. The nuclear reactors can be in a subcritical or critical state. The neutron energy spectrum can be that of a fast reactor or a thermal reactor, though this alteration causes large differences in the cross-sections of the neutron-induced reactions.

The strategy for transmutation is generally categorised into two concepts: a heterogeneous cycle or a homogeneous one. A heterogeneous cycle uses dedicated fuel which contains a large fraction of MA without uranium. The homogeneous system basically uses commercial power reactors whose fuel contains a few per cent MA.

An accelerator-driven system (ADS) is considered a powerful tool for effective transmutation of MA because it can be operated safely with high MA contents even in a homogeneous mode.

MA fuels for both strategies commonly have a difficulty (more or less) with their heat generation and radiation emission (gamma-rays and neutrons). From this point of view, the heterogeneous strategy has an advantage, since MA can be transmuted in a concentrated manner without the long-distance transportation of MA-added fuel between a commercial reprocessing plant and numerous commercial power reactors. Nevertheless, dedicated MA fuels for the heterogeneous concept are faced with various technical challenges. The selection of the strategy, therefore, should be made carefully, based on the particular national circumstances, its prospect of operating or accessing a nuclear fuel cycle (*e.g.* if a country decides not to recycle then transmutation strategies are also impossible) and on the progress of various aspects of R&D.

When the experimental R&D for ADS and other transmutation systems are considered, three areas should be included: *i)* basic databases for MA and LLFP such as nuclear data, critical experiments and material properties; *ii)* fuel and fuel cycle technology; *iii)* specific activities for ADS. Although many of the broader aspects of these activities are described in other sections of this report, it is sensible to focus on them here in terms of the transmutation technology. Thus, the technical issues for each item are reviewed and the required facilities to achieve the technology are discussed in the following sub-sections.

3.4.1 Basic database for MA and LLFP

A basic database of MA and LLFP data is indispensable for both of the transmutation strategies. Nuclear data for MA and LLFP nuclides are very important because they may influence the safety of the transmutation system as well as the transmutation performance of the system. Although we have accumulated lots of differential and integral experimental data so far for principal nuclides such as ^{235}U, ^{238}U and ^{239}Pu and have modified the nuclear data to reproduce the results of integral experiments, data for MA and LLFP are of extremely poor quality with which to design a transmutation system with high precision.

As explained in Section 3.1, differential nuclear data measurement experiments were very active world wide in the 1960s to 80s, but many facilities such as electron linacs to make TOF measurements have been shut down. Recently, accelerator facilities such as high-energy proton accelerators or heavy ion accelerators, which were originally built for the study of fundamental physics, are playing an important role to measure the nuclear data of MA such as the n_TOF facility at CERN (2008).

Integral validation of nuclear data is also important. Some activities of sample irradiation in reactors and critical experiment using reactivity worth samples and activation foils have been implemented. However, very few critical experiments have been conducted fuelled with kilogramme order amounts of MA. There is one exception in the BFS Np experiment in Russia. It is, therefore, significantly important to make such critical experiments using sizeable amounts of MA for transmutation systems research. It is from this viewpoint that the Transmutation Physics Experimental Facility (TEF-P) (J-PARC, 2008) is being proposed under the Japan Proton Accelerator Research Complex project (J-PARC, 2008a).

A materials properties database for MA and LLFP is also important in order to design the fuels for transmutation systems. It is, however, difficult to measure the physical and chemical properties of these materials because the material itself is rare and the amount of the materials permitted to be dealt with in a facility is generally restricted by the licence. It is, therefore, recommended that hot cell laboratories [like the Minor Actinide Laboratory (MA-Lab) at ITU Karlsruhe (ITU, 2008a)] be retained and that a sound way be developed to procure MA and LLFP samples for such materials property measurements, and also for nuclear data measurements and reactor physics experiments.

3.4.2 Fuel and fuel cycle technology

To achieve a meaningful amount of transmutation of long-lived nuclides, large amounts of MA should be loaded in a transmutation system as a part of its fuel. For example, about 1 tonne of MA would need to be transmuted per year for a 40 GW(e) fleet of LWRs, which implies that 10 tonnes of MA should be loaded in the transmutation systems because the transmutation rate is expected to be about 10% MA/year. If we consider the total inventory of MA in the transmutation cycle (*i.e.* including the cooling, reprocessing and fuel fabrication stages) amounts of MA several times larger yet would have to be reasonably managed for both homogeneous and heterogeneous strategies. It is, therefore, of great importance to establish the technology for the MA-bearing fuel and its fuel cycle.

For the homogeneous strategy, it will be possible to add MA to fast reactor fuel up to about 5% of the heavy metal; addition of MA to fast reactor fuel (oxide or metal) is considered not so influential to the material properties if it is restricted to a few per cent. The irradiation behaviour of such fuel is, however, still to be verified. The impact of MA will also possibly appear in the fuel fabrication, transportation and handling processes because of its high heat generation, high radioactivity and high neutron emission rate.

As for the dedicated transmutation fuel for the heterogeneous transmutation method, neither the fuel properties nor their irradiation behaviour are reliable. Although the handling technique of such dedicated fuel has more challenging features than in the homogeneous strategy, the amount of the MA-bearing fuel to be dealt with is much smaller (less than one-tenth) and hence MA can be controlled in a concentrated manner.

The reprocessing of the irradiated MA-bearing fuel is another key issue for transmutation technology and is still under development.

Taking these above-mentioned aspects into consideration, test reactors to irradiate MA-bearing fuel and hot cell facilities to conduct post irradiation experiments are essential for the research and development of transmutation technology. The hot cell facilities are also to be used for the fabrication of the irradiation pin and the demonstration of the reprocessing technology for irradiated MA-bearing fuel. Moreover, the MA needs to be supplied in a practical manner through a separation plant of demonstrative scale, considerations that equally apply to ADS and MA-bearing fuelled fast reactors.

3.4.3 Specific activities for ADS

An ADS system is a coupling of a proton accelerator, a spallation target and a subcritical reactor including a power generation system. The acceleration energy of the protons will be in the range of 0.4 to 1.5 GeV and the beam power will be greater than 10 MW, which is about 10 times more intense than existing accelerators. The primary candidate for the spallation target is lead-bismuth eutectic (LBE). LBE has good properties as a spallation target: a wide range of liquid state (397-1 943 K), high density (10 500 kg m^{-3}) and hence large neutron yield. Recent designs of the subcritical reactor parts of ADS systems also adopt LBE as the primary coolant. (See Section 3.6.1.2 for more discussion on liquid metal test systems.)

Since ADS is a hybrid system, the coupling experiments between an accelerator and a subcritical reactor are of great importance to verify its feasibility as well as the engineering developments for each component. The technical issues for ADS and relevant experimental R&D activities are summarised item by item as follows.

Accelerator

For the proton accelerator portion of an ADS system, a superconducting linac is regarded as a promising candidate to fulfil the required performance in beam intensity and energy efficiency. The problem with a linac in comparison with circular accelerators such as cyclotrons would be the cost. In addition to these requirements, the reliability (or stability) in order to reduce the beam trip frequency and the power level controllability are important and particular features of an accelerator to be used for ADS in comparison with conventional accelerators used in physics research.

The Los Alamos National Laboratory has already accumulated remarkable experience with proton linac operation since the 1960s and is delivering an 800 MeV × 1 mA = 0.8 MW pulsed proton beam. Proton linacs are being constructed or are in commissioning under the following projects: SNS in USA with 1 GeV × 2 mA = 2 MW, the J-PARC project in Japan with 400 MeV × 0.33 mA = 0.13 MW and the PEFP project in the Republic of Korea with 100 MeV × 1.6 mA = 0.16 MW. All of these proton linacs are operated in pulsed mode, though a continuous beam is preferable for ADS. As for the continuous mode operation, a ring cyclotron at PSI in Switzerland has shown noteworthy performance so far with 590 MeV × 1.6 mA = 0.94 MW.

In addition to these accelerator facilities, several R&D activities for particular areas such as an intense ion source, a low energy part and a superconducting cryomodule have been conducted across the world.

Although most of the R&D activities for ADS accelerators can be covered by these various activities, it is considered necessary to build a dedicated accelerator in order to demonstrate its reliability, controllability, economy and safety for application to a nuclear energy system. Such a demonstration accelerator would be coupled with a subcritical reactor as an experimental ADS, which is described later – see *Accelerator-reactor coupling experiments* below.

Spallation target

Materials for spallation targets are solid heavy metals (Pb, Ta, W, U) and liquid ones (Hg, Pb-Bi). As previously mentioned, recent designs of ADS in the world usually adopt Pb-Bi (lead-bismuth eutectic: LBE) as the spallation target.

Two types of target design are mainly studied: a window type and a windowless one. The window type has a physical boundary called the beam window, usually made of steel alloy, between the LBE target and the evacuated beam duct. The beam window should be able to accept several tens of mA of proton beam which clearly will generate heat deposition in the beam window and induce spallation reactions in the window material. The beam density at the beam window is, therefore, restricted at about 30 μA cm^{-2}. Moreover, it should be noted that the beam window of a real ADS would be irradiated by fission neutrons from the subcritical core region as well as protons and spallation neutrons from the target. The beam window, therefore, should be replaced periodically for reasons of corrosion and irradiation damage.

At present, the status of material irradiation data is too poor to make a reliable design for a window-type target. Recently, the MEGAPIE (PSI, 2008) project demonstrated the feasibility of a high power LBE target at the SINQ facility at PSI. Its post-irradiation test will produce valuable knowledge. It is, however, recommended that a dedicated irradiation facility for the spallation target material for ADS be built so that a materials properties database can be prepared covering a wide range of design conditions such as temperature, oxygen content and flow velocities of the LBE, beam density and irradiation period.

To avoid the above-mentioned technical challenges in the beam window, a windowless design is being investigated, mainly in European countries. The basic idea is to maintain a free surface of the LBE jet by inertia. The vapour of the LBE and other spallation products elements and activation products are evacuated between the target region and the accelerator part to prevent the accelerator being contaminated. However, the stable control of such a free surface might be difficult when a high power proton beam is incident. To show the engineering feasibility of this type of target, therefore, a demonstration using a real proton beam with megawatt class is considered necessary before connecting it to a subcritical reactor, as well as mock-up experiments without beams.

Subcritical core

The significant issues for this part of an accelerator-driven system relate to the cooling of the core region and its reactor physics.

The candidate for the primary coolant of the subcritical core of an ADS which has been studied most is LBE. However, the corrosive nature of LBE above 500°C is one of the technical challenges for ADS. To overcome this problem, the control of the oxygen concentration in the LBE is believed to be one of the promising methods. Another is to develop new materials with particular elements or surface coatings. In any case, the experimental verification of the materials in flowing LBE is essential to the use of LBE as the core coolant. From this point of view, many LBE loop facilities have been built around the world and databases are being accumulated. The results from such facilities are not always consistent with each other due to the lack of reproducibility of the experimental conditions. It is therefore recommended that an international benchmark of experiments be organised to establish a world standard of the materials properties for the LBE coolant. Moreover, an integral test to verify the feasibility of oxygen control in a realistic reactor vessel would be necessary before constructing a large-scale LBE-cooled nuclear system.

The thermal-hydraulics of the LBE coolant should also be verified by experimental work. The high speed of the LBE flow (more than 2 m s^{-1}) might cause local erosion of the materials in the core. Large-scale components such as heat exchangers and pumps are also still to be developed for LBE. [See the FP6 VELLA Initiative (VELLA, 2008).]

It should be noted that when we use LBE as the primary coolant and/or spallation target, the management of activation products such as ^{210}Po and spallation products are important for the safe operation of the system.

The reactor physics and control of subcritical reactors should be expected to be dissimilar to those of conventional critical ones, and these aspects would be influential to the performance and the safety of the whole system. By analogy with the R&D for fast reactors, experimental verifications with use of zero power critical/subcritical facilities are important for robust development of ADS for transmutation. Several experimental projects such as MUSE in France (Billebaud, 2004) and YALINA in Belarus (Kiyavitskaya, 2006) were made or are under way using DT and DD neutron sources. The KUCA facility in Japan (KURRI, 2002) is about to start subcritical experiments using a proton accelerator and a thermal spectrum critical assembly. The TEF-P facility is also planned in Japan (J-PARC, 2008) to connect a spallation neutron source from a 400-600 MeV proton beam with a fast-spectrum critical assembly. These planned experimental works ought to be important bases for ADS and transmutation technology.

Accelerator-reactor coupling experiments

Before proceeding to large-scale ADS experimental facility with several tens of megawatts of thermal power, a capability in determining the reactivity level must be demonstrated. Special attention will be given to the investigation of on-line reactivity monitoring techniques and experimental techniques used at beam trips for the determination of the reactivity. Therefore, there is need for a lead fast critical facility connected to a continuous beam accelerator. The "Generator of Uninterrupted Intense NEutrons at the lead VEnus Reactor" project (GUINEVERE) (Baeten, 2007; SCK•CEN, 2006, 2008b) is to be carried out and will make use of the modified VENUS critical facility (SCK•CEN, 2008d) located at SCK•CEN Centre in Mol which will be coupled to an adapted neutron generator based on a continuous beam accelerator (GENEPI) working in current mode. The experimental programme on GUINEVERE is expected to produce valuable knowledge about the controllability of an ADS system including start-up and shutdown procedures and the management method of the beam trip transients.

In terms of very recent activities it can be noted that the Kumatori Accelerator-driven Reactor Test Facility (KART) has recently come into operation (KURRI, 2004) whereas, on the negative side, the TRADE facility at ENEA has been cancelled.

3.4.4 National and international projects

European projects for ADS and transmutation

The "ADOPT" programme (SCK•CEN, 2004) was mentioned in the previous report (NEA, 2003) where a summary of that programme may be found. However, the ADOPT programme has been completed and had been replaced by Integrated Project EUROTRANS (2008), for which overview information is available in the papers by Knebel, *et al.* (2005, 2007) at 8-IEMPT, Las Vegas, November, 2004 (NEA, 2008) and at 9-IEMPT, Nîmes, September 2006 (NEA, 2008a).

In summary, IP EUROTRANS is part of the EURATOM 6th Framework Programme in the Thematic Priority Area "Management of Radioactive Waste: Transmutation" for which the focus is the evaluation of the industrial practicability of transmutation of high-level nuclear waste in an accelerator-driven system (ADS) together with the development of the basic knowledge and technologies needed. Implementation of P&T of a large part of the high-level nuclear wastes in Europe would require the demonstration of the feasibility of several installations at an "engineering" level. The R&D activities have been described through four "building blocks":

i) demonstration of the capability to process a sizeable amount of spent fuel from commercial power plants in order to separate Pu and MA;

ii) demonstration of the capability to fabricate, at semi-industrial level, the dedicated fuel needed to load a dedicated transmuter;

iii) availability of one or more dedicated transmuters;

iv) realisation of a specific installation for processing of the dedicated fuel unloaded from the transmuter, and fabrication of new dedicated fuel.

IP EUROTRANS is dealing with the third building block: the transmuter. Consequently, the objectives of IP EUROTRANS are:

- To carry out a first advanced design (realisation in a short term, say about 10 years) of a 50 to 100 MW(th) eXperimental facility demonstrating the technical feasibility of Transmutation in an Accelerator Driven System (XT-ADS), as well as to accomplish a generic conceptual design [several 100 MW(th)] of the European Facility for Industrial Transmutation (EFIT) (realisation in the long-term). This step-wise approach is termed as the European Transmutation Demonstration (ETD) approach.
- For the above devices, provide validated experimental input (such as experimental techniques, dynamics, shielding, safety and licensing issues) from experiments on the coupling of an accelerator, an external neutron source and a subcritical blanket.
- To develop and demonstrate the necessary associated technologies, especially accelerator components, fuels, heavy liquid metal technologies, and the required nuclear data.
- To prove its overall technical feasibility, and to carry out an economic assessment of the whole system.
- To direct input to other FP6 Projects: PATEROS (Partitioning and Transmutation European Roadmap for Sustainable nuclear energy) (2008a) and SNF-TP (Sustainable Nuclear Fission Technology Platform) (CEA, 2008i; CORDIS, 2008c).

Other developments world wide

Developments elsewhere in the world and at a national level include:

- USA: Lead-cooled fast reactor (LFR) studies:
  - Ongoing R&D to test and analyse LFR materials with the objective of selecting key structural materials and cladding for lead-bismuth compatibility to accelerate the understanding of materials corrosion. Testing of LFR materials in LBE has occurred in the LANL DELTA Loop (Development of Lead-Bismuth Target Applications) since 2002 (Li, 2007).

- USA: sodium-cooled fast reactor (SFR) studies:
  - Ongoing efforts to retrieve and test structural material specimens (T91 and H91) irradiated to 100-200 dpa at 400-700°C in the Fast Flux Test Facility (FFTF) (Brunon, 2004).
  - Test irradiations of SFR fuels under the FUTURIX-FTA agreement including metal and nitride fuels (Pu, Np, and Am – fertile and fertile free – metal and nitride matrix) ongoing until 2009 in collaboration with France in the Phénix reactor. The effort was initiated to support European and American strategies of long-lived radioactive waste transmutation in ADS or critical fast neutron reactors (Jaecki, 2005). The objective is to provide data on the performance of CERCER (oxide), CERMET, metallic and nitride fuels loaded with very high concentrations of plutonium, neptunium and americium under fast spectrum irradiation. Uranium-bearing and uranium-free compositions are included in the experimental test matrix, as well as helium and sodium-bonded fuel pin designs (Jaecki, 2005).
  - Through the Generation IV International Forum, the USA signed the SFR Advanced Fuels Project Arrangement (PA) for SFR fuel research. The project covers advanced fuel forms evaluation and fabrication, minor actinide bearing fuels, and high burn-up fuels. The fuel forms being considered are oxide, metal, nitride and carbide. Besides the United States (through the DOE), other signatories to the PA include Japan (JAEA), France (CEA), European Community (JRC), and Korea (KAERI).
- USA: gas fast reactor (GFR) studies:
  - Test irradiations of SFR- and GFR-related fuel matrix material are under way in collaboration with France and the Phénix reactor under the FUTURIX MI (MI: Inert Materials) agreement. Post-irradiation examinations of the test materials will occur in 2009 (Carbonnier, 2007).
- France: GEDEPEON (Gestion de Déchets Radioactifs par des Options Nouvelles) programme (formerly “GEDEON”) (CEA, 2008b).

Reviewing particular facilities, the following are worth noting:

- Republic of Korea (Seoul National University (SNU): at SNU, a Pb-Bi cooled transmutation reactor known as PEACER [Proliferation-resistant, Environment-friendly, Accident-tolerant, Continual and Economical Reactor (SNU, n.d.)] has been developed since 1998.
- Belgium (SCK•CEN): studies in the field of LBE technology since 1997 have been related to the MYRRHA project (SCK•CEN, 2007) which is aimed at the development of a research reactor driven by an accelerator, where LBE is used as spallation target and coolant. MYRRHA is a small Pb-Bi-cooled XADS (40 MW(th) core power, driven by a 600 MeV × 3 mA proton beam current to be delivered by a linear accelerator on a liquid Pb-Bi windowless spallation target). (A 350 MeV cyclotron option had been discussed previously but is now outdated as no cyclotron can provide the required beam stability.) [NB *This programme is largely incorporated in IP-EUROTRANS*.]
- Japan: both ADS and lead-cooled fast reactor (LFR) systems using LBE are under development. At JAEA an ADS with 800 MW thermal power has been designed, where 250 kg of minor actinides and some long-lived fission products can be transmuted annually. R&D has been conducted on ADS using LBE as a spallation target and a coolant, and research using J-PARC is also planned. LFR systems using LBE as a coolant have been studied both at Tokyo Institute of Technology (TIT) and JAEA. One of the LFR systems studied at TIT is designated as PBWFR (Pb-Bi Cooled Direct Contact Boiling Water Fast Reactor) (Sofue, 2004).
- Russia: there has been, recently, renewed interest in lead and LBE coolants for civilian fast reactors, *e.g.* the lead-cooled BREST (NIKIET, 2008) reactor design and the LBE-cooled SVBR concept (Stepanov, 1998).

OECD/NEA activities

The OECD NEA International Workshops on the Utilisation and Reliability of High-power Proton Accelerators (NEA, 2005a, 2008p, 2008vv) have information on accelerator-driven systems. Discussions focused on accelerator reliability; target, window and coolant technology; subcritical system design and

ADS simulations; safety and control of ADS; and ADS experiments and test facilities. The proceedings contain the technical papers presented at the workshops as well as summaries of the working group discussions held.

Fast neutron irradiation facilities

It should be noted that within the OECD, only two countries possess fast experimental neutron reactors: France with Phénix [250/150 MW(e)] (CEA, 2008e) and Japan with JOYO [140 MW(th)] (JAEA, 2008e). Phénix will definitively stop in 2009. Hence, progress in knowledge acquisition might be hindered (a statement that equally well applies to systems other than just ADS).

The United States envisages constructing one or more advanced recycling reactors under the GNEP (2008) programme, while France intends to have one Gen. IV SFR by 2020. Nevertheless, no alternative resources will be available among the OECD member states during the period 2009-2014.

In non-OECD countries, Russia has an experimental fast reactor, BOR60, likely to be stopped in 2010, as does India – the FBTR [40 MW(th)] – and is currently constructing a 500 MW(e) prototype fast breeder reactor (PFBR) (IGCAR, n.d.; WNA, 2008b). China is constructing its own reactor, the CEFR [65 MW(th)].

Discussion of the associated topic of partitioning, which is often linked with transmutation, can be found in Section 3.5. [NB *The IAEA has created a database which covers accelerator-driven systems (ADS) and partitioning and transmutation (P&T) related R&D issues (IAEA, 2008)*.]

3.4.5 Conclusions and recommendations – ADS and transmutation systems

As ADS systems rank among the most innovative and challenging concepts, their technological development calls, naturally, for extensive R&D programmes including more experiments than for other, already well-established systems.

A number of recommendations have been embedded in Section 3.4 above. The following are of particular note:

- An international roadmap for ADS is of importance.
- Waste transmutation technologies are important for the sustainable development of nuclear energy all over the world. The technical challenges for ADS technology spread over a wide range of fields. It is, therefore, highly desirable to share the experimental efforts in a systematic way. The MEGAPIE project (PSI, 2008) was a good precursor for international collaboration in this field.
- An intermediate goal before the realisation of transmutation using ADS should be an experimental ADS system. European counties are implementing an R&D programme for the XT-ADS project, which would be an experimental ADS with several tens of megawatts of thermal power. There is some feeling that a global programme (perhaps in a similar form to the ITER project in fusion energy development) is desirable.
- Before proceeding to such a demonstration stage, establishment of the technical basis from which to deal with MA in nuclear energy systems and to couple a proton accelerator with a fast spectrum reactor are extremely important for the purpose of ensuring reliable design of the system, safety assessment and training of young scientists and engineers. From this viewpoint, the Transmutation Experimental Facility under the J-PARC (2008) project in Japan is expected to play important roles.

3.5 Fuels

The study of nuclear fuel in order to ensure its appropriate behaviour in-core has been an issue of long standing. This section reviews the facilities available for both performing tests on fuel (irradiation facilities and test reactors) and for the subsequent examination of the fuel and its cladding (hot cells). The issues of fuel cycle chemistry and partitioning are discussed next and the section ends with a number of recommendations.

3.5.1 Fuel development and testing

Since the very beginnings of the peaceful applications of nuclear energy, experiments carried out in both in-pile and out-of-pile facilities have provided important information on the behaviour of nuclear fuel under normal operation, accident conditions and during storage of spent fuel and have thus supported the reliable and safe operation of nuclear power plants. The development of modern numerical models for fuel behaviour simulation codes has been largely based on data derived from experimental programmes and test facilities.

As noted earlier in Section 3.2 (*Reactor development*), current trends in the utilisation of nuclear fuel are characterised in many nuclear power plants by up-rating of a station's output and increasing the burn-up of the fuel. Increased operating power, extended burn-up, coolant temperature and density changes create special challenges for the fuel. New phenomena [*e.g.* rim formation (Matzke, 1997)] can occur or the effect of some processes (*e.g.* cladding corrosion and hydriding) can become significant despite being negligible at earlier times when burn-up was lower. New fuel materials and new fuel designs have been and are being developed to meet these more stringent requirements in modern nuclear power plants. For the future, Gen. IV reactors will have their own demands for new types of reactor fuel.

Experimental investigations are required to support:

- the evaluation of fuel behaviour at the high and very high burn-up ranges;
- testing of new fuels;
- the development of new theoretical models to cater to these high burn-up and new types of fuels through measured benchmarking data.

While these experimental data should cover the whole range of normal operation, part of the focus clearly has to be on the fuel safety margins during accidents.

Existing facilities – first of all research reactors with well instrumented fuel samples – are mainly suitable for the execution of tests with new fuel types. The examination of fuels after irradiation and testing (*e.g.* mechanical, corrosion, and accident simulation) requires sophisticated post-irradiation examination (PIE) methods in order to identify the role of different phenomena and track the structural changes. The experimental programmes should generate key information for safety and licensing assessments on extended fuel utilisation and degradation of fuel materials under the combined effects of water (or other coolant) chemistry and nuclear environment. The examination of new fuel types should be based on the comparison with currently used fuels and should focus on pointing out the advantages of new designs proving their applicability in the required range of parameters.

The following sections deal with irradiation facilities, fuel fabrication and development, hot cells and other hot laboratories, and post-irradiation examination. The particular issue of partitioning is covered in Section 3.5.2.2. Materials issues *per se* are discussed in Section 3.6.

3.5.1.1 Irradiation facilities

In the context of fuel development and testing, the purpose of irradiation facilities (in practice various kinds of research reactors) is to satisfy the needs arising from: *i)* the support of existing reactors types regarding the knowledge of the properties and behaviour of fuels and materials applied in them; *ii)* the development of new reactor types. [NB *The special requirements of accelerator-driven systems are treated separately in this report; see Section 3.4.*]

Irradiation facilities for fuel development and testing are mainly facilities that are able to provide a combination of nuclear and operational conditions resembling those found in existing or prospective power reactors. Their source of neutrons is fissions, but the cores and inherent characteristics of such facilities may be quite different from those of the reactors to be simulated. The range of existing reactor types, their further evolution as well as developments for the spectrum of new reactors also require a certain specialisation among research reactors. However, it must be accepted that the conditions provided for fuel development and testing are often a compromise. The Jules Horowitz Reactor (JHR) project (CEA, 2008c; EC, 2008a) is a development of particular note which aims to support the needs of both existing power reactor designs and those of future technology systems.

The needs for and expectations of irradiation facilities are thus, on a worldwide basis:

i) *Support for the whole spectrum of existing and envisaged types of nuclear power reactors*. Most currently operating commercial power reactors are moderated (with light water, heavy water, or graphite) and thus have a more or less thermal neutron spectrum. The same is true for a majority of research reactors, but some of them also have high-power density cores able to generate a high fast neutron flux. Thus, current types of thermal spectrum reactors, including LWRs, PHWRs and graphite-moderated gas-cooled reactors, can be served adequately with existing irradiation facilities such as:

   – ATR, USA (INL, 2007);
   – BR2, Belgium (SCK•CEN, 2008);
   – HANARO, Korea (KAERI, 2001);
   – HBWR, Norway (IFE, 2005; NEA, 2008bb);
   – HFR, Netherlands (NRG, 2008);
   – JMTR, Japan (JAEA, 2008f);
   – OSIRIS, France (CEA, 2008d).

   All of these are included in RTFDB and their totality represents a considerable experience in fuel development and testing. When such reactors are considered for permanent shutdown in the foreseeable future, the reasons are generally a combination of costs of operation, refurbishment needs and age.

   Although some current and planned research reactors are able to provide, locally, a high fast-neutron flux, their natural neutron spectrum differs from those of metal-cooled reactors, *e.g.* those foreseen in the context of Gen. IV development. Some flux tailoring can be applied, but eventually research reactors especially designed to serve their needs will have to be considered, including prototype reactors augmented with ancillary systems required for fuels and materials testing.

ii) *Support for different types of operation and transients*. The dominating mode of operation of current commercial reactors is by necessity and definition "normal operation" which includes regular start-up and shutdown sequences, moderate power changes and steady-state operation. These modes are covered by MTRs with their inherent characteristics. Fast power changes (ramping) are handled in various ways, depending on the design features of the MTR, often by moving the fuel rod closer to the periphery of the core (thus utilising the neutron flux gradient in the reflector) or by locally controlling the neutron flux (and thus test rod power) with a neutron absorber (*e.g.* ^{3}He). Issues related to normal operation are outlined in Section 3.5.1.2 on fuel fabrication and development.

   Specialised research reactors exist for addressing fuel behaviour during reactivity insertion accidents (RIA). These reactors can generate a strong power pulse by quickly inserting reactivity of several $'s worth. Since the beginning of the 1990s, several national and international programmes for studying the response of high burn-up fuel during RIA have been conducted. The reactors involved are mainly the French CABRI reactor (CEA, 2008g), the Japanese NSRR (JAEA, 2008h), and the Russian BIGR (Bibilashvili, 2000) and the Kazakhstan IGR reactors (Asmolov, 1996).

   The Japanese NSRR RIA programme is carried out in the Fuel Safety Research Laboratory in the Department of Reactor Safety Research of JAEA. The objectives of this programme are to:

   – investigate the behaviour of irradiated fuel rods under RIA conditions;
   – determine the fuel rod failure threshold of irradiated fuel rods and to clarify the influences of fuel burn-up;
   – clarify the mode, mechanism and consequence of the failure of irradiated fuel rods;
   – verify the adequacy or modify the safety criteria previously defined for lower burn-up fuel.

Similar objectives apply to the CABRI REP Na Programme (sodium loop) (Papin, 2007) which was initiated in 1992 by the Institut de Protection et de Sûreté Nucléaire, (IPSN) and conducted in collaboration with Éléctricité de France (EDF) and with a participation of the United States Nuclear Regulatory Commission (NRC). Both high burn-up UO_2 fuel and MOX fuel were studied (eight tests with UO_2 fuel and four tests with MOX fuel using mostly refabricated rods from PWR fuel).

A special feature of the NSRR is the ability of this reactor to operate in multiple pulse mode. This has been successfully applied to study the behaviour of high burn-up fuel during power oscillations caused by ATWS (Nakamura, 2004).

The conditions provided by the CABRI, NSRR and IGR/BIGR research reactors are not in all respects representative of those prevailing in commercial nuclear power plants. Some of the main differences are:

- pulse width (often sharper pulse);
- coolant conditions:
  - temperature (<100°C, or LWR);
  - pressure (low pressure);
  - cooling medium (sodium or water);
  - flow (may be stagnant).
- flux shape (short cores).

Some of these characteristics contribute to conservatism of results. For example, pulses which are narrower than calculated for LWRs produce a more adiabatic heat-up and therefore a greater load on the cladding. Likewise, lower coolant temperatures make the cladding material less ductile and more prone to failure. However, questions remain about the effect of departure from nucleate boiling (DNB) on cladding failure, the influence of internal rod pressure, and the possibility of fuel-coolant interactions after failure. The Institut de Radioprotection et de Sûreté Nucléaire (IRSN) (IPSN's successor) therefore decided to replace the sodium loop in CABRI with a pressurised water loop (PWL) and to propose the CABRI International Programme (CIP) to the nuclear industry and regulatory agencies.

Loss of coolant accident (LOCA) testing is mostly done in hot lab facilities (ring compression tests, double-sided oxidation in steam environment) and only to a very limited extent in irradiation facilities. At present, the OECD Halden Reactor Project is carrying out integral in-pile tests on fuel behaviour under LOCA conditions where the decay heat is simulated by a low level of nuclear heating. The primary objective is to observe the overall fuel behaviour under expected and bounding conditions, including the extent and effect of fuel relocation into the ballooning volume.

LOCA bundle tests are very demanding both from a cost and experiment execution point of view.

In-pile LOCA testing of VVER fuel has been carried out on the MIR reactor in Russia with fresh and irradiated fuel rods from power reactors (Goryachev, 1997, 2005).

The high neutron flux at LOHENGRIN permits the performance of in-pile tests of thin actinide targets suffering extremely high radiation damage (of the order of 50 dpa/day). The effects of self-sputtering (radiation-enhanced) diffusion into the backing, etc., are monitored via the fission product energy spectra, while the temperature of the nuclear heated targets is measured with an infrared pyrometer. This system was applied to study the fission-enhanced diffusion of uranium in zirconium and zirconia (Bérerd, 2005).

Although currently not operating, the Transient Reactor Test Facility (TREAT) (ANL, 2008a) at the Idaho National Laboratory is being considered for possible restart. TREAT has unique capabilities to evaluate reactor fuels and structural materials under conditions simulating various types of in-core nuclear excursions and transient under-cooling situations. In the past, TREAT has been particularly useful in the development of high-temperature fast reactor fuels and materials.

3.5.1.2 Fuel fabrication and development

Improvements in LWR fuel fabrication and design have for many years enabled the utilities to reduce operating costs while maintaining high reliability and safety margins. Consideration is given to cost savings through the development of improved and more cost efficient products which can operate at higher ratings and reach higher burn-ups, thus reducing requirements for shutdown time, waste handling and disposal.

Fuels are also being developed for next-generation reactors, although to a more limited extent. Here, considerations are not only in relation to in-core performance, but also the fuel cycle as a whole, including measures for improved proliferation resistance. These developments are the natural domain of research reactors.

A primary objective of LWR fuel development is improved fuel reliability, which utilities regard as the most important feature. Although zero failure is the ultimate aim, one failed fuel rod in one million irradiated is regarded as being achievable. The dominating failure cause is related to debris entering the reactor core after a maintenance shutdown. Also pellet-clad interaction (PCI) failures still occur, for example due to control rod movements. It has been experienced that a small primary failure (pin hole or crack) can cause the development of a very substantial secondary failure away from the location of the first, leading to the release of significant amounts of radioactive products into the primary system of the reactor.

Improvements to fuel designs are coping with a number of fuel reliability issues. Examples for PWRs are bundle bow, cladding corrosion and vibration, and for BWRs cladding corrosion phenomena caused by modified water coolant chemistries. Related research reactor and Lead Test Assembly (LTA) programmes will continue and be needed also in the future.

Increased burn-up is a means to improve fuel utilisation and to decrease back-end fuel cycle costs which may amount to a considerable fraction of the total fuel cycle costs. For extended exposure to be possible, a number of modifications of standard UO_2 fuel have been and are still being introduced. These must be thoroughly addressed in research reactor testing programmes in order to establish the necessary database for safety assessments. Examples are the impact of burnable neutron absorbers (Gd, Er) on fuel behaviour, additives to influence grain size and consequently fission gas release as well as mechanical properties, densification and swelling, high burn-up structure formation, fuel-clad bonding and others. Efforts are also being extended to achieving parity of MOX fuel with UO_2 fuel.

Higher discharge burn-up implies longer residence time in the reactor and thus improved cladding materials have been developed and are also being screened in research reactor studies.

Improved fuel utilisation is supported by low-leakage core-loading patterns where the most burnt assemblies are loaded at the periphery and fresh fuel in the more central regions of the core. Because of this, and also due to higher enrichments, the fuel initially has to sustain higher power, which in turn necessitates an improvement of the critical power ratio (in BWRs) and superior departure from nucleate boiling performance (in PWRs). Substantial improvements have been and are still being achieved, and the permissible power peaking is rather limited by the LOCA criteria.

Operational flexibility, *i.e.* the ability to follow the demand for electricity on the grid, will also be more and more required of nuclear power stations in a deregulated market and a grid with an increasing fraction of less predictable power from wind and photovoltaic sources. This requires fuels that must be able to withstand rapid power changes and the ensuing mechanical and chemical loads. Further developments are needed in the area of improved flexibility.

The items mentioned above benefit from experimental programmes conducted in research reactors and which investigate performance indicators such as fuel temperature, fission-gas release, fuel densification and swelling, and pellet-clad (mechanical) interaction (PCMI) as reflected by axial and diametric length changes. Separate effects as well as integral behaviour studies address:

- thermal performance and degradation of fuel thermal conductivity with increasing burn-up;
- pellet-cladding interaction due to fuel swelling and pellet-clad bonding;
- fission gas release for different types of fuels (standard UO_2, UO_2 with additives, Gd-bearing fuel, MOX fuel) and its influence on thermal performance and rod pressure;

- cladding creep properties for the entire range of exposure;
- cladding creep response to compressive and tensile stresses;
- tolerable rod pressure limits;
- cladding corrosion and hydrogen pick-up.

In-core instrumentation is essential for performance studies of fuels since it provides direct insight into phenomena while they are going on and cross-correlations between interrelated phenomena. For example, in the experiments conducted in the Halden HBWR (IFE, 2005; NEA, 2008bb), fuel rods are usually instrumented with fuel thermocouple, rod pressure transducer for fission gas release assessment, fuel stack elongation detector for measuring densification and swelling, and clad elongation detector for axial pellet-clad interaction (PCMI).

In order to provide the correct thermal-hydraulic conditions, BWR, PWR, VVER and CANDU loop systems are installed in a number of research reactors. They can also be used to study the effect of different coolant water chemistries on cladding and materials behaviour.

Fuel development and testing for high-temperature gas-cooled reactors (HTGRs) have their own special requirements. The SFEAR report (NEA, 2007d) recommends that international collaboration should be "strongly considered due to the importance of fuel performance to HTGR safety, the long lead time and the cost of fuel testing".

Virtually all of the past and ongoing world wide irradiation testing research of TRISO coated fuel particles (CFPs) includes accelerated irradiations in MTRs. Although there was significant large-scale operating experience with these fuels in plants such as Peach Bottom and Fort St. Vrain (USA), Dragon (EURATOM/UK) and the AVR in Germany, accelerated irradiations in MTRs were also required for accident simulation tests to qualify the fuel. Since the understanding of CFP behaviour, failure and fission product release is still incomplete, accident simulation heat-up tests after real time MTR or power reactor fuel irradiations are needed to resolve the issues. The same is true for reactivity events involving a large energy deposition in the fuel over a very short time interval (< 1 s). Some limited testing was conducted in Japan in support of the licensing of HTTR (JAEA, 2008c) for a postulated control rod ejection accident. Further fuel irradiation experiments involving such reactivity insertion events are required in order to understand the margins to failure. Hence, test reactors capable of testing HTGR fuel in steady-state and transient conditions are essential in order that fuel performance can be established.

Inert matrix fuel (IMF), in which plutonium is embedded in a uranium-free matrix, is also worth of mention; this allows plutonium to burn without breeding any new plutonium by neutron capture in ^{238}U and thus a more efficient consumption of plutonium is achieved compared with mixed-oxide fuel (MOX). IMF fuel is being tested in the Halden reactor (Hellwig, 2003).

As is noted elsewhere in this report, Gen. IV reactors are intended to burn minor actinides (MA) generated in thermal reactors. New fast reactors will be supplied with special MA-bearing fuels that should meet the Gen. IV goals, as identified in the GIF Roadmap:

- potential to operate with a high-conversion fast-spectrum core for increasing resource utilisation;
- capability of efficient and nearly complete consumption of transuranium elements as components of the fuel;
- high level of safety obtained by the use of innovative and reliable solutions including passive safety measures;
- enhanced economics achieved with the use of high burn-up fuels, fuel cycle (*e.g.* disposal) benefits, reduction in power plant capital costs and lower operation costs.

The current reactor concepts describe different MA-bearing fuel designs with ceramic (carbides, oxides and nitrides) and metallic matrices. Cladding materials include oxide dispersion strengthened (ODS) steel and SiC for formats involving pins and ceramic matrices (*e.g.* SiC, ZrC, TiN) for dispersion fuels in plate or block format (Babelot, 2006; Mitchell, 2006). Testing of some metallic MA-bearing fuel has already been performed at ANL (Meyer, 2002), and irradiation tests of oxide fuel pins are planned (Babelot, 2006) in the Phénix (CEA, 2008e), JOYO (JAEA, 2008e), BOR-60 (RIAR, 2008) and HFR (NRG, 2008) reactors. After the selection of potential fuel types and solving the problems of MA-bearing fuel fabrication (see Section 3.5.2.2) the experimental investigation of high burn-up fuel should be carried out in test reactors with fast neutron spectra. The testing of fuels should reach very high burn-ups (250 GWd/tU), the different mechanisms leading to the loss of cladding integrity should be examined under steady-state and transient conditions and the potential re-criticality problems should be analysed.

It is worth mentioning at this stage the International Fuel Performance Experiments (IFPE) Database (NEA, 2008oo). The aim of the project is to provide a comprehensive and well-qualified database on zirconium clad UO_2 fuel for model development and code validation. This work is carried out in close co-operation and co-ordination between OECD/NEA, the IAEA and the IFE/OECD/Halden Reactor Project. More details are given in Section 4.5.3.

3.5.1.3 Hot cells

Hot cells are the basic tools/equipment for the examination of irradiated materials including fuel. Some hot cells are capable of receiving NPP full-length fuel rods or sometimes fuel assemblies and are able to provide primary information on the state of the irradiated fuel, while some others can handle only short fuel rods from test reactors.

Hot cells are equipped with special instruments for:

- the investigation/examination of irradiated materials (*e.g.* SEM);
- carrying out experiments (*e.g.* mechanical testing) with irradiated fuel.

The HOTLAB project of the EU's 6^{th} Framework Programme established a database (SCK•CEN, 2008c). The database is hosted by the SCK•CEN in Belgium and provides free access to information on 25 facilities. The information accessible includes contact details for each site together with information on the specific techniques that are available, *e.g.* burn-up, leak test, rod puncture, neutron radiography to name just a few.

All the facilities listed in this HOTLAB database are included in RTFDB. However, for the full information on the respective sites, the reader is referred to the more extensive information available via (SCK•CEN, 2008c).

The aim of the programme is to share information and to encourage co-operation both between HOTLAB partners and to a wider community in support of nuclear energy safety and waste management goals.

The HOTLAB database is limited to a specific European remit; while reasonably extensive it does seek further information from other European laboratories outside the HOTLAB project in order to extend its coverage.

One of the outputs of the HOTLAB Project was a report which addressed "Present Hot Cell Situation and Needs" (EC, 2005). Regarding the current situation it concludes that current work covers a broad range of activities supporting safe and economic operation of nuclear plants plus some work for new reactor systems. The emphasis is on issues relating to: *i)* fuel cycle; *ii)* lifetime assessment of structural core components.

The report notes a substantial reduction in the total number of hot cell laboratories in Europe in recent times, but observes that that many are indicating high levels of utilisation of basic analytical techniques and hence implying a limited capacity to absorb any significant increase in demand.

For the near term, the report concludes that there will be continuation of present work aimed at improvements in the safety and economy of the current reactors.

The longer term is seen to be strongly dependent on the research towards new reactor systems. However, it notes that most of future programmes are likely to be organised on a world basis with consequent technical and political implications.

The IAEA also has a database on post-irradiation facilities within its Integrated Nuclear Fuel Cycle Information Systems (INFCIS) package (IAEA, 2007a), which also includes information on:

- Nuclear Fuel Cycle Information System (NFCIS);
- World Distribution of Uranium Deposits Database (UDEPO);

- Nuclear Fuel Cycle Simulation System (VISTA);
- Minor Actinide Property Database.

The Post-irradiation Examination Facilities Database (PIE) portion of INFCIS is derived from a worldwide catalogue of such facilities that the IAEA issued in the 1990s. It includes a complete survey of the main characteristics of hot cells and their PIE capabilities. See Section 3.5.1.4 for more details.

There is a further category in the NFCIS database which is "commercial", but as this is unlikely to have any research interests, the 501 facilities listed there have not been reviewed for inclusion in RTFDB.

Under current development there is an activity to merge the European HOTLAB database into the IAEA PIE/NFCIS database in order to produce a new version which is as complete as possible. At the same time, it can be noted that the annual HOTLAB congress which used to be purely European in scope has, since 2007, been open to worldwide attendance. In summary, good co-ordination has developed between the IAEA and the HOTLAB project with the aim of making good use of the synergies of their earlier independent activities.

3.5.1.4 Post-irradiation examination (PIE)

As noted earlier, the IAEA INFCIS database (IAEA, 2007a) has information on PIE facilities; as of September 2008 it lists 31 facilities. Similarly, the EU's HOTLAB project and database (SCK•CEN, 2008c) were also mentioned earlier (see Section 3.5.1.3). Many of the facilities listed in that database are, in fact, for post-irradiation examination and the facility details provide information on the specific techniques that are available, *e.g.* burn-up, leak test, rod puncture, neutron radiography.

The EU HOTLAB project report (EC, 2005) contains a detailed review of the purposes and current techniques employed in PIE as viewed from a European perspective. For fuel itself this includes: *i)* post-operational surveillance (fission gas release, thermal properties, etc.); *ii)* analysis of reactivity insertion accidents (RIA), LOCA tests, etc. (NEA, 2007d); *iii)* failed fuel examinations; *iv)* development of MOX and other advanced fuels; *v)* refabrication of irradiated rods. For testing irradiated clad materials and fuel the mechanical properties and fission product release are obvious areas of work.

The HOTLAB project report notes that it expects that, in the future, novel fuels (*e.g.* doped fuel pellets) and claddings will be tested in separate effect experiments, for example, RIA, LOCA, transient behaviour of fission gas release and PCI. These experiments are conducted out of pile since it is more cost effective than integral tests performed in-pile. These separate effect tests can be used to help select the best candidate materials.

In addition, new systems such as ADS, new reactor designs such as being considered for Gen. IV, and programmes such as GNEP will bring new demands on PIE techniques and facilities. Developments in the fusion community requiring the analysis of plasma facing and divertor materials such as beryllium, tungsten and carbon reinforced carbon and copper alloys may have some cross impact on the fission power utilisation of a limited PIE resource.

While there is a natural interest in new developments, it should be noted that the closure of reactors reaching the end of their commercial lives will also lead to the ability to analyse real irradiated components in order to confirm lifetime prediction techniques.

The NEA has recently established an Expert Group on Assay Data of Spent Nuclear Fuel (NEA, 2008J). This NEA expert group was set up to pursue two activities in parallel: updating the NEA spent fuel isotopic composition database (SFCOMPO) (NEA, 2008tt), and writing a state-of-the-art report on the assay data of spent nuclear fuel. The major assignments of the expert group include:

- analysing the SFCOMPO database in order to assess the current situation and the need for new experimental data;
- collecting new isotopic composition data from post-irradiation examination (PIE) and incorporating them and their associated operating histories/data into the SFCOMPO database. The expert group will review the format of the SFCOMPO database;
- archiving original reports on any PIE data included in the SFCOMPO database, as well adding data references that were used in the original development of the data.

Further details on the Expert Group's aims and activities, including papers at a special session at the 8th International Conference on Nuclear Criticality Safety, 28 May-1 June 2007, St. Petersburg, Russia (NEA, 2008i) can be accessed via the Expert Group's pages at the NEA website (2008J).

3.5.2 Fuel cycle chemistry

The development of peaceful utilisation of nuclear energy to the level of ~15% of total world electricity generation has been ascribed to an effective, safe performance of not only nuclear power plants but also all the fuel cycle facilities. Generally fuel cycle chemistry supports a wide range of research activities.

In the front-end cycle, uranium mill tailings are receiving more attention because they pose a potential hazard to public health and safety (US NRC, 2006). Total fuel cycle cost including treatment of such waste could be minimised by employing more efficient uranium recovery and purification processes which would require less chemical additives and prevent release of the harmful heavy metals in ores to the environment.

Within the scope of the NEA NSC, current topics of significance in this area are:

- recycling technologies of uranium (U) plus plutonium (Pu) plus minor actinides (MAs), which are, principally, Np, Am and Cm – as well as current U and Pu interests;
- MOX developments; fuel fabrication, in particular, is key;
- reprocessing including partitioning is also important to support recycling;
- waste treatment technologies corresponding to new recycling scenarios and schemes will also need to be developed, *e.g.* ceramics as an alternative to glass form.

3.5.2.1 Reprocessing and MOX fabrication

A reprocessing operation is a pivotal part of the closed nuclear fuel cycle and has been operated so far aiming to:

i) maximise the utilisation of nuclear energy from U by recycling fissile U and recovering/ reusing Pu, *e.g.* as MOX fuels;

ii) minimise the volume of high-level waste (HLW) by solidifying it as a waste-form, *e.g.* vitrification, for long-term storage/disposal in a geological repository.

[NB *Actual technological/societal maturity shows, however, only a minor fraction, ~10-15%, of the net Pu produced in power reactors in the world is being reused as MOX fuel for the production of electricity*.]

As described in Section 3.5.1.2, *Fuel fabrication and development*, high burn-up is an important issue in the near term in order to improve fuel utilisation. A MOX powder is produced by mechanical mixing of UO_2 and PuO_2 or by treatment of U-Pu mixed nitrate solutions. As a PuO_2 particle localised within MOX fuel results in the formation of an undesirable hot spot during reactor operation, a more homogeneous mixing of U and Pu at the atomic scale is desirable in order to achieve higher burn-up. Recently CEA has developed an innovative co-conversion process based on co-precipitation of mixed U(IV)–Pu(III) oxalates, which is one of the key operations in the CEA-AREVA NC COEX™ process (Grandjean, 2007). The mixed An(IV)-An(III) crystallographic site is responsible for the homogeneous distribution of actinides in the co-precipitates. It has advantages in producing more homogeneous mixing compared with mechanical mixing. Extensive studies to implement the process when MAs are involved have been carried out at the ATALANTE research complex (CEA, 2008).

In comparison with the improvement of current U-Pu oxide fuel, development of MA bearing fuel, *e.g.* U-Pu-MA oxides, is a longer-term objective. In Europe, experimental rods and targets bearing MA can be manufactured at ITU's MA-lab (ITU, 2008a), ATALANTE (CEA, 2008), and Petten. CEA LEFCA in Cadarache also has the capacity to handle limited quantities of MA [for more details of LEFCA see the HOTLAB report (EC, 2005)]. In Japan, JAEA has developed MA oxide fuel at the Alpha Gamma Facility (AGF) (JAEA, n.d.). In the USA in the 1980s, Oak Ridge National Laboratory (ORNL) had developed the Modified Direct Denitration (MDD) process to produce MOX powder from co-processed U Pu nitrate solutions and they have recently re-examined it on a glove box scale for co-conversion of mixed-actinide nitrate solutions (Walker, 2007).

Concerning MOX reprocessing, the first reprocessing of spent MOX fuel (Pu 4%, 35 GWd/t) was undertaken in France on a larger scale (10 t) using a continuous process at the La Hague plant in 2004. The main problem of fully dissolving PuO_2 was overcome. However, at present the general policy is not to reprocess spent MOX fuel, but to store it and await the advent of fuel cycle developments related to Gen. IV reactor designs (WNA, 2008).

3.5.2.2 Partitioning and required technologies

Growing concerns about the deterioration of the global environmental and the risk of nuclear proliferation has led to the evolution of additional requirements for the future sustainable utilisation of nuclear energy. Responding to these contemporary requirements, partitioning and transmutation (P&T) is a radioactive waste management option complementary to geological repositories being investigated among OECD and non-OECD countries. Within the OECD, France, Japan, Korea, the United States and other parts of the EU are carrying out R&D on P&T through the GNEP, Gen. IV and EURATOM Framework Research Programmes. [NB *Transmutation is discussed in Section 3.4.*]

In a partitioning process most of: *i)* the transuranics (TRU: Np, Pu, Am, Cm); *ii)* long-lived fission products (LLFP: ^{129}I and ^{99}Tc); *iii)* heat-generating nuclides (^{90}Sr and ^{137}Cs), are partitioned by chemical separation in addition to U. During the first 300 years after discharge of spent UOX fuel from a reactor, the thermal burden of the HLW on the repository principally resides with the ^{90}Sr and ^{137}Cs and this restricts the design conditions of the repository. Consequently, removal of these heat-generating nuclides from the HLW can relax the specifications for a repository. After about 300 to 500 years the HLW radiotoxicity is dominated by the MAs (Np, Am, Cm), while after more than 200 000 years it reaches the uranium ore radiotoxicity threshold and this is regarded as having no potential impact on the environment. Thus the removal of all the MAs from HLW would markedly reduce the long-term radiotoxicity of the waste and make it below that of the original uranium ore after ~3 000 years.

A particular issue is curium, which is a minor actinide alongside Np and Am. Among the Cm isotopes generated by neutron capture in nuclear reactors, the most abundant by far is ^{244}Cm (half-life: 18 years) which accounts for 92.6% of the curium in UOX fuel with a 33 GWd/t burn-up (Sénat, 2008). During the first 300 years after discharge of spent fuel from a reactor ^{244}Cm is also of relevance, along with ^{137}Cs and ^{90}Sr; though its significance is as a strong α and neutron emitter. Hence, the presence of Cm in facilities dedicated to MA-loaded nuclear fuels would have an important radioprotection impact. An alternative solution could be to separate Cm and to store it for decades, in order to await the decay of ^{244}Cm into ^{240}Pu.

Over the last couple of decades, considerable scientific and technical effort has been devoted to developing partitioning processes through domestic and international projects: France [SPIN (Salvatores, 1995)]; Japan [OMEGA (Mukaiyama, 1999; Minato, 2007)], USA [AFCI (US DOE, 2008)], GNEP (2008), bilateral co-operations and EURATOM Framework Programmes [NEWPART (Madic, 2000), PARTNEW (CORDIS, 2008a), EUROPART (CEA, 2008a), CALIXPART (CORDIS, 2008), PYROREP (CORDIS, 2008b)]. Significant scientific and technical progress has been made. [NB *More information on EUROPART is given later in this section.*] In Europe, the newest R&D programme relating to partitioning studies has just started under the 7th EU Framework Programme (FP7) (CORDIS, 2008d).

In a partitioning strategy, separation of MAs (or TRU) is a highly important step. Among the elements in the MAs, Np can be partitioned within a modified Purex process by controlling the process conditions. Recently, finely controlled separation of Np has been demonstrated on a laboratory scale at the ATALANTE facility (CEA, 2008) using genuine spent fuel.

Partitioning of Am and Cm is a much more difficult task since it necessitates discrimination of Am and Cm from lanthanides (Ln), whose chemical properties are very similar but which occur in amounts more than 20 times larger. Extensive research has been undertaken on the MA separation methods based on hydrometallurgical and pyrometallurgical processes.

Concerning hydrometallurgical processes, the separation processes for TRU, *e.g.* DIAMEX (France), TRUEX (USA, Japan), DIDPA and TODGA (Japan) and TRPO (China) have been developed and tested in the relevant "hot" laboratories. As these processes separate TRU and Ln together from the rest of the fission products, an additional step is necessary to isolate the Am-Cm group from the Ln. Several techniques have been proposed and tested on highly active process streams using selective extractants (Cyanex 301, C5-BTBP, etc.) or the Talspeak technique with conventional reagents (HDEHP-DTPA). The major drawback of these methods is the degradation of the organic extractants and reagents caused by radiolysis and hydrolysis. Gamma irradiators are still indispensable tools to study the stability against radiolysis. The *Module Avancé de Radiolyse dans les Cycle d'Extraction-Lavages* (MARCEL) at the CEA Marcoule site provides a unique gamma irradiation field which simulates operating conditions of a reprocessing plant (*e.g.* aqueous/organic emulsion flow). JAEA has studied radiolysis of amide extractants using the ordinal ^{60}Co facility at its Takasaki site (Sugo, 2007).

Concerning pyrochemical processes, because fuels (or target materials for transmutation by fast reactors or ADS) take various forms – as oxide, metal, nitride, carbide or fluoride, etc. – a suite of pyrochemical techniques has been investigated. Examples are: the classical fluoride volatility method, various methods using molten salts for metallic fuels, and dissolution and precipitation in a chloride bath for oxide fuel. The methods using molten salts are classified as shown in Table 3.

Table 3: Various methods and processes using molten salts

Purpose of method	Explanation of processes
I. Uranium recovery (on a solid cathode)	a) Metal (anode), via LiCl-KCl melt $\rightarrow$ U metal on solid metal cathode (SMC) b) Oxide, via LiCl-KCl melt and chemical reduction by Li -Li_2O $\rightarrow$ U metal ppt. Put ppt on an anode $\rightarrow$ U metal on SMC c) Oxide (anode), via NaCl-KCl (or – CsCl) melt and chlorination/dissolution by Cl_2 gas bubbling $\rightarrow$ UO_2 on SMC d) Oxide (anode), direct electro-reduction in molten salt $\rightarrow$ U metal on SMC
II. TRU (plus Ln) recovery	a) Extracts Pu or TRU (- Ln) in liquid metal cathode (LMC), after 1-a) or 1-b), via LiCl-KCl melt b) Deposit PuO_2 or MOX on SMC, or TRU (-Ln) oxide precipitation, after 1-c), via NaCl-KCl or NaCl–CsCl melt and Cl_2 + O_2 gas bubbling c) Direct electro-reduction of oxide fuel (anode) via molten salt $\rightarrow$ TRU metals in LMC
III. TRU-Ln grouped separation	Partitioning of TRU and Lns between two immiscible phases; molten salt and liquid metal.
IV. Other	Fluoride molten salt (LiF-NaF and LiF-NaF-KF), Cs/Sr removal, waste treatment, etc.

Since these proposed hydro- and pyrochemical methods are not yet being operated on a large scale, a considerable chemical engineering effort will be needed in a phase of scaling-up the laboratory methods first to pilot scale and subsequently to industrial prototype; the up-scaling factor from laboratory to industrial scale is of the order of 10 000. Considering the strict regulation limiting the quantities of MAs to be handled in shielded facilities and their construction cost, possessing R&D facilities which fulfil the requirements on MAs is becoming a determining factor among the P&T oriented countries. [Some examples of facilities that do meet the requirements are: ATALANTE in France (CEA, 2008), ITU (2008) in Germany, NUCEF and CPF in Japan (JAEA, 2008g, 2008i), DOE laboratories, *e.g.* FCF (Benedict, 2007), in the United States.] For example, the new MA laboratory of JRC-ITU has an authorisation for maximum 150 g ^{241}Am and 5 g ^{244}Cm. Further, the capability of the attached analytical laboratories, for measurement of chemicals and radioactivity and identification/ speciation of samples, is an indispensable factor.

3.5.2.3 Partitioning processes under development

Focusing on the facilities involved, the main achievements in several countries are briefly elaborated upon below.

France

R&D by CEA pertaining to partitioning has been carried out in the frame of the Actinex project (Musikas, 1991). The most important of the latest results are shown below.

i) Np and Tc partitioning in the Purex process: a validation experiment has been conducted in a new shielded process line, the *Chaîne Blindée Procédé* (CBP) hot cell at ATALANTE (CEA, 2008) using 13 kg of LWR irradiated fuels. The test performance was satisfactory.

ii) The DIAMEX process for co-extraction of An(III) and Ln(III) was verified in the CBP line.

iii) The DIAMEX-SANEX process for group separation of Am(III)-Cm(III) from Ln(III) has also been conducted in the CBP line. The results showed ~99.9% MA recovery with less than 0.3 wt.% of Ln remained in MA fraction and less than 0.06% loss of MA in the Ln fraction.

iv) The CCCEX process for Cs separation has been demonstrated using a genuine high active raffinate.

CEA has also been conducting R&D work to establish a novel concept "Grouped EXtraction of ActiNides" (GANEX) consistent with the Gen. IV reactor fuel cycle. It is to be demonstrated at La Hague from 2008 (WNA, 2008e) as part of a French-Japanese-United States Global Actinide Cycle International Demonstration (GACID).

Regarding research on pyrochemical processes based on electrolysis, the conditions of both Methods I-a and II-a listed in Table 3 were investigated. In particular, Method II-a was assessed with Am in ATALANTE. A test with liquid Cd cathode was performed for LiCl-KCl-$PuCl_3$ (1.55 wt.%)/$NdCl_3$ (0.98 wt.%) in a glove box. The recovery of Pu in liquid Cd was not satisfactory (~79%) and was accompanied by around a quarter of the Nd initially present. Since the performance of An(III)/Ln(III) partitioning depends on the affinity of the metallic solvent for An and Ln, other liquid metals (*e.g.* Ga) are being studied as an alternative. Concerning an An recovery based on salt/metal extraction, a feasibility of Pu-Am separation from Ce-Sm has been demonstrated at ATALANTE in a system of LiF-AlF_3 and liquid AlCu. For carbide material (after conversion to oxide, dissolved in LiF-AlF_3 and fluorinated by hydro-fluorination) a similar investigation has been carried out. Fundamental data including distribution coefficients and separation factors were obtained.

CEA is establishing *L'Institut de Chimie Séparative de Marcoule* (ICSM) at its Marcoule site. The ICSM has a function as a training and education centre for university students studying basic chemistry as well as technical and scientific support to the ATALANTE facility developing an advanced reprocessing process.

Japan

R&D for the OMEGA P&T programme (Mukaiyama, 1999; Minato, 2007) has continued for more than 20 years in Japan. The main organisations promoting P&T are the JAEA and the Central Research Institute of Electric Power Industry, CRIEPI (2008). Partitioning processes based on aqueous chemistry have been studied and demonstrated by JAEA and pyrochemical process mostly by CRIEPI. This R&D has been progressing in parallel with that of reprocessing technology for the Fast Breeder Reactor (FBR) fuel cycle. The main results and facilities involved are briefly explained below.

i) The Four-group Partitioning (DIDPA) process: This process was developed for the double-strata fuel cycle and established in the 90s (Morita, 1999). Tests at the NUCEF facility (JAEA, 2008g) using concentrated Purex rafinate solution exhibited a recovery yield of Np, Am and Cm by extraction with DIDPA higher than 99.95%; ^{99}Tc and Sr-Cs products were recovered by precipitation and ion exchange, respectively.

ii) New Extraction System for TRU Recovery (NEXT) process: Recently, the JAEA has proposed and developed the NEXT process, which consists of crystallisation of uranyl nitrate hexahydrate, simplified tri-butyl phosphate (TBP) extraction for U-Pu-Np coextraction, and separation of Am-Cm by extraction chromatography which has replaced a solvent extraction process called SETFICS. Each of the processes has been tested at the Chemical Processing Facility (CPF) (JAEA, 2008i) using a solution of dissolved spent fuel and showed satisfactory results.

iii) Alternative methods: Several options to the NEXT process are also under development: *e.g.* solvent extraction of U with monoamide, total recovery of TRU with TODGA (Tachimori, 2002).

As for research on pyrochemical processes, after basic studies for a long period, CRIEPI pursued the experiments on Method I-a as in Table 3 at a scale of 1 kg U, and a system using Method II-a with Pu were tested successfully. A system using Method III-a with liquid Cd or Bi metal at 450°C has been investigated. For oxide fuels (UO_2, PuO_2, Am_2O_3, NpO_2 as well as MOX pellets) a feasibility of the reduction by Li or electrochemical means to metal was examined. Chlorination of high-level liquid waste (HLLW) coming from Purex-type reprocessing by a similar process (Method I-c) but with carbon reductant is being studied in a hot cell in ITU. A multi-stage extraction using counter-current flow for Method III is being examined by using a salt stream from the electro-refining of chlorinated HLLW.

With regard to nitride materials, JAEA has pursued the study on direct electrolysis of UN, NpN, PuN and (U, Pu)N, and production of AnN in liquid Cd with N_2 gas bubbling. Oxide electro-winning from a modified RIAR[16] process and waste treatment processes (sodalite, borosilicate glass, glass bonded sodalite) were developed. A new modular facility for TRU pyrochemistry, TRU-HITEC, composed of three hot cells, has been completed in JAEA (2008m).

Korea

Since 1997, Korea Atomic Energy Research Institute (KAERI) has been developing an Advanced Spent Fuel Conditioning Process (ACP) to reduce the volume and heat load of spent fuel. The ACP is a pyrochemical process that converts a spent oxide fuel into a metal form (*e.g.* uranium ingot) in a high-temperature molten salt bath. The Advanced Spent Fuel Conditioning Process Facility (ACPF) was constructed to demonstrate the technical feasibility of this process; two inactive demonstrations using simulated fuels were undertaken in 2006 (Lee, 2007). ACPF consists of six major devices: a slitting machine, vol-oxidiser, reduction reactor, smelter, waste salt treatment device and safeguard neutron counter.

USA

DOE conducts partitioning R&D under the GNEP programme at its national laboratories. Laboratory and bench-scale work is ongoing at Argonne National Laboratory, Idaho National Laboratory, Los Alamos National Laboratory, Oak Ridge National Laboratory, Pacific Northwest National Laboratory, Savannah River National Laboratory, and Sandia National Laboratories. Partitioning methods are being developed for treatment of LWR and fast spectrum reactor spent fuels. The research covers work on oxide and metal fuel separation processes.

Near-term activities include:

- demonstration of U/Pu/Np separations;
- iodine (and other radioactive gas) capture and waste form development;
- Am/Cm separations, waste/product form development;
- fission product process and waste form development;
- electro-metallurgical process waste/product purity development.

Long-term activities include:

- more complete separations and waste form development;
- equipment design, testing and optimisation.

The research includes efforts to identify reasonable and plausible waste forms and disposition paths for all process wastes with a goal of no liquid wastes requiring long-term storage.

Partitioning technology options under evaluation include:

- aqueous processing, such as UREX+, developed for LWR fuels (see Table 4);
- electrochemical processing, also known as pyroprocessing, developed for fast reactor metal and oxide fuels.

Neither of these approaches separates pure Pu. The aqueous process uses an aqueous acid to dissolve used fuel and the electro-refining approach uses a chloride salt. Laboratory investigations are working to achieve a >99.9% removal of TRU and fission products (Cs, Sr). As of 2006, laboratory-scale demonstrations of UREX+ extraction processes have demonstrated recovery efficiencies of 99.992% (U), 99.99% (Np), 99.99% (Pu), 99.99% (Am), 99.999% (Cm), 99.2% (Cs), 99.9% (Sr), and 98.3% (Tc) (Laidler, 2007).

Since 1996, spent fuel from the decommissioned EBR-II reactor at Idaho National Laboratory (INL) has been conditioned using electrochemical processing. In this process, the irradiated metallic fuel is chopped and anodically dissolved in molten LiCl-KCl salt. Uranium migrates to a metallic cathode, and the transuranics are left in the salt together with the active metal fission products to be incorporated

16. Research Institute of Atomic Reactors (RIAR).

Table 4: Examples of potential UREX+ aqueous processing options (Laidler, 2007)

Process	Product #1	Product #2	Product #3	Product #4	Product #5	Product #6	Product #7
UREX+1	U	Tc	Cs/Sr	U+TRU+Ln	FP		
UREX+1a	U	Tc	Cs/Sr	U+TRU	All FP		
UREX+2	U	Tc	Cs/Sr	U+Pu+Np	Am+Cm+Ln	FP	
UREX+3	U	Tc	Cs/Sr	U+Pu+Np	Am+Cm	All FP	
UREX+4	U	Tc	Cs/Sr	U+Pu+Np	Am	Cm	All FP

In all cases, iodine is removed as an off-gas from the dissolution process.

Processes are designed for the generation of no liquid high-level wastes.

U: contributor to dose rate, and the mass and volume of high-level waste.

Tc: long-lived fission product, minor contributor to long-term dose.

Cs/Sr: primary short-term heat generators, affect waste form loading and repository drift loading.

TRU: Pu, Np, Am, Cm, primary long-term dose rate contributors.

Ln: lanthanide, fission products.

FP: fission products (other than caesium, strontium, technetium, iodine and the lanthanides mentioned explicitly).

into a ceramic waste form. Group transuranic separations using a liquid cadmium cathode have also been demonstrated. Noble metal fission products (including Tc) are melted together with stainless steel cladding hulls to produce a metallic waste form (Shelly, 2005). A treatment rate of 159 kg/year has been reached which is greater than laboratory-scale (US DOE, 2006). Separation among transuranic elements and of Cs and Sr has not been demonstrated. Laboratory-scale processes for separation of Tc and I have been demonstrated and work is ongoing to improve separation efficiencies (US DOE, 2006). The new separations technologies are being tested in the Fuel Conditioning Facility [FCF (Benedict, 2007)] at INL that contains two electrorefiners: Mk-IV for treating spent EBR-II fuel and Mk-V, with a larger throughput, for showing technical feasibility. An engineering scale metal waste furnace is being installed in the Hot Fuel Examination Facility (HFEF) to recycle the electrorefiner salts (Benedict, 2007).

The partitioning technologies must be demonstrated at engineering scale before they can be considered feasible for eventual commercial use. Construction of a pilot facility, the Advanced Fuel Cycle Laboratory (AFCL), is planned for the mid-term to undertake engineering-scale studies of aqueous and electrochemical partitioning technologies.

Russia

Concerning hydrometallurgical partitioning technology, the notable efforts by the Khlopin Radium Institute (KRI, 2008) to establish processes for simultaneous separation of ^{90}Sr and ^{137}Cs for large-scale applications should be mentioned. A counter-current solvent extraction process was developed at the Mayak-RT1 reprocessing plant. Treatment of over 1 000 m^3 of Mayak HLW, resulting in recovery of ~560 PBq of ^{137}Cs-^{90}Sr, was achieved. Since the nitrobenzene and nitro-aromatics diluents used are regarded as unacceptable due to environmental, health and plant safety issues, KRI and INL (USA) are collaborating on improving the process to overcome these environmental issues and to incorporate an extraction process. This new process for the combined extraction of ^{137}Cs, ^{90}Sr and actinides was named the Universal Extraction (UNEX) process. It has been successfully demonstrated at laboratory scale (waste volumes up to 100 litres) in a centrifugal contactors line using radioactive waste from the United States and Russia.

Regarding research on pyrochemical processes, as a result of abundant experience since the 1950s, a variety of processes have been envisaged and investigated. Along with the P&T strategy, the Dimitrovgrad Dry Process (DDP) for MOX fuel has been developed. The main steps of DDP are composed of Methods I-c and II-b as shown in Table 3. Three alternatives were considered at RIAR based on DDP: recovery of UO_2, recycle of only PuO_2 and recovery of MOX. The demonstration experiment was carried out by using irradiated fuel assemblies from the BOR-60 reactor, and the most important result of the experiment was the high recovery rate of Pu: 95.6%. In 1992 RIAR began research under the DOVITA programme: Dry reprocessing, Oxide fuel, Vibropac, Integral, Transmutation of Actinides. Plenty of experiments on Np recycle were carried out and the electro-co-deposition of Np-Am with UO_2 by Methods I-c and II-b is ongoing. In Russia many institutes and organisations have been participating in R&D for a pyrochemical closed fuel cycle. Various research subjects are progressing: reductive extraction of An- and Ln-fluorides in Li, Be/F and Li-Na, K/F systems; selective deposition of An oxides from a fluoride melt (LiF-NaF eutectics); and oxidation reduction reprocessing in molten alkali nitrates.

China

A reprocessing pilot plant was commissioned in 2003. A large-scale commercial reprocessing plant is planned to be built around 2020. Although China has a closed fuel cycle strategy, it is anticipated that spent fuel will accumulate approximately 4 000 t(HM) by 2010. Partitioning studies have been carried out mainly by the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University and the China Institute of Atomic Energy (CIAE). INET proposed a Total Partitioning (TP) process, which consists of a trialkyl phosphine oxides (TRPO) process for separating TRU and ^{99}Tc, an extraction process for Sr and an ion exchange process for Cs. CIAE developed the Podand Amide Process, in which An and Ln are extracted with one extractant, and Sr(II) with another (Zheng, 2004).

India

Along with a fast reactor fuel cycle programme based on thorium resources, the Indian Department of Atomic Energy (DAE) has been proceeding with R&D related to the Prototype Fast Breeder Reactor (PFBR) (IGCAR, n.d.; WNA, 2008b; BARC, 2008) fuel cycle utilising MOX fuel. Hence the Indira Gandhi Centre for Atomic Research (IGCAR, 2008) and Bhabha Atomic Research Centre (BARC, 2008a) have started an ambitious collaborative programme. R&D issues in reprocessing include development of an advanced Purex process including control of Np and Tc, HLW partitioning for MA, Sr and Cs. A pilot plant, the Lead Mini Cell (LMC), was constructed in IGCAR. LMC is provided with a leak-tight stainless steel containment box with sufficient lead shielding and can treat β,γ radioactivity of the order 1 PBq.

As for research on pyrochemical processes, a laboratory-scale facility has been set up and has operated since the 1990s at IGCAR; it has been used to study various chemical steps of a molten salt electrorefining process for metallic alloy fuels. More recently, the feasibility of electrorefining of UO_2, method I-d in Table 3, has been established.

Europe

The Integrated Project "EUROPART" (EUROpean research programme for the PARTitioning of minor actinides from high active wastes issuing the reprocessing of spent nuclear fuels) (CEA, 2008a) operated within the 6th Framework Programme of the EU and was completed on 30 June 2007.

The research undertaken within EUROPART concerned the partitioning of long-lived radionuclides (LLRN) contained in nuclear wastes from reprocessing. After separation, the LLRN will be destroyed, *i.e.* converted to short-lived or stable nuclides by nuclear means (the so-called P&T strategy) or conditioned into stable dedicated solid matrices (the P&C strategy). The target elements were the actinides Am to Cm for wastes from reprocessing of UOX or MOX fuels, and U to Cm from reprocessing of spent fuels and targets [see the nuclear double strata concept (Condé, 2002; Madic, 2004)].

Five Work Packages related to the separation techniques involving hydrometallurgy, while four further Work Packages involved those using pyrometallurgy. More detail on the hydrometallurgy and pyrometallurgy methods is given on the European Commission Research website relating to EUROPART (EC, 2008). More information on the specific Work Packages and their aims can be found on the EUROPART website itself (CEA, 2008a).

As well as the scientific, technical and financial issues of EUROPART, other issues such as ethical matters, science and society, gender issues, and benefits for Europe were incorporated into the programme. Training and education of the young researchers participating in EUROPART was also regarded as an important part of the project.

As noted in the paper on EUROPART at the "Separation for the Nuclear Fuel Cycle in the 21st Century" meeting at Anaheim (Madic, 2004), the facilities relating to the programme cover:

- computer, organic synthesis, analytical chemistry and structural chemistry laboratories, including those connected to large facilities, such as nuclear reactors (for neutron scattering experiments) or synchrotrons (for X-ray absorption spectroscopy studies);

- radiochemical laboratories for studying α, β, γ radionuclides (involving the use of glove boxes and hot cells to carry out hot tests);
- irradiation facilities for studying the resistance of the ligands towards radiolysis.

Further details on these types of facility will be found in the relevant section of this report and via the entries in the RTFDB.

Other European programmes worth a brief mention are:

- *ACTINET-6* (2008). ACTINET is designed to take steps in order to bring both research infrastructures and human expertise in Europe to an adequate performance level, thus contributing to the creation of the European Research Area in the fields of physics and chemistry of actinides. The three lines of action are:
  - stimulating the emergence of a European infrastructure policy by pooling the major actinide laboratory facilities, see (ACTINET, 2008a);
  - promoting excellence by supporting ambitious shared research programmes, taking advantage of the pooled facilities;
  - increasing the attractiveness of actinide sciences among European students and young researchers, and allowing the next generation of actinide scientists and engineers to gain hands-on experience as part of their training.

  The ACTINET consortium currently includes 30 institutions from 13 European countries, including the European Commission, DG Joint Research Centre Institute for TransUranium Elements, as members; these range from large national laboratories to university departments, thus combining major experimental facilities, academic and applied research expertise and capacities, and training experience.
- *RED-IMPACT* (2008). This project investigates the impact of partitioning, transmutation and waste reduction technologies on final waste disposal. While it has no associated facilities, it is worth mentioning as it should provide information on what might need to be done in the future.

IAEA

While being a very broad programme, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of the IAEA looks at the whole range of innovative nuclear technologies for both reactors and fuel cycles including the environment, spent fuel and waste, but also institutional aspects and infrastructure. INPRO is aimed at examining the prospects of nuclear technology against this very broad background (IAEA, 2008b).

Further information on facilities relating to fuel cycle chemistry, but linked specifically to accelerator-driven systems can be found in Section 3.4 and in Section 3.8, *Nuclear and radiochemistry research*.

3.5.3 Conclusions and recommendations – fuel

Regarding fuel development and testing the following issues are seen as important:

- Extension of the lifetime of existing key facilities:
  - specifically the Halden Reactor which is 50 years old; there is a need to extend its availability.
- Looking to the future, the development of the Jules Horowitz Reactor (JHR) Co-ordination Action (EC, 2008a) is of particular note.
- Maintenance and extension of the IFPE database (NEA, 2008oo), see Section 4.5.3;
- Development of new facilities for Gen. IV conditions – high temperature, high fluxes, different coolant types, etc. (It is noted that some specific loops are under development, testing or construction.) See also the discussion in the sections on transmutation (3.4) and materials (3.6).

Moving to the topic of TRISO coated fuel particles (CFPs) for high-temperature gas-cooled reactors (HTGRs), it is worth noting that the testing of these fuels is also mentioned in the SFEAR report (NEA, 2007d) where it is recommended that international collaboration should be strongly considered due to the importance of fuel performance to HTGR safety, the long lead time and the cost of fuel testing. In addition, it will be important to maintain existing test reactors, such as CABRI, NSRR and ATR, due to their ability to also test HTGR fuels.

In relation to hot cells and post-irradiation examination:

- Long-term availability of hot cells for fuel examination must be guaranteed.
- The conclusions of the HOTLAB project (EC, 2005) should be considered:
  - "A survey of the near term hot cell work reveals that no substantial change is expected at least for the next five years. Fuel issues will keep a considerable part because the fleet of commercial reactors will change their technical basis only slowly: There will be work related to further fuel economy and safety, reprocessing, and storage and transportation. Since old nuclear plants will be decommissioned, lifetime assessments of various kinds of structural components will be further analysed. The shutdown of old plants will permit a check of lifetime prediction techniques by analysis of irradiated real components."
  - The report goes on to list a number of topics for which hot cells are currently used.
  - The report then considers the longer term, but concludes that it is not possible to draw conclusions in regard to long-term hot cell utilisation and points to the Gen. IV programme as a guide to potential utilisation. It concludes that the introduction of new systems will be shared among many countries and therefore, long-term assessments always require continuous observation of global developments.

Finally, in relation to fuel cycle chemistry:

- A considerable chemical engineering effort will be needed in a phase of scaling-up the proposed partitioning processes to pilot scale and subsequently to industrial prototype.
- Considering: *i)* the strict regulations limiting the quantities of MA which can be handled in shielded facilities; *ii)* the construction costs, possessing R&D facilities which fulfil the MA handling requirements is becoming a determining factor among P&T oriented countries.
- Irradiation facilities for studying the resistance of ligands towards radiolysis are essential for the development of new organic reagents used for an aqueous partitioning process (*e.g.* MARCEL at CEA Marcoule). An actinide laboratory capable of handling significant amounts of actinides is also needed to study the impact of alpha radiolysis.
- Organic synthesis laboratories, analytical chemistry laboratories and structural chemistry laboratories are also important to develop partitioning processes – see Section 3.8, *Nuclear and radiochemistry research*.

3.6 Materials

Experience has indicated that materials issues have been at the heart of many problems in past and currently-operating reactors. Present pressures to extend operating life, and even to raise power levels, mean that assessment and control of the potential effects of materials ageing, through processes such as cracking, embrittlement, fatigue and corrosion, have to be undertaken.

On the other hand, proposed developments such as new reactor designs or other systems such as ADS are presenting further new materials challenges.

[NB *Consideration of nuclear fuel materials is found in Section 3.5*.]

The original state of a material's composition, its process of manufacture, the defects arising from manufacture and use, and the various forms of load to which it is exposed both during normal and transient operation combine to affect the safety margins available in the design and how these margins vary over time. Equally, the sensitivity of the examination and testing methods available affect our ability accurately to assess the actual state of the materials in question.

This section will review some particular materials issues relevant to current and future nuclear power systems and then reviews the types of facility required for their study. That is not to say that there are no other topics of interest; for example in the VHTR there is "need for ceramics and composites for control rod cladding and other specific reactor internals, as well as for intermediate heat exchangers and gas turbine components for very-high temperature conditions" (GIF, 2007). However, the general principles illustrated in the sections below show that there is a need to build databases of information on the behaviour of the materials across the range of normal and transient conditions expected to be experienced. A useful review is given in papers at the Workshop on Structural Materials for Innovative Nuclear Systems (SMINS), Karlsruhe, Germany, June 2007 (Roos, 2008) and other current research interests are evidenced at the Spring Meeting, May 2008, of the European Materials Research Society (E-MRS, 2008).

3.6.1 Cladding and structural materials[17]

As in Section 3.2 the discussion of materials can be divided into near-term and longer-term perspectives.

In the near term, the focus is on existing or evolutionary reactor designs, meaning a concentration on LWR, pressure-tube reactors (CANDU and RBMK) and the remaining graphite-moderated reactors in the UK. Evolutionary designs exist for the LWRs and CANDU, while the others are unlikely to see further development other than life extension. There are only a few fast reactors currently operating; see Section 3.2.4 for a discussion on likely developments. Also potentially within the near term is the pebble bed design which has evolved from earlier HTRs. Note, however, that the projected service lives for new builds are proposed to be markedly longer than for existing plants (*e.g.* 60 years rather than 40 years), so that effects of corrosion or other chemical interaction will be significantly beyond those associated with current experience.

In the longer term the whole range of design concepts that exist within programmes such as GNEP and beyond that to Gen. IV come into consideration with a much broader spectrum of materials issues such as molten salts, very high temperature capabilities for VHTRs and performance under very high pressures such as associated with supercritical water-cooled reactors. As the range of conditions is generally beyond those currently experienced, new characterisation tests performed under predicted operational or transient conditions are required in order to provide the required knowledge base of information covering the properties under irradiation and/or corrosion conditions.

3.6.1.1 Irradiation embrittlement and corrosion in water reactors

Working outwards from the fuel, the first material of interest is the cladding. Within a period looking beyond the immediate next few years, the designs are almost all water-cooled whether in LWR or pressure-tube. The cladding material is zirconium alloy based (*e.g.* zircaloy) and the issues are essentially those of pellet-clad interaction and corrosion issues such as oxide protective layers and the effect of hydride-induced embrittlement which also affect zirconium-based pressure tubes in the case of those designs. Note that fuel and its clad will usually be removed and replaced at regular intervals, but that the integrity of pressure tubes has to be preserved throughout life.

Within an LWR reactor and out to the pressure vessel (RPV) the issues of corrosion and embrittlement also feature significantly in the work to ensure that longer life and higher irradiation conditions required by current considerations can be met. A major concern has been the possible failure of the RPV by catastrophic brittle fracture. The sequence of events leading to the hardening and subsequent embrittlement of RPV steel under neutron irradiation has been understood in broad terms since the 1950s. The spectrum of neutrons irradiating the RPV steel, together with γ rays, leads to damage in the steel through production of vacancies and interstitials. Hardening of the steel arises because clusters, in particular, provide barriers to movement of dislocations through the lattice and this hardening, in turn, affects brittle fracture behaviour. This issue also applies to the ageing of pressure tubes in pressure tube reactors.

17. The assistance of J. Knott, University of Birmingham, UK in preparing this section based on text for a paper to be presented at the 8th Conference on Structural Integrity Assessment (*www.fesi.org.uk/fesi-esia.html*) is gratefully acknowledged.

In recent years, it has been found that neutron irradiation-induced hardening in submerged arc welds may be attributed to: *i)* obstacles resulting from clusters of point defects; *ii)* copper precipitation hardening. While initially surprising because the RPV temperature was lower than that expected to be consistent with precipitate hardening in steels, it emerged that the point defects generated by irradiation increase diffusion rates such that copper can become mobile at these lower temperatures.

Trace impurity elements have been recognised as having two effects on the occurrence of intergranular fracture during stress relief/tempering: *i)* during the heating cycle ("stress-relief cracking" or "reheat cracking"); *ii)* during the final slow cool to room temperature. Effects of impurities on brittle fracture have been particularly marked for VVER steel.

Similar cracking problems were experienced with "under-clad cracking". The clad is an approximately 5-mm layer of austenitic stainless steel, deposited over the inner surfaces of the RPV, with the aim of minimising the production of corrosion products. Under-clad cracking arises from differences in the thermal expansion coefficients for ferritic and austenitic steels resulting in thermal mismatch stresses. A second feature is that inter-diffusion can occur across the dissimilar metal interface during cooling after deposition of the clad. Chromium from the austenite, in particular, can interact with the carbon in the ferrite.

The avoidance of both stress relief cracks and underclad cracks by controlling trace impurity elements to low levels is recognised. Additionally, low levels of impurities are beneficial with respect to resistance to environmentally-assisted cracking (EAC). However, such specifications affect cost, and the need for tight specifications will be subjected to close scrutiny in future builds. While current designs such as EPR and AP1000 continue to have the RPV clad, improving water conditions in PWRs may mean that it is possible to contemplate un-clad RPVs, and thus the question of whether any potential future PWR would be clad or unclad remains open.

An additional issue is the evaluation of the potential effect of γ ray irradiation enhancement on the corrosion processes, to determine the magnitude of any radiolytic component (particularly over the projected extended service lives).

Beyond the RPV, while in a lower radiation environment, the experience over the decades in relation to stress corrosion cracking (SCC) in both primary circuit and steam generator tubing has also required attention to the behaviour of alloys in the particular chemical regimes existing in the primary and secondary circuits and indicates that the well being of the whole plant over a lifetime also depends on the consideration of those environments beyond the core of the reactor. Care needs to be exercised particularly for dissimilar metal joints, such as the "safe-end" weld between the ferritic pressure vessel nozzle and austenitic piping in the PWR primary circuit.

Use of supercritical water-cooled reactors (SCWRs) and hence existence of higher reactor outlet temperatures than in current designs would require new alloys for fuel cladding and in-core components. Key properties of candidate materials requiring data and understanding are:

- oxidation, corrosion and stress corrosion cracking (SCC);
- strength, embrittlement and creep resistance;
- dimensional and micro-structural stability.

Research programmes include out-of-pile tests on un-irradiated and irradiated alloys, together with in-pile tests in an SCW loop (or loops) to be constructed. The ability to predict and control water chemistry will be essential to minimise corrosion rates, SCC and to minimise deposition both in-core and out-of-core. Enhanced understanding of the chemistry of supercritical water is required relating to the marked change in the density and chemical properties of SCW through the critical point. Radiolysis effects may well be different to those experienced in current LWRs (GIF, 2007).

3.6.1.2 Radiation damage and corrosion in liquid metal reactors

A high priority R&D need for heavy liquid metal (HLM) systems is the study of changes in the properties of materials in a radiation environment, and the combined effects of radiation and corrosion. Protective oxides as barriers against liquid metal corrosion may be compromised by radiation-enhanced transport of ions in oxides. Modelling tools are insufficient to analyse the existing data and hence extrapolate the results to very long times. Only a basic understanding exists of the triggers and kinetics of corrosion processes with long incubation periods, such as breakaway oxidation at high temperatures.

For the materials of current interest, basic data on compatibility and mechanical property changes in the liquid metals are already available, mostly under out-of-pile conditions. However, for specific design requirements the following needs are of highest priority:

- long-term corrosion behaviour of steels and their coatings;
- corrosion tests in lead-bismuth eutectic (LBE) with low oxygen concentration;
- corrosion erosion and friction mechanism;
- development of a mass transfer model using reliable solubility and diffusivity data and also corrosion data for non-isothermal systems;
- mechanical behaviour of structural materials and the corrosion protection barriers in a representative temperature – stress field and more specifically:
  - creep;
  - fatigue and creep-fatigue;
  - fracture mechanics;
  - creep and fatigue crack growth.

In addition a very high priority is placed on the assessment of mechanical properties in the liquid metal and under irradiation of the steels and coatings.

These properties need to be measured in the relevant ranges of temperature, neutron fluence, stresses and HLM flow velocity for the different components. Further details will be found in the LBE Handbook (NEA, 2007).

After the completion of this materials testing programme, the performance of manufactured materials in well-defined shapes, *e.g.* tubing as fuel cladding, will need to be assessed.

HLM chemical properties

Solubility and diffusivity data of oxygen, some metallic elements (Fe, Cr, spallation products such as Po, etc.) and some oxides (*e.g.* iron oxides, chromium oxides, etc.) are of importance for:

- assessment of corrosion rates;
- design and engineering of purification systems;
- source term assessment.

The LBE Handbook (NEA, 2007) notes that effort is needed to produce reliable solubility and diffusivity data.

It should be noted that there is still a lack of reliable data on the chemical behaviour of impurities in liquid metals, their evaporation from liquid metal surfaces under different atmospheres, their adhesion to metallic surfaces, etc. Such data are, however, crucial for a realistic simulation of the radionuclide distribution during routine operation and accident scenarios. Experimental studies on such topics have been performed at PSI in the frame of the MEGAPIE (PSI, 2008) and TARGISOL (2008) projects and are still ongoing in the frame of the EURISOL (2009) project. [See also the references by Neuhausen, *et al*. (2004, 2005, 2006, 2006a).]

HLM thermal/physical properties

The LBE handbook (NEA, 2007) contains most relevant thermal and physical properties of Pb, Bi and LBE with recommended correlations. For the high temperature range, basic properties such as the liquid density, vapour pressure and liquid adiabatic compressibility, have had to be estimated up to the critical point using semi-empirical models based on the extrapolation of low temperature data. The production of experimental data in the high temperature range has been recommended, so that the computed values can be validated.

Liquid metal test facilities

In the development of lead-cooled fast reactor (LFR), the corrosion characteristics and corrosion behaviour of the reactor coolant together with the structural and cladding materials is of particular significance. CRIEPI (2008) has been investigating corrosion behaviour of stagnant LBE at 650°C on high chromium martensitic stainless steel, a potential material for LFRs (GIF, 2007).

The following discussion will largely concentrate on LBE studies, but it should be noted that research on sodium continues in relation to liquid metal-cooled reactors. For example, Toshiba has recently opened a new facility (WNN, 2008a).

The NEA has recently published the LBE Handbook (NEA, 2007) and the following discussion is largely based on information in that document.

HLMs have been studied since the early development of fission energy as reactor core coolants for fast reactors and more recently for fusion energy blanket applications. More recently, they are being considered for ADS systems and as target materials for high power neutron spallation sources. As an example, the MEGAPIE neutron spallation target (PSI, 2008) has been designed and constructed in the frame of ADS development, but it will also be used as a neutron source for materials investigation with thermal neutrons. (Discussion on liquid metals which is specific to ADS can be found in Section 3.4.)

LBE was chosen as the coolant for a number of Alpha class submarine reactors in the former Soviet Union, for which there was an R&D emphasis on the chemistry control of the liquid metal to avoid plugging due to slag formation and to enhance corrosion resistance of the steels specifically developed for such service. The lead-cooled BREST (NIKIET, 2008) reactor design and the LBE-cooled SVBR concept (Stepanov, 1998) moved the concept into the civilian fast reactor field with the Gen. IV LFRs now becoming of international interest (see Section 3.2.4.4) and new missions including hydrogen production, nuclear waste transmutation, and small modular reactors with long-life cores for supplying electricity and heat in remote areas and/or developing economies.

The use of liquid metals as heat transfer media has implications in relation to neutronics, thermal-hydraulics, safety and economics as well as materials, but it is only the last aspect that is considered in this section. Particular materials issues for LBE are:

- acceptable corrosion and mechanical degradation of structural and containment materials, and lifetime of equipment;
- high stability of the liquid metal (*e.g.* limited chemical reactions with secondary coolants and air or formation of spallation products).

[NB *A comparative assessment of thermo-physical and thermo-hydraulics characteristics of lead, LBE and sodium can be found in IAEA TECDOC 1289 (IAEA, 2002)*.]

A compilation of the existing OECD experimental facilities with their main parameters and key objectives is given in Chapter 12 of the LBE Handbook (NEA, 2007) with descriptions of the HLM facilities available at the laboratories of the expert groups participating in the production of the handbook.

The handbook notes that these "facilities cover almost all basic studies needed to design HLM nuclear systems working at temperatures up to 550°C." However, it notes that further needs can be envisaged for applications at temperatures above 600°C and for specific analysis concerning safety aspects in representative conditions, specific component testing in prototypical conditions and In-service Inspection and Repair (ISI&R).

Section 14 of the Handbook (NEA, 2007) provides "Perspectives and R&D Priorities for Heavy Liquid Metal Coolant Technologies". It concludes that there are a number of technological gaps to be filled before a prototype reactor or ADS system can be designed, constructed and operated using lead or LBE cooling. A number of scientific issues remain open, including fundamental physical, chemical and transport properties of HLM and associated materials and coolant chemistry. However, no apparent "show-stoppers" exist with, perhaps, the exception of uncertainties in materials performance over very long service life in cores at temperature >500-550°C.

In Europe, HLM technology is being developed mainly for transmutation where EUROTRANS (2008) aims to demonstrate its technical feasibility; see Section 3.4.4. However, the European ELSY project (Cinotti, 2006) is a study of an ~600 MW(e) LFR, see Section 3.2.2, which will take advantage of the results emerging from the DEMETRA domain of EUROTRANS. Some specific Pb technology developments have been foreseen, *e.g.* high temperature materials characterisation in Pb and steam generator tube rupture studies. [NB *The generation of highly radioactive, and hence heat generating, polonium as an activation product in the coolant is much lower in pure Pb than in LBE. Omitting bismuth from the coolant, therefore, reduces the problems associated with decay heat removal*.]

In the United States, within the Gen. IV LFR programme, the current focus is on reactors of small or medium size with >20 years core life and no on-site refuelling. The peak cladding temperatures are limited to 650°C and core outlet temperature of ~560°C.

In Japan, J-PARC has a transmutation element to design, build and test a LBE-cooled ADS transmutation system with an LBE spallation target; system conditions are similar to those in EUROTRANS. Other programmes are developing concepts using HLM in advanced reactors or in intermediate heat exchangers, with conditions similar (or even less demanding) to the USA LFR.

In South Korea ADS and the PEACER reactor programmes are developing HLM in collaboration with international partners.

It is also worth mentioning at this point the recent establishment of the Virtual European Lead Laboratory project and its website (VELLA, 2008). This site explains that:

> *"VELLA has the intent to homogenise the European research area in the field of lead technologies for nuclear applications in order to produce a common platform of work which continues also after the end of the initiative. VELLA plans both to create a network of all the principal laboratories and to strongly connect the different groups of experts, to have a common definition of the good operational practices and to promote the exchange of the scientific results by means of appropriate and innovative tools and procedures. It also has the significant objectives to promote the access to the main existing facilities in the EU to different specialist groups, support the technological development and the qualification activities and create a homogenous European 'scientific community', organised to support all the required technological challenges and the necessary research requirements."*

3.6.1.3 Materials aspects in high-temperature reactors

Using the GNEP and Gen. IV programmes as a guide to the range of research, the following can be said about the metallic materials aspects of future designs.

i) Very high-temperature reactor (VHTR). There is a requirement for new materials in order to accommodate an outlet temperature goal of 1 000°C, and to have a capacity for safe operation beyond normal conditions. In relation to metals, high-temperature materials will be required for internals, piping, valves, heat exchangers and for gas turbine sub-components (GIF, 2007).

ii) Sodium-cooled fast reactor (SFR). The experience gained on earlier and continuing fast reactors indicates particular needs in the region of high fast neutron dose. For example, there is a need to withstand fluences of up to 200 dpa in order to obtain ~20% burn-up (Maloy, 2007) at irradiation temperatures of 400-550°C using low activation ferritic/martensitic (F/M) steels. In the longer term aiming at more than 20% burn-up and higher temperatures (> 550°C) indicates a need for advanced materials such as oxide dispersion strengthened (ODS) F/M steels. The effects of small elemental additions (such as Si and Ta) on irradiation and corrosion resistance need to be understood. A major materials problem concerned with the SFR occurs in the steam generator, where liquid sodium at ~500°C flows through (over 30 km of) ~10 mm diameter tubing of 1-2 mm wall thickness to raise steam from a counter flow of liquid water.

iii) Gas-cooled fast reactor (GFR). As with the SFR, development is required of materials with superior resistance to fast neutrons under very high-temperature conditions. "The GCFR will take advantage of synergies with VHTR, where the development of a high performance helium turbine and coupling technologies for process heat applications is being undertaken" (GCFR, 2007). Links exist with the FP6 projects RAPHAEL (2008) and ExtreMat (Bayern, 2007) for materials development.

Future applications may evolve that require higher temperatures, but are not within current planning horizons. Extensive R&D on materials and coolant technology would be needed if that were to occur and materials such as oxide dispersion strengthened (ODS) steels and/or advanced ferritic/

martensitic steels would be of interest for temperatures ~650-700°C where Pb, rather than LBE, would probably be used in HLM nuclear systems. At >750-800°C refractory metals and alloys, ceramics and composites are potential candidates. Very different coolant technology, and design, construction and operating methodology will be required. Irradiation stability, fatigue strength, fabrication, etc., are challenging and Pb "will likely be the only choice since LBE and the associated technologies no longer offer any intrinsic or experiential advantages" (NEA, 2007).

3.6.2 Moderator materials[18]

Graphite has been used as a moderator since the birth of nuclear power in the 1940s. It forms the core of the present operating Magnox, AGR and RBMK reactors as well as fuel elements and reflectors in experimental/prototype HTR designs such as PBMR (South Africa), HTR-10 (China) and HTTR (Japan) as well as future Gen. IV VHTRs. Graphite use as a fixed moderator in most graphite reactors implies that it is subject to high neutron flux through the reactor life and therefore subject to a considerable amount of irradiation damage. This leads to significant graphite dimensional and property changes.

The prediction of graphite components' life and behaviour depends on empirical relationships obtained from small samples irradiated in MTRs.

Life extension of existing plants and development of new types of graphite for new HTR/VHTRs has led to the need for the development of a more sound fundamental understanding of the relationship between irradiation damage to the graphite microstructure and bulk property changes. This understanding is required to interpolate and extrapolate the current database and models to the end of the life for existing plants and to the more onerous conditions required in the new designs.

In the UK Magnox and AGR dimensional and property changes due to fast neutron irradiation is further complicated by the simultaneous radiolytic oxidation of the graphite in the carbon dioxide coolant. The development of cracks in some UK moderator components earlier in life than was expected has lead to extensive industry-led research programmes including a MTR programme at HFR Petten. In addition a Graphite Technical Advisory Committee (GTAC) has been created in the UK to advise the regulator on such issues (UK HSE, 2007).

Typically, graphite research needs to develop an understanding of the changes in the following properties during irradiation in-core under typical operating and transient conditions: dimension, coefficient of thermal expansion, Young's modulus, strength, thermal conductivity, density and irradiation creep; see for example (Marsden, 2006). In relation to graphite, there is the additional question of the variability arising from different manufacturing routes. With a view to the potential effects during later decommissioning the location of impurities and isotopic content are important as well as how these will behave under conditions that lead to leaching or mobility so that the means to remove or treat them can be understood. Some of these tests are not easy to undertake, for example irradiation creep tests where the sample has to be loaded while under irradiation. A review of graphite selection for Next Generation Nuclear Plant (NGNP) is given in (Burchell, 2007) and other current issues relating to graphite were identified in the recent "PIRT" process (US NRC, 2007). The VHTR programme within the Gen. IV project (GIF, 2007) notes that the work package on graphite has made significant progress. See also the review by van der Laan, *et al.* (2008).

The use of graphite-moderated reactors since the beginning of the applications of nuclear energy in the early 1940s means that there are significant amounts of graphite (~250 000 tonnes world wide) that will require handling as waste from reactors such as the Magnox in the UK and France, AGR in the UK and RBMK in Russia and various plutonium production reactors. While waste is not a topic for this report, it does mean that some of the research work required on graphite materials does have a backward-looking aspect in relation to decommissioning as well as being forward-looking in relation to new designs such as PBMR, HTR and VHTR (IAEA, 2006). This issue is being dealt with under a four-year European Union FP7 programme named CARBOWASTE (Banford, 2008).

3.6.3 Facilities required for material science and materials testing

Three types of facilities are required for: *i)* materials irradiation; *ii)* modelling validation/materials characterisation; *iii)* materials testing.

18. The assistance of B. Marsden, University of Manchester, UK, in preparing this section is gratefully acknowledged.

3.6.3.1 Materials irradiation

Along with fuels, most nuclear structural materials will be subject to irradiation conditions ranging from moderate to severe, often coupled with high in-service temperatures. Thus *irradiation facilities* where the operating scenarios can be reproduced under controlled conditions are a prerequisite, for example:

- reactors (neutrons: thermal, fast spectra);
- spallation sources;
- particle accelerators (protons, alphas, ions);
- dedicated facilities (for example for irradiation by 14 MeV neutrons appropriate to fusion).

Examples include:

- Reactor:
  - mixed spectrum: HFIR (ORNL, 2008), ATR (INL, 2007), HFR at Petten (NRG, 2008), BR2 at Mol (SCK•CEN, 2008), OSIRIS at CEA Saclay (CEA, 2008d), and the Jules Horowitz Reactor (CEA, 2008c; EC, 2008a);
  - fast spectrum: JOYO in Japan (JAEA, 2008e), BOR-60 in Russia (RIAR, 2008).
- Spallation sources: MEGAPIE (PSI, 2008), STIP programme (at PSI-SINQ source) (Dai, 2003) and the new Materials Test Station (MTS) (LANL, 2008; Cappiello, 2006) at LANSCE (2008a). [NB *MTS "will be the only facility in the United States that will be able to produce the necessary irradiation capabilities until a new fast test reactor can be built" (Cappiello, 2006)*.]
- Ion implanters (accelerators): IRMA (CNRS Orsay, part of the JANNUS facility (Serruys, 2007), under construction).
- Dedicated facilities with particular MeV fusion neutron interests that will also have fission reactor applications: Projected IFMIF facility (ENEA, 2008).

3.6.3.2 Modelling validation and materials characterisation

Models and simulations can be validated by comparison with carefully planned experiments using techniques with operational windows in the appropriate region of the conjugate energy and momentum transfer space. At the shorter time and length scales, *neutron and synchrotron radiation scattering techniques* are two almost unique experimental approaches which require large facilities and investment in ancillary sample environments (for example, to handle active samples) but whose results correlate directly to the simulated static and dynamic properties. These approaches have proved fruitful in many other branches of materials science.

Researchers combine results from these two complementary techniques with others obtained by techniques such as positron annihilation spectroscopy, electron microscopy or atom probe tomography so as to gain insight on the relevant mesoscopic phenomena. Results from transmission measurements, radiography and tomography (realised with epithermal, thermal or cold neutrons, with X-rays and gammas or protons) are also relevant for modelling and simulation in the mesoscopic and macroscopic ranges and are also available at large facilities and to some extent in more conventional laboratories.

Strain scanning techniques based on neutron and X-ray diffraction allow the determination of stress in components in a non-destructive way and have been used to validate finite element calculations. In fact, there now exists a close coupling between diffraction techniques and micro- and nano-mechanics.

Examples of neutron scattering and synchrotron radiation facilities include:

- Reactor-based: ILL (international, located in France) (ILL, 2008), FRM-II (Germany) (TUM, 2008), OPAL (Australia) (ANSTO, 2008).
- Spallation sources: ISIS (UK) (ISIS, 2008), SNS (ORNL, 2008b), LANSCE (Los Alamos) (LANSCE, 2008a), European Spallation Source (planned, EU) (ENP, 2003a).

- Synchrotron radiation sources: European Synchrotron Radiation Facility (international, located in France) (ESRF, 2008), Advanced Photon Source (APS) (Argonne National Laboratory) (ANL, 2008), SOLEIL (France) (SOLEIL, 2007), Diamond (UK) (DLS, 2008).

Most facilities listed under this heading will have dedicated instruments (beam lines) particularly suited for nuclear materials studies relevant to the validation of the multi-scale modelling (Maloy, 2007) and simulation approach, and to materials development in general. Typically these will be:

- neutron spectrometers to allow measurements of density of states and phonon dispersion relations to compare with results of *ab initio* calculations;
- diffractometers for phase identification and characterisation (including grain size);
- X-ray absorption spectroscopy (XAS) beam lines for the investigation of short range order, co-ordination spheres, effects in ODS steels and other nano-structured materials;
- magnetic circular dicroism beam lines and polarised neutron beam lines (alloy behaviour related to magnetic effects);
- small angle neutron diffractometers (pore sizes, voids, inclusions);
- white beam tomography and radiography and energy resolved transmission techniques for meso- and macroscopic three-dimensional imaging;
- strain scanning (thermal neutrons, high energy X-rays) for comparison with finite element stress calculations and crystallographic elastic constant determination.

3.6.3.3 Materials testing

Facilities will continue to be required for mechanical testing (especially determination of elastic properties and fatigue behaviour) and chemical compatibility studies under combined temperature, irradiation and chemical environment conditions close to the in-service ones. Examples are loops to study corrosion by liquid metals (Na, Pb-Bi, Hg), supercritical water or carburisation/decarburisation.

Examples of mechanical testing and corrosion loops include:

- liquid metals – see the numerous examples in the LBE Handbook (NEA, 2007);
- corrosion in supercritical water (loops at JRC Petten and CIEMAT);
- loops for (impure) He/material compatibility for (V)HTR (CEA Grenoble, ENEA Brasimone, EdF, Areva);
- large-scale creep testing facilities (CEA "Airbus" facility);
- carburisation/decarburisation loops (JRC Petten);
- corrosion + irradiation facilities (ICE facility at LANL).

Recent interest in multi-scale modelling and simulation of nuclear fuels and structural materials has prompted the constitution of a new NEA Nuclear Science Committee activity: "WPMM: Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems" (NEA, 2008E). At the time of writing, the definition of the precise portfolio of activities within the new working party remains to be agreed upon by the member countries. However, the background and scope of WPMM may serve to give some indication on the research and test facilities which simulation tools, existing and under development, may require for their experimental validation.

Certainly, these facilities are not specifically intended for this purpose and extensive use of them will be made for the broader purpose of materials development, characterisation and qualification in the nuclear energy field. It should be noted that some of the facilities will also be used (extensively) for experimentation in other non-nuclear energy-related disciplines.

3.6.4 Conclusions and recommendations – materials

Facilities will continue to be required to cover the range of requirements for: *i)* materials irradiation; *ii)* modelling validation/materials characterisation; *iii)* materials testing.

In particular, the continuing availability of materials test reactors (MTRs) and the facilities that such reactors are able to provide is an essential feature of the study of materials of interest to reactors and other branches of nuclear science. The scope of the irradiation capabilities will need to increase as the demands from work on new reactor types evolves.

Equally, the availability of large facilities such as spallation sources and reactors for analysis of materials is deemed essential. Spallation sources can also be valuable sources of neutrons for materials irradiation to complement the facilities available at MTRs.

3.7 Safety

Safety is, of course, an overriding issue in all nuclear-related research. It should be noted that a NEA CSNI activity led to the recent publication of the SFEAR report (NEA, 2007d) which focused directly on this issue of safety. The following sections have therefore been largely derived in association with that report and with the co-operation of the CSNI/SFEAR group. Note, however, that the SFEAR remit was expressly limited and it does not, as a result, cover the whole range of topics in this present report. To be explicit, the focus of the SFEAR report was on "the safety issues, research needs and supporting research facilities associated with currently operating water-cooled reactors in NEA member countries. These reactors include pressurised water reactors (PWRs), boiling water reactors (BWRs), pressurised heavy water reactors (PHWRs) and Russian designed VVERs." In addition specific attention was given to HTGR reactors. Fast reactors, however, were not considered by the SFEAR group.

Also recorded below are some of the SFEAR report recommendations. The SFEAR group recognised that implementation of their recommendations would be dependent "upon interest and commitment of the 'host countries' to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries." Further information on the SFEAR report is provided in Section 4.1.

Another topic considered in the SFEAR report was nuclear fuel; due to the overlap with the discussion in Section 3.5, the information has not been repeated here.

In contrast, information on systems of relevance to nuclear safety but not unique to the nuclear industry such as plant control and monitoring, seismic effects and fire assessment have not been amplified here and further details should be sought in the SFEAR report (NEA, 2007d).

3.7.1 Thermal-hydraulics

Thermal-hydraulics (T/H) became one of the main nuclear safety disciplines when postulated accidents such as LOCAs and other thermal-hydraulic transients were identified as the dominant safety concern for LWRs. Because full-scale experimentation was not feasible in most situations, significant computational developments had to be undertaken in order to be able to simulate such transients properly, the results of which were required for the safety cases of these reactors. Numerous national and international experimental programmes provided the data necessary for understanding the phenomena and simulating them. [NB *Some thermal-hydraulics facilities are of importance for studies of basic concepts, not just for safety-related studies*.]

The CSNI has always given careful consideration to the issue of thermal-hydraulic code validation as well as the experimental database needed for such validation. The results from these programmes provide a sound basis for model validation of traditional, widely used code systems such as ATHLET (Germany), CATHARE (France), RELAP (USA), etc. However such data are often insufficient for validating code systems coupling 3-D neutronics and thermal-hydraulics. For such purposes complementary data from nuclear plant measurements and experiments are often used. Several exercises are being carried out jointly by the Nuclear Science Committee and CSNI relative to BWRs, PWRs and VVERs.

In recent years, the CSNI has taken initiatives to support safety relevant thermal-hydraulic facilities that were in danger of closure. This was done through the establishment of international projects addressing issues of broad international interest and centred on the technical capabilities of selected facilities. These projects are still ongoing and include SETH (NEA, 2008jj), PKL (NEA, 2008ee; AREVA, 2004), PSB-VVER (NEA, 2008ff) and ROSA (NEA, 2008hh).

The SETH project started with an experimental programme in the PKL (AREVA, 2004) and PANDA (PSI, n.d.) facilities, which were recommended for international consideration (NEA, 2001). The PKL experiments addressed the issue of potential boron dilution accidents in PWR reactors. They are being continued under a new PKL project. The PANDA experiments are to provide data on containment three-dimensional gas flow and distribution that are important for code prediction capability improvements, accident management and design of mitigating measures. In relation to VVER reactors, the international PSB-VVER project was started with the objective of providing the unique experimental data needed for the validation of thermal-hydraulic codes used for the safety assessment of VVER-1000 reactors. Following a JAERI[19] proposal, the CSNI also recommended making all necessary steps to establish an international experimental project to be conducted in the Japanese ROSA facility.

Over time the scenarios of primary concern shifted from the large break LOCA to small breaks and other incidents (*e.g.* boron dilution) and, as a result, the thermal-hydraulic research effort shifted accordingly to cover the more complex phenomena associated with these categories of accidents. Improved computational tools were also developed in order to handle these properly. Although reactivity-related accidents and transients were, of course, considered from the beginning of the deployment of LWRs, increasing emphasis has been placed upon the accidents having a neutronic origin or strong neutronic aspect. Accordingly, it has been realised that multi-dimensional, coupled thermal-hydraulic/neutronic computations are needed to reduce the conservatism of the earlier, simpler analyses and/or to properly simulate some complex situations. Although many of the existing facilities are not instrumented sufficiently enough that they can be used to validate finely detailed analysis tools (*e.g.* CFD codes), they are included in this section for completeness.

The new concerns that regulators faced after the Chernobyl accident, more generally related to the understanding and simulation of situations and phenomena in reactors designed in the former Eastern European countries, provided additional requirements for research and development. The emergence of advanced LWRs having passive safety systems opened another new area of relatively new phenomena and situations that had to be addressed. Likewise, power upgrading of existing reactors may also require refinement and further validation of existing analytical tools and more extensive experimental databases. The thermal-hydraulic safety issues that could benefit from additional research are listed in the SFEAR report (NEA, 2007d).

In spite of these continuing developments, often conducted internationally, a number of thermal-hydraulic safety issues still require attention. New issues will certainly also arise in relation to the design and safety analysis of future reactor systems. For example, as current plants continue to make operational changes (*e.g.* up-rating of power) analysis will be needed to assess changes in safety margins and plant response to off-normal conditions. Furthermore, the increasing use of risk-informed regulation (IAEA, 2005) will require better tools and data.

LWRs: CSNI carried out a review of current LWR system behaviour conducted through large integral test programmes (NEA, 1996). This publication includes a systematic selection of openly available data for code validation. Thermal-hydraulic data are routinely needed to assess analytical tools, particularly when new issues or designs are to be analysed. The current emphasis on risk-informed regulation in some countries, additional interest in more detailed analysis of certain types of accidents and new plant designs provide incentive for maintaining experimental capability to validate analytical tools.

VVERs: The CSNI also carried out a review of the level of validation of thermal-hydraulic codes applied to the analysis of VVER reactors (NEA, 1993). The aim was to supplement the review done on integral and separate-effect test facilities, including special features of VVER reactor systems with respect to large and small break LOCAs.

PHWRs: For PHWR reactors, experimental programmes have been carried out in specialised facilities (*e.g.* large-scale header facility, cold water injection facility) using full-size components such as headers, fuel channels and end fittings. The RD-14M facility (AECL, 2004; IAEA, 2007c; Richards, 2002) has been used for a comprehensive programme on emergency core cooling system effectiveness, natural circulation and shutdown cooling using a full height, multi-channel test simulation.

For facilities associated with thermal-hydraulics the SFEAR activity (NEA, 2007d) has provided an extensive list and all of the ones referenced by name in the SFEAR report are incorporated into RTFDB. The table below can equally be consulted.

19. Now JAEA.

Canada	RD-14M and MTF
Czech Republic	SKODA-VVS
Finland	PACTEL
Germany	PKL
Japan	LSTF LST/ROSA and THYNC
Korea	ATLAS and MIDAS
Russia	PSB-VVER
Switzerland	PANDA
USA	APEX and PUMA

However, as has been noted earlier, the SFEAR report (NEA, 2007d) only considered facilities that met certain criteria:

> *"The scope of this activity is limited to safety issues and facilities associated with nuclear reactor design, construction and operation (spent fuel storage is not included in the scope of this activity) in OECD member countries and Russia…"*
>
> *"…the group focused on those facilities that have unique capabilities, would be very expensive to replace and have high relevance to the resolution of current safety issues*."

Therefore the list of facilities in that report should not be regarded as being comprehensive, though, of course, it has attempted to give details on the larger, more significant ones.

In contrast, the current review has attempted to collect information for the RTFDB from a wider range of facilities and, as a result, other sources of information have been reviewed.

For example, the European Network for the Consolidation of the Integral System Experimental Data Bases for Reactor Thermal-hydraulic Safety Analysis (CERTA, 2008) co-ordinated by the EC JRC at Ispra has information on a number of facilities. Some are included in the SFEAR report but it also lists some others (mostly shut down).

- Operational:
  - PACTEL (Finland);
  - PKL (Germany);
  - PMK-2 (Hungary);
  - PANDA (Switzerland).
- Shut down:
  - BETHSY (France);
  - UPTF (Germany);
  - LOBI (Italy/JRC Ispra);
  - PIPER-ONE (Italy);
  - SPES (Italy);
  - FIX-II (Sweden).

The PKL facility continues to be supported within the OECD PKL-2 project (NEA, 2008ee) which is just starting. Within this project safety issues relevant to both current PWR plants and new PWR design concepts will be investigated focusing on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations. Tests on heat transfer mechanisms in the steam generators in the presence of nitrogen will be complemented by tests in the PMK test facility for horizontal steam generators.

In the framework of the OECD SETH-2 project (NEA, 2008jj), which started in 2007, relevant post-accident containment phenomena are being investigated in the PANDA and MISTRA facilities with the aim of improving the modelling and validation of computational fluid dynamics and lumped parameter computer codes.

As an indication of current evolution in facilities, it can be noted that the present PACTEL loop in Finland is VVER-440-oriented. However, in the recent NEA/CSNI/GAMA meeting a proposal to modify PACTEL to a western PWR-type facility was presented [see (NEA, 2008ww) for information on the CSNI Working Group on Accident Management and Analysis].

The Nuclear Power Engineering Corporation (NUPEC), Japan, made a series of void measurements using full-size mock-up tests for both BWRs and PWRs. Based on state-of-the-art computer tomography (CT) technology, the void distribution is visualised at a mesh size smaller than the sub-channel under actual plant conditions. In this test facility NUPEC also performed steady-state and transient critical power test series' based on the equivalent full-size mock-ups. Considering the reliability of not only the measured data, but also other relevant parameters such as the system pressure, inlet sub-cooling and rod surface temperature, these test series' supplied the first substantial database for the development of truly mechanistic and consistent models for void distribution and boiling transition. This database was released to the OECD/NEA for the participants in a benchmark study aimed at advancement in the field of two-phase flow theory with particular relevance to the evaluation of safety margins for nuclear reactors.

The AVR pebble-bed HTGR at Jülich, Germany operated from 1967 to 1988; it provided an important experimental basis for modelling neutronics aspects and improving fluid dynamics modelling for pebble-bed modular reactors and HTRs. This experimental facility is being dismantled and, in view of renewed interest in HTGRs in the frame of the Gen. IV initiative, similar facilities will be needed for the future deployment of such reactors. In particular thermo-fluid loops for assessing the behaviour of such systems should be maintained and facilities for studying coupled physics and thermo fluid-dynamics need to be available. Since 1998, the EC has supported the RAPHAEL Integrated Project (RAPHAEL, 2008) addressing the viability and performance issues of an innovative system for the next generation of power plants, the very high-temperature reactor (VHTR), which can supply both electricity and heat for industrial applications.

The following conclusions were derived from the SFEAR report (NEA, 2007d):

In the short term:

"In the thermal-hydraulics area, four facilities are in short-term danger. Two of these facilities support PWR thermal-hydraulic work (PKL and APEX). However, there are other facilities for PWR-T/H work not in short-term danger (e.g. ROSA). Thus no recommendation for short-term action is needed for PWR T/H facilities. For BWR T/H facilities, both existing large integral BWR thermal-hydraulic test facilities (PANDA and PUMA) are in danger of being closed in the next 1-2 years. These facilities are unique and expensive and at least one should be maintained to be available for supporting research related to current or future BWR safety issues. Accordingly, preservation of one integral BWR thermal-hydraulic test facility (either PANDA or PUMA) is considered essential for preserving a BWR thermal-hydraulic research infrastructure. SESAR is of the view that PANDA is the preferred facility for preservation due to its scale, replacement cost and versatility (i.e. it is useful in the severe accident as well as thermal-hydraulic area). Accordingly, CSNI action is recommended in the short term to support a co-operative research programme in PANDA. It should be noted that CSNI actions resulting from the SESAR/FAP report played a major role in the preservation of PANDA over the past 5 years."

In the longer term:

"It should be noted that the SESAR/FAP report (NEA, 2001) recommended that in the long term thermal-hydraulic facilities for each major reactor type should be maintained in North America, Europe and Asia. However, given the current situation with respect to safety research programme funding, the SFEAR group is of the opinion that this recommendation is no longer practical and recommends that the long-term strategy for facility preservation focus on ensuring at least one thermal-hydraulic facility for each reactor type be maintained world wide."

The SFEAR report went on to identify the following in relation to thermal-hydraulics:

- *Technical expertise needed*: Modelling and analysis.
- *Facility capability needs*: Large-scale integral test facility for each reactor type.
- *Important factors for facilities*: Scale, temperature and pressure capability are key factors. Also, the completeness of the facility with respect to factors such as: auxiliary systems, number of loops and instrumentation capability is important.

3.7.2 Severe accidents

Severe accidents (SA) are generally considered to be events beyond the traditional design basis of currently operating nuclear power plants. The prevention or mitigation of SA is the largest contributor to reducing risk to the public from the operation of nuclear power plants. SA scenarios involve an initiating transient, such as a LOCA, accompanied by the postulated failures of multiple safety systems, thus compromising the capability of shutting down the reactor or maintaining adequate cooling of the fuel, resulting in significant damage to the fuel (core melting), possibly leading to the release of significant amounts of radioactivity from the primary system into the containment. Under certain circumstances, the containment may also be postulated to fail or to be bypassed (*e.g.* through steam generator tube failure in a PWR), resulting in a major radioactive release to the environment. Although generally not considered during initial licensing, SA have been assessed through specific plant reviews, generic analysis and the development of accident management programmes.

For many years important national and international programmes have been undertaken in the field of severe accidents and their results have been shared through international "networks". CSNI has played a major role in organising and administering co-operative research programmes in this area of severe accidents. These programmes include RASPLAV (conducted in Russia to assess the thermal load on the RPV lower head under core melt conditions) (NEA, 2008gg), Sandia Lower Head Failure (conducted in the United States to assess the mechanical behaviour of the RPV lower head under pressurised severe accident conditions) (NEA, 2008ii), MCCI (conducted in the United States to assess ex-vessel molten core debris coolability and interaction with containment concrete) (NEA, 2008dd), MASCA (conducted in Russia to measure the physical properties of molten core material) (NEA, 2008cc; RRC KI, n.d.) and SERENA (an analytical programme assessing the state of knowledge related to fuel-coolant interactions) (NEA, 2008kk). In addition, CSNI sponsored an effort aimed at assessing accident management strategies and identify areas of consensus. This was called the Senior Group of Experts on Severe Accident Management (SESAM) (NEA, 2007). These programmes have contributed to knowledge about severe accident phenomena, the resolution of questions related to severe accidents and the potential for accident management measures to successfully terminate or mitigate accident progression. They have also served to avoid premature shutdown for certain key facilities. However, important issues remain and need to be studied to support the continued safe operation of nuclear power plants via severe accident management and/or reducing the potential for SA scenarios, as well as supporting the licensing of new LWR and PHWR designs.

The severe accident issues and phenomena that could benefit from additional research are related to reducing the remaining uncertainties in accident progression and mitigation and to understanding the safety implications caused by changes in plant design or operating characteristics (*e.g.* high burn-up fuel, MOX fuel).

The issues can be grouped according to the phases of progression of a severe accident and the phenomena present in each of those phases, as follows:

- in-vessel phenomena [core heat-up, clad/fuel melting and relocation, combustible gas generation, fuel-coolant interaction (FCI)];
- ex-vessel phenomena [vessel failure, core-concrete interaction, direct containment heating (DCH), FCI, combustible gas generation];
- source term [quantity, chemical form, transport and timing of fission product release from the fuel, reactor coolant system (RCS) and containment];
- containment integrity (capability of the containment to withstand during a severe accident the conditions caused by combustible gas burning, decay heat, molten core attack);
- accident management (actions that can be taken to terminate or mitigate the consequences of a severe accident).

The prevention of severe (core damage) accidents and how to manage them if they do occur remains an important objective for the continued safe operation of LWR and PHWR nuclear power plants. Although in-vessel melt progression is fairly well understood, there remain significant uncertainties in predicting whether or not molten core material will remain in-vessel, the consequences of molten core material getting out of the reactor vessel (*e.g.* coolability, combustible gas) and source term generation. These uncertainties also encompass the prediction of the best accident management strategies for preserving RPV and containment integrity and reduction of the amount of radioactive material available for release to the atmosphere.

Resolution of severe accident issues through prevention or mitigation is the goal of the remaining research. This can be accomplished by design changes, analysis showing the issue is of low safety significance or developing strategies to terminate or mitigate severe accidents prior to their resulting in the release of large quantities of radioactive material to the environment. To reduce uncertainties, current research should be conducted at sufficient scale to investigate the important phenomena and use real materials, whenever possible.

PHWR have similar severe accident issues as LWR; however, the core melt progression in a pressure tube reactor presents additional challenges associated with propagation of pressure tube failure, fuel-coolant or fuel-moderator interaction and the potential to over pressurise the calandria and cause calandria and additional pressure tube rupture.

The following short- and long-term conclusions have been derived from the SFEAR report (NEA, 2007d):

"In the severe accident area, most facilities supporting the resolution of the following safety issues for BWRs, PWRs, VVERs and ALWR are in danger in the short term:

- *pre-core melt conditions;*
- *combustible gas control;*
- *coolability of over-heated cores.*

Based upon a review of the facilities in short-term danger[20]*, the group concluded that the following facilities should be preserved due to their importance to resolution of the above issues (as illustrated by their high relative ranking), replacement cost, versatility, and value in long-term infrastructure preservation:*

- *PHEBUS (Belpomo, 2005);*
- *QUENCH (FZK, 2008a);*
- *MISTRA (CEA, 2005)."*

There is further, detailed discussion on each of these recommendations in the SFEAR report (NEA, 2007d) and the details should be sought there. However, attention is drawn to the fact that the SFEAR report notes that "it should be recognised that implementation of the above recommendations are dependent upon interest and commitment of the 'host countries' to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries."

[NB *In the interim since the drafting of the SFEAR report, confirmation has been received of the planned shutdown of PHEBUS (Belpomo, 2005). Despite the general recognition of the importance of the facility, especially outside France, operating cost considerations and sparseness of proposals/customers for new programmes prevailed in the shutdown decision.*]

The SFEAR report noted the following critical facilities to be monitored in the long term:

- *Integral testing*: PHEBUS (Belpomo, 2005).
- *In-vessel phenomena*: QUENCH (FZK, 2008a), VERDON (CEA, 2008h), KROTOS (CEA, 2006a, 2008f), Fuel Channel Safety Facility (Ref. [3] in NEA, 2007d).
- *Ex-vessel phenomena*: MCCI (NEA, 2008dd), VULCANO (CEA, 2006a), ThAI (NEA, 2008ll), KROTOS (CEA, 2006a, 2008f).
- *Containment mixing/combustion*: PANDA (PSI, n.d.), LSCF (Krause, 2007), ThAI (NEA, 2008ll), MISTRA (CEA, 2005).

20. Nine such facilities are listed in Table 4-1 of the SFEAR report (NEA, 2007d).

The SFEAR report also identified the following in relation to severe accidents:

- *Technical expertise needed*: phenomena, progression modelling and analysis.
- *Facility capability needs*: reactor or ex-reactor testing of FP release and transport, core debris cooling, combustible gas control and AM strategies.
- *Important factors for facilities*: use of prototypic materials and large scale are important.

Independent of the SFEAR activity, the EU SARnet project (SARnet, 2005) has recently proposed some research priorities for the EURSAFE Research Issues (ERI) in this area (Schwinges, 2007; Trambauer, 2007):

High priority: 6 issues:

- core coolability during reflood and debris cooling, corium coolability in lower head;
- ex-vessel melt pool configuration during MCCI, ex-vessel corium coolability by top flooding;
- melt relocation into water, ex-vessel FCI;
- hydrogen mixing and combustion in containment;
- oxidising impact (Ru oxidising conditions/air ingress for HBU and MOX fuel elements) on source term;
- iodine chemistry in RCS and in containment.

Medium priority: 3 issues:

[NB *"These items should be investigated further as planned in the different research programmes."*]

- hydrogen generation during reflood and melt relocation in vessel;
- integrity of RPV due to external vessel cooling;
- direct containment heating.

For four issues, the current knowledge is considered sufficient, assessing the state and progress of knowledge and the risk and safety relevance and taking into account ongoing activities outside the frame of SARnet. These issues are therefore assessed as low priority and they could be closed after the related activities are finished:

- corium coolability in external core catcher;
- corium release following vessel rupture;
- aerosol behaviour impact on source term (SGT and containment cracks);
- core reflooding impact on source term.

Three further items are marked as "issue could be closed". Due to the current risk significance and the state of knowledge no further experimental programme is needed:

- integrity of reactor coolant system and heat distribution;
- ex-vessel core catcher and corium-ceramics interaction, cooling with water bottom injection;
- FCI including steam explosion in weakened vessel.

Furthermore there are other experimental programmes not included in the SFEAR report that are worth noting. For example the CODEX tests (Hózer, 2003, 2006), produced significant information on VVER in-vessel phenomena and on air ingress issues.

3.7.3 Reactor control

The control of nuclear power plants has evolved considerably over the last fifty years. In particular, control room designs and instrumentation and control (I&C) systems used in NPPs have seen great advances. The licence renewal programmes for existing Gen. II and Gen. III reactors include, among other things, modernisation of control rooms and a step-wise introduction of digital I&C systems. New

Gen. III NPPs to be constructed and the planned Gen. III+ NPPs differ significantly from earlier designs in a number of ways, not least in that the control rooms have computerised, seated, workstations, and digital I&C is used extensively. The Gen. IV reactors, which are even more advanced, will be likely to employ operation concepts that are considerably different from those used today, and the designs will clearly embrace advanced digital I&C technologies that are expected to continue to evolve rapidly.

The introduction of digital systems raises concerns related to design, licensing, implementation and dependability. Important questions are how to:

i) show that the digital systems are safe enough;

ii) document that a computerised control room is as at least as safe as the analogue room it replaces;

iii) demonstrate that human performance will be acceptable in a future, highly automated, operational environment.

Facilities such as the Halden Project's HAMMLAB (IFE, 2005a) are required to respond to questions related to the quality and reliability of human performance in control rooms, as well as issues related to the development and introduction of digital technologies and applications, for both the short- and long-term perspective.

3.7.4 Conclusions and recommendations – safety

As noted at the beginning of Section 3.7, this part of the report has largely been derived in collaboration with the effort that led to the SFEAR report (NEA, 2007d). Thus the conclusions and recommendations that follow are essentially those in the SFEAR report and further details should be sought in that reference. However, the following is a summary and pertains to both the short and long term:

- CSNI efforts at facility preservation should focus on large facilities, whose loss would mean the loss of unique capability as well as the loss of substantial investment that, in the current climate of tight resources, would not likely be replaced. Preservation also includes maintaining expertise, knowledge, capabilities and personnel essential to infrastructure conservation. (Previous CSNI efforts have kept several large facilities active over the past five years, thus helping the current SFEAR effort. However, many large, expensive and unique facilities are projected to close over the next one to five years.)
- Both CSNI and CNRA should take steps to encourage industry co-operation by emphasising: *i)* the responsibility of industry to develop sufficient data to support their applications; *ii)* the benefits of co-operative research; *iii)* the value of preserving critical research infrastructure.
- Because of the large numbers of hot cells and autoclaves, each country is recommended to monitor the status of these essential facilities and bring to CSNI's attention any concerns regarding loss of critical infrastructure.
- Certain safety issues have no large-scale facilities identified for the conduct of relevant research. The appropriate CSNI Working Groups should evaluate whether or not such facilities are needed to support resolution of these issues.

Short term

The following recommendations are directed toward those actions that CSNI could take in the short term to prevent the loss of key facilities in imminent danger of closure:

- In the thermal-hydraulics area, PANDA and PUMA are in danger of being closed in the next one to two years. These facilities are unique and expensive and at least one should be maintained available. Further arguments and a preference are given in the SFEAR report (NEA, 2007d).
- In the severe accident area, most facilities supporting the resolution of the following safety issues for BWR, PWR, VVER and ALWR are in danger in the short term:
  - pre-core melt conditions;
  - combustible gas control;
  - coolability of over-heated cores.

- The SFEAR report recommends that three specific facilities should be preserved due to their replacement cost, high relative ranking and versatility.
- In the other technical areas (fuels, and integrity of equipment and structures) no short-term CSNI actions are recommended.
- The SFEAR report recognises that implementation of the above recommendations depend upon the interest and commitment of the "host countries" to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries.

Long term

- In the longer term, it is recommended that CSNI adopt a strategy for the preservation of a research facility infrastructure, based upon preserving unique, versatile and hard-to-replace facilities. (Consistent with their remit, this recommendation is based upon supporting currently-operating LWR and PHWR and the licensing of future ALWR and APHWR.) The strategy should include consideration of short- and long-term priorities, cost of preservation (and would this detract substantially from other programmes/facilities) and contingency plans in case of facility loss.
- The factors used in the SFEAR report to arrive at conclusions and recommendations could be useful in developing a long-term strategy for assessing and initiating future co-operative research projects. These include:
  - facility operating and replacement cost;
  - the ability to define a useful experimental programme;
  - long-term resource implication and priorities;
  - industry participation;
  - host country long-term plans and commitment.
- A table of critical research facility infrastructure needs is given in the SFEAR report; those considered unique, hard to replace and having high relative importance in their technical area are identified. CSNI are recommended to monitor the status of these facilities in the longer term with a goal of taking action, as appropriate, to ensure that critical facilities are available for each reactor type to meet the critical research infrastructure needs. In addition, for new reactors and technologies, CSNI should take an active role in encouraging and organising co-operative research efforts, thus contributing to infrastructure preservation. As with the short-term recommendations above, host country interest will be an important factor in determining which facilities to preserve.

3.8 Nuclear and radiochemistry research

Nuclear and radiochemistry laboratories are multi-purpose research facilities and are used for studies in various fields, examples being fuel cycle chemistry in nuclear science as well as migration of long-lived radionuclides from a waste repository. While the latter is one of most important issues of nuclear and radiochemistry research it is out of the scope of the present Expert Group study.

Facilities described in the present section are dedicated to basic actinide chemistry that supports fuel research and reprocessing of the spent fuels as shown in Section 3.5 on fuel as well as Section 3.4 on ADS and transmutation systems. The objective of the research is to study chemical characteristics and specific behaviour of actinides under various system conditions, such as aqueous solution, highly concentrated salt solution, molten salt and liquid metal. X-ray beam lines at synchrotron radiation facilities are also essential tools for analysis and speciation of actinide elements and are described later in this section.

3.8.1 Actinide chemistry laboratories

The actinide elements are the 15 elements with atomic numbers 89 through 103, the first member of which is actinium and the last member is lawrencium (Morss, 2006). These are usually referred to as the "actinides" though IUPAC prefers the term "actinoids" (Holden, 2004). All actinide isotopes are radioactive and are characterised by an increasing number of 5f electrons. Extensive investigations have been performed on actinide compounds of nuclear interest. Actinium and the elements americium through lawrencium are similar in many chemical respects to the lanthanide elements that fill the 4f electrons.

A firmer understanding of actinides themselves is needed in order to characterise actinide compounds. The description of the electrons partially occupying the 5f orbitals in these systems is necessary to reliably predict the properties of nuclear fuel and waste forms in the solid state or separation systems in the liquid state.

These basic studies of actinide science have mainly been carried out at universities and national institutes. However, most of the radiochemistry laboratories in universities are for education and do not have their own facilities capable of handling the significant amounts of actinides, except uranium and thorium, required to measure the physical and chemical properties. Plutonium, especially, is an element difficult to deal with in university laboratories due to licence restrictions. Alternative experiments techniques use trace amounts of target actinides or lanthanides but the results obtained do need to be verified by using technologically appropriate amounts of actinides.

It is hard and expensive for each university or institute to construct and maintain these facilities for actinide chemistry taking necessary safety and security measures into account. In this context, some countries have been organising networks of facilities dedicated to actinide chemistry research in order to share their facilities and encourage young researchers in universities.

ACTINET in Europe

ACTINET (2008) is a consortium dedicated to advanced actinide sciences which gathers more than 30 European research institutions. Special facilities such as glove boxes and hot cells are necessary for such advanced research. ACTINET pools selected parts of the major facilities of some large European institutes [CEA, ITU, INE, SCK•CEN, Forschungszentrum Rossendorf (FZR), and Paul Scherrer Institut (PSI)]. Further information on ACTINET is given in Section 3.5.2.3 on partitioning processes under development and in (ACTINET, 2008, 2008a). The Minor Actinide Laboratory (MA-Lab) at ITU Karlsruhe (ITU, 2008a) is a key facility which possesses integrated hot cells that are capable of dealing with irradiated fuel as well as measuring basic chemical properties.

Actinide research facilities in Japan

Excepting the private sector, only a few institutes and universities provide hot cells and glove boxes capable of handling the significant amounts of Np, Pu, Am and Cm required to measure their physical and chemical properties. The following are major facilities in Japan, though they have not yet been as efficiently organised as ACTINET.

The JAEA set up a pyrochemistry cell called "TRU-HITEC" (JAEA, 2008m) for studies on the high temperature chemistry of actinide compounds, *e.g.* measurement of an oxidation-reduction reaction. TRU-HITEC has steel and polyethylene walls for shielding gamma and neutron radiation and allows measurements under conditions of a very high purity, inert atmosphere with grams of americium and milligrams of curium.

The International Research Centre for Nuclear Materials Science of Tohoku University has a hot laboratory near JMTR (JAEA, 2008f) and the experimental fast reactor, JOYO (JAEA, 2008e). It has also undertaken research on new types of nuclear fuel using hydrides as well as measurements of the properties of actinide compounds, *e.g.* heavy fermion systems, itinerancy, localised character and superconductivity.

Kyoto University Research Reactor Institute (KURRI, 2003) has a hot laboratory for its research reactor (KURRI, 2006). Irradiated materials (up to 185 TBq) can be handled in three hot cells. The facility has also been used for measurements of chemical and electrochemical properties of actinides in aqueous and molten salt media.

Actinide chemistry laboratories in USA

Laboratories owned by the DOE possess facilities for fuel research and are listed in the RTFDB database under the "Radioactive Material Handling Facilities" facility type, *e.g.* ANL Hot Labs and the Fuel Conditioning Facility (FCF) at INL. Analytical and radiochemistry laboratories attached to these facilities are capable of measuring chemical properties of actinide compounds.

The Chemistry and Metallurgical Research Facility (CMR) building at LANL (n.d.) was completed in 1952 as one of the world's first research and experimental facilities for analytical actinide chemistry, metallurgy and materials. This facility today houses research and experimental activities for analytical chemistry, plutonium and uranium chemistry and metallurgy, and support functions.

Radiochemistry laboratories at universities (*e.g.* Florida State and Washington State) have an important role in education. The Glenn T. Seaborg Institute (GTSI) (LLNL, 2008) serves as a national centre for the education and training of the next generation of scientists in the fields of nuclear chemistry, chemical engineering, materials science, environmental chemistry and chemical biology. It has its own on-site Seaborg institute at LLNL, LBNL and LANL; LLNL concentrates on nuclear and bio-nuclear science, LBNL focuses on the impact of radionuclides in the environment, and LANL stresses nuclear science studies of plutonium and heavier elements.

3.8.2 Analysis and speciation facilities

Facilities for fuel treatment require routine analysis and thus usually have an adjacent analytical laboratory installed, *e.g.* CBP with CBA at ATALANTE (CEA, 2008), which is equipped with many instrumental and chemical analysis tools. In this section, a large facility dedicated to the advanced characterisation of actinides, generically designated a synchrotron radiation facility (SRF), is adopted as a typical useful probe.

The recent availability of synchrotron radiation has revolutionised actinide chemistry. Most of the synchrotron studies published to date have focused on X-ray absorption spectroscopy (XAS), which is often divided into two experiments: Extended X-ray Absorption Fine Structure (EXAFS) and X-ray Absorption Near Edge Structure (XANES). EXAFS and XANES are widely used to determine the co-ordination environment and oxidation states, respectively, of an actinide in a non-crystalline sample. Powerful light sources and a micro-focusing technique enable these measurements with milligrams of an actinide element. The main research themes using the beam lines at SRFs are as follows:

- *General actinide chemistry*: structure, electronic states, co-ordination-bonding, database.
- *Separation science and technology*: metal-ligand complex, molecular design of novel extractants.
- *Nuclear waste forms and remediation*: speciation and stability in amorphous solid phase.
- *Radionuclides in the environment*: speciation, interaction with micro-organisms.

In the field of fuel cycle chemistry (partitioning), molecular scale information (*i.e.* atomic conformation, inter-atomic distances, co-ordination numbers, oxidation state, electronic and bonding state) about actinide (An) and/or lanthanide (Ln) ions and their complexes present in an aqueous solution and an organic solvent is essential for understanding and predicting their actual behaviour. Inner or outer sphere complexation mechanisms in various solvent phases or the structural character of metal polyhedra have been clarified by measurement of X-ray absorption spectra of target atoms. EXAFS is a very valuable tool for understanding the co-ordination modes of An when involved in aqueous or non-aqueous solutions. Knowledge thus obtained has been used, for example, for the design of novel extractants in the development of sophisticated separation process between An(III) and Ln(III).

The synchrotron radiation facilities in OECD member countries that are capable of measuring radioactive materials are listed as follows:

USA

- *Advanced Light Source (Berkeley-ALS)*: 1.5-1.9 GeV, molecular environmental science beam line (LBNL, 2008).
- *Advanced Photon Source (ANL-APS)*: 7.0 GeV, actinide facility for APS (ANL, 2008).
- *Stanford Synchrotron Radiation Laboratory (SSRL)*: 3.0-3.5 GeV, molecular environmental beam line (SSRL-MES) (SLAC, 2008).

Europe

- *European Synchrotron Radiation Facility*: 6 GeV, Rossendorf Beam Line (ESRF-ROBL:BM20), radioactivity – 185 MBq (total) (ESRF, 2008).
- *SOLEIL (Source optimisée de lumière d'énergie intermédiaire de Lure)*: 2.15 GeV, MARS (MAtiére Radioactive á SOLEIL) beam line: radioactivity – 18.5 GBq per sample (SOLEIL, 2007.)
- *FZK-INE-ANKA (Ångstromquelle Karlsruhe)*: Synchrotron Environmental Laboratory 2.5 GeV, INE beam line: actinides samples (FZK, 2008).
- *Swiss Light Source (SLS)*: micro XAS beam line (SLS, 2002).

Japan

- *Photon Factory*: 2.5-3.0 GeV, BL-27A, B beam lines – sealed radioactive samples; Th, U, Tc, Np, Am, Cm (KEK, 2008).
- *SPring-8*: 8 GeV, 22XU beam line – sealed radioactive samples; Th, U, Tc, Np, Am, Cm (JASRI, 2008).

SSRL at Stanford, especially, is an essential facility which has been used for measuring plutonium samples. The newly constructed MARS beam line at SOLEIL is the fourth beam line in Europe for studying radionuclides, after the ROBL, INE and the Micro XAS beam lines. The MARS beam line was designed to deal with highly radioactive samples with an activity of up to 18.5 GBq, and its user access is targeted for 2008 (Sitaud, 2006).

The NEA Nuclear Science Committee has sponsored a series of Euroconferences and NEA Workshops on Speciation Techniques and Facilities for Radioactive Materials at Synchrotron Light Sources, "Actinide-XAS". The proceedings of the fourth meeting, held in 2006, are available (NEA, 2007c), and the fifth workshop was held in July 2008 (NEA, 2009).

The disposal of high-level radioactive wastes is not an objective of the present Expert Group study but it is worth mentioning that X-ray absorption spectroscopy provides unique insights into the redox reactions of actinides in the waste form matrix or at mineral-water interfaces. Detailed information on the chemical forms of radioactive Sr, U, Np, and Pu in the waste form showed that different species of the same element are present in individual waste.

3.8.3 Conclusions and recommendations – nuclear and radiochemistry research

It is recommended that integrated hot cell laboratories [like the Minor Actinide Laboratory (MA-Lab) at ITU Karlsruhe] be retained to measure basic physical and chemical properties of actinide compounds.

Hot cells and glove boxes owned by universities are important tools for education. A network (like the pooled facilities in ACTINET) is an important approach for the effective sharing of facilities and to promote international collaboration.

Synchrotron radiation facilities capable of measuring radioactive samples should be retained, *e.g.* SSRL, for measurement of plutonium samples. In addition there are future requirements to measure properties of actinides and LLFP in spent fuel directly by means of X-ray absorption spectroscopy. Special beam lines like MARS at SOLEIL are needed for measuring highly radioactive samples.

3.9 Miscellaneous facilities

This section identifies a couple of topics that are relevant to the current study but which, perhaps, do not merit a full section of their own.

3.9.1 Nuclear process heat for hydrogen production

Intimately linked with the reactor concepts associated with Gen. IV, the production of hydrogen from nuclear process heat bears consideration as it has its own research lines not otherwise noted above. The NEA Nuclear Science Committee has recently organised a 3rd Information Exchange Meeting on the Nuclear Production of Hydrogen at JAEA in Japan (NEA, 2006) to follow on from earlier sessions it also organised. A 4th Information Exchange Meeting on the Nuclear Production of Hydrogen is to be held in Chicago in April 2009 (NEA, 2008q). (The Expert Group has noted that much of the current worldwide research activity is oriented on the *non*-nuclear part. The following is a brief review of the third meeting with an emphasis on those aspects that relate to the need for facilities of interest to the Nuclear Science Committee.)

At the meeting held in October 2003, the participants concluded that substantial experimental progress was being made in developing nuclear hydrogen production technology and stressed the need to pursue the use of currently available nuclear reactor types and methods for producing hydrogen (*e.g.* electrolysis using LWR-produced electricity), while continuing R&D work on promising new concepts that may become better suited to centralised production.

The 3rd Information Exchange Meeting discussed the current scientific and technical issues related to the nuclear production of hydrogen divided into five technical sessions:

i) *The prospects for hydrogen in future energy structures and nuclear power's role*. This session illustrated the role that nuclear power can play as a non-carbon-emitting power source and so help to meet growing world energy demand and offset demands on fossil fuels and the effects of man-made greenhouse gas emissions. The use of nuclear heat to assist steam methane reforming or to produce hydrogen through water cracking could provide carbon-free solutions to other sectors of the world economy. The viability of these technologies will depend on the technical success of the ongoing R&D, but also on their market economics. It was noted that the nuclear production of hydrogen through electrolysis or thermochemical processes may already be competitive with steam methane reforming, given rising natural gas prices. Other factors such as the availability of uranium and the development of reprocessing and waste storage options may affect the extent of nuclear power's future role.

ii) *The status of nuclear hydrogen research and development efforts around the globe*. It was noted that major nuclear hydrogen R&D programmes are under way in many parts of the world, including Japan, China, Korea, France, Canada and the United States. The largest research efforts are directed at the sulphur-iodine thermochemical process coupled to very high-temperature gas-cooled reactors. Significant work is also being done on high-temperature steam electrolysis and other thermochemical cycles. There are expectations that large-scale demonstration of nuclear hydrogen production will be achieved before 2020. Current test reactors in Japan (HTTR) and China (HTR-10) are capable of reaching temperatures to allow testing of some nuclear hydrogen production concepts. Canadian efforts are focused on lower-temperature options that would be compatible with the super-critical water reactor being developed, including water electrolysis. The EU has separate nuclear research programmes and hydrogen development programmes, but the potential of nuclear-generated hydrogen is recognised.

iii) *Integrated nuclear hydrogen production systems*. This session considered: *i)* nuclear reactor concepts tied to hydrogen production; *ii)* technology linking hydrogen production facilities to nuclear heat sources; *iii)* co-generation concepts and technology; *iv)* integration of nuclear hydrogen production with a growing hydrogen economy. Different reactor concepts are being developed to support hydrogen generation, *e.g.* JAEA is considering the Gas Turbine High-Temperature Reactor 300 (GTHTR300) with core outlet temperatures compatible with the sulphur-iodine hydrogen production cycle while General Atomics is collaborating on a modular helium reactor suited for the sulphur-iodine process and high-temperature electrolysis. International collaborators are looking at issues surrounding the coupling of the hydrogen-generation processes and the nuclear heat source. Well designed coupling of the heat sources to needs can optimise hydrogen production efficiency.

iv) Nuclear hydrogen technologies and design concepts. This session considered thermochemical cycles, electrolysis and nuclear-assisted steam methane reforming. Significant progress since 2003 was reported on the R&D of these nuclear hydrogen technologies, *e.g.* a bench-scale demonstration of the sulphur-iodine cycle was made at JAEA in 2004 and plans are in place for larger demonstrations over the next decade. These experiments are leading to better insights into an optimised design of the system and ways to overcome remaining technical issues.

Most countries participating in the meeting are studying high-temperature steam electrolysis, *e.g.* steam electrolysis hydrogen production at 100 litres/hour has recently been demonstrated in the United States.

Several alternatives to the sulphur-iodine process and steam electrolysis are being considered. Thermo-electrochemical cycles at various stages of development are being studied, including two hybrid sulphur-based cycles, plus the copper-chloride cycle, the magnesium-chloride cycle and the copper ferrite cycle. Screening tools have been developed to rapidly assess whether further research is warranted for less mature thermo-electrochemical cycles.

The Tokyo Institute of Technology is exploring a novel idea for on-board reforming of methane with carbon dioxide capture to provide hydrogen as a transportation fuel. The reformer cartridges would be regenerated in a 550°C reaction with hydrogen. The heat and hydrogen could be produced through the various nuclear technologies being discussed.

v) Basic and applied science in support of nuclear hydrogen production. The technical and commercial viability of the nuclear hydrogen production options being pursued is not assured. Fundamental advances in the materials and processes may be key to their commercial adoption. Advanced membranes and catalysts, for instance, could improve the efficiency of difficult chemical separations in the sulphur-iodine process and the hybrid sulphur processes. Other issues considered were: *i)* thermodynamic data; *ii)* membrane separation; *iii)* catalysis; *iv)* heat transfer technology; *v)* safety research; *vi)* oxygen disposition.

Recommendations to the NSC on possible further international collaboration in this field were also formulated during the meeting. A fourth meeting was proposed and is now scheduled for 2009 (NEA, 2008q) in view of the IAEA International Conference on "Non-electric Applications of Nuclear Power: Seawater Desalination, Hydrogen Production and other Nuclear Applications" in April 2007 (IAEA, 2003) with NEA co-operation.

The need for consideration of safety issues to ensure that the chemical and nuclear facilities pose no risk to each other was noted in the light of the probable requirement that a hydrogen production facility (or any other facility that requires process heat) should be located as close to the nuclear heat source as possible to minimise heat transfer losses.

Of interest to the current Expert Group's activities, the need for co-operation on matters such as those listed below implies an equal need for facilities to elucidate the corresponding information on:

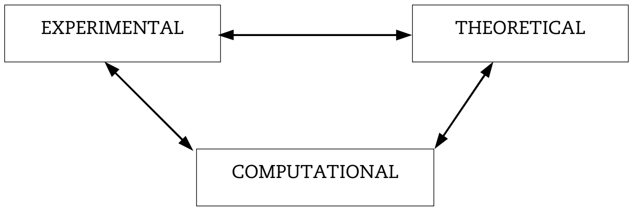
- safety;
- materials and chemical property measurement and verification;
- materials development, including structural materials, membranes and catalysts;
- advanced fabrication techniques.

3.9.2 Simulation and high-performance computing infrastructure

Until quite recently science had been progressing though the application of two distinct methodologies: experimental and theoretical.

$$\boxed{\text{EXPERIMENTAL}} \longleftrightarrow \boxed{\text{THEORETICAL}}$$

However, the development of digital computers has transformed the pursuit of science because it has given rise to a third methodology: the computational mode.



The intent of this additional mode is to numerically solve scientists' mathematical models in their full complexity. A simulation that accurately mimics a complex phenomenon contains a wealth of information about that phenomenon.

High performance computers have therefore become a significant part of the infrastructure for research and development relating to nuclear science application as well as in other spheres.

An extraordinary gain in processing speed has been achieved in computers for the last two decades through improvement of components' speed and architectural innovation. Further large gains can be achieved by investment in the development of novel algorithms, well adapted to the new multiprocessor computer platforms. In particular, new solvers of physics equations required for design, safety and operation of nuclear reactors need to be produced.

The benefit will be, besides shorter calculation times for the same problem, the possibility of using more refined models and the removal of approximations that are no longer justified on the grounds of reducing large numbers of computations.

System optimisation will be possible in much shorter times and with higher precision.

Areas of recent development in the nuclear science field are:

- coupling of neutronics and thermal-hydraulics;
- interpretation of in-reactor experiments;
- large masses of data produced can be more efficiently "mined" for a better use of the results, better determination of confidence bounds and for improving understanding through fast visualisation.

It is recommended that efforts for innovating computing methods in nuclear applications be strengthened and that financial means for achieving this goal be made available to researchers.

TOP500 supercomputer sites

A convenient measure of the evolution of the power of computers can be gained from the "TOP500 project". This was started in 1993 to provide a reliable basis for tracking and detecting trends in high performance computing. Twice a year, a list of the sites operating the 500 most powerful computer systems is assembled and released. The best performance on the Linpack benchmark is used as performance measure for ranking the computer systems. The list contains a variety of information including the system specifications and its major application areas (TOP500, 2007).

Chapter 4: Related NEA activities

This chapter aims to provide a brief introduction to other recent and ongoing work at the NEA which is relevant to the present Expert Group activity. It should not be taken to be an exhaustive compilation; rather, it is limited to those activities with at least some involvement or implications in relation to research and test facilities.

The NEA work areas included in this section are:

- nuclear safety and regulation;
- nuclear development;
- radiological protection;
- nuclear science;
- joint projects – selecting those relating to facilities.

A number of NEA work areas are not included: radioactive waste management was considered as being outside the remit of the Expert Group, while nuclear law, sustainable development and civil society are not likely to relate directly to facilities.

The work of the Data Bank is explicitly mentioned in Section 3.1 on nuclear data, and will not be discussed any further in this chapter.

4.1 Nuclear safety and regulation

Within the activity of the Nuclear Safety and Regulation Work Area (NEA, 2008y), the assurance and maintenance of high standards of safety in the use of nuclear energy within member countries is a particular goal. This focuses on the work of two committees: the Committee on the Safety of Nuclear Installations (CSNI) (NEA, 2008d) and the Committee on Nuclear Regulatory Activities (CNRA) (NEA, 2008c) plus a number of projects. Some of these projects have been mentioned or alluded to earlier, some of which are ongoing and some are complete; for example:

- The Cabri Water Loop Project (NEA, 2008aa);
- The Halden Reactor Project (NEA, 2008bb);
- The PKL-2 Project (NEA, 2008ee);
- The MASCA-2 Project (NEA, 2008cc)
- The RASPLAV Project (NEA, 2008gg);
- The SESAR Thermal-hydraulics (SETH) Project (NEA, 2008gg).

For a complete list see the Nuclear Safety and Regulation Work Area website (NEA, 2008y).

Relating to the current Expert Group's activity on facilities, there is relatively low involvement of the CNRA with research facilities; however, the Working Group on the Regulation of New Reactors (WGRNR) (NEA, 2008C) clearly overlaps with the discussion in Section 3.2 on reactor development.

On the other hand, the Committee on the Safety of Nuclear Installations (CSNI) has a number of topics that relate to facilities and these are discussed below in Section 4.1.1.

Equally, in meeting its aims of supporting the development of effective and efficient regulation and oversight of nuclear installations, and also helping to maintain and advance the scientific and technological knowledge base the Nuclear Safety and Regulation Work Area also maintains a number of databases:

- CCVM: CSNI Code Validation Matrix Integral Test Data.
- CCVM: CSNI Code Validation Matrix Separate Effects Test Data.
- STRESA: CSNI Code Validation Matrix online.

These are discussed further in Section 4.1.2.

4.1.1 Committee on the Safety of Nuclear Installations (CSNI)

Significant activity in relation to research facilities has been undertaken by the Committee on Safety of Nuclear Installations (CSNI). In particular a number of working groups exist. The following have linkages to the needs for research and test facilities (but, again, refer to the website for a full list):

- *Working Group on Integrity of Components and Structures (IAGE)* (NEA, 2008A). The main topics investigated include the integrity of metal components, the integrity and ageing of concrete structures, and the seismic behaviour of structures and components.
- *Working Group on Accident Management and Analysis (WGAMA)* [295]. For both current and advanced reactors this working group aims to advance the current understanding of the physical processes and address the safety issues associated with:
  - reactor coolant system thermal-hydraulics and related safety and auxiliary systems;
  - in-vessel behaviour of degraded cores and in-vessel protection;
  - containment behaviour and containment protection, phenomena-based validation matrix for ex-vessel (containment) models and codes (containment-CVM);
  - fission product release, transport, deposition and retention.
- *Working Group on Risk Assessment (WGRISK)* (NEA, 2008B). The main mission of the working group is to advance the understanding and utilisation of probabilistic safety assessment (PSA) in ensuring the continued safety of nuclear installations in member countries. WGRISK has been active in several areas, including:
  - human reliability;
  - software reliability;
  - low power and shutdown risk.
- *Working Group on Human and Organisational Factors (WGHOF)* (NEA, 2008zz). The main missions of this working group are to improve the current understanding, to advance the utilisation of methodologies for human and organisational factor assessment and to address emerging safety issues in order to maintain and improve the safety of nuclear installations in member countries. The group meets to:
  - exchange information and experience about safety-relevant human and organisational issues;
  - discuss in detail, compares and benchmarks programmes;
  - indicate where further research is needed;
  - collaborate with other working groups.
- *Working Group on Fuel Safety (WGFS)* (NEA, 2008yy). This working group has been set up to address cross-cutting issues related to fuel behaviour in accident conditions, including work on associated aspects of thermal-hydraulics, oxidation, chemistry, mechanical behaviour and reactor physics.

- *Working Group on Fuel Cycle Safety (WGFCS)* (NEA, 2008xx). Advancement of the understanding of relevant aspects of nuclear fuel cycle safety in member countries is the main mission of this working group. It:
  - meets to exchange information on relevant matters including licensing systems, safety philosophy and safety standards to improve mutual understanding;
  - maintains a database on incidents involving fuel cycle facilities (FINAS);
  - indicates where further research is needed;
  - prepares state-of-the-art fuel cycle safety reports;
  - collaborates with other groups as necessary.

The CSNI has over many years relied on a senior group of experts on nuclear safety research (SESAR) to assess the need for and strategy of maintaining key research facilities. This activity was first conducted by CSNI in the late 1990s and led to a number of actions by CSNI to establish co-operative research projects directed at developing information relevant to safety issues on operating LWRs and PHWRs, while at the same time preserving key facilities and programmes. A report on this activity was issued by NEA in 2000 titled: *Nuclear Safety Research in OECD Countries. Major Facilities and Programmes at Risk (SESAR/FAP Report)* (NEA, 2001). In response to the recommendations expressed in that report, the CSNI has undertaken initiatives, notably in the thermal-hydraulic, severe accident and fire safety areas, as summarised in Table 5. These initiatives mainly consisted of initiating and carrying out internationally-funded OECD co-operative projects on relevant safety issues, centred on the capabilities of key facilities identified in the SESAR/FAP report. Several such projects were initiated and are currently ongoing, constituting a means for effectively maintaining basic technical infrastructure through international co-operation. The CSNI projects normally involve 10-15 countries, all contributing to the cost of the experimental programme. A list of recent projects is given in Table 6.

Table 5: Impact of SESAR/FAP facility recommendations on SESAR/SFEAR

SESAR/FAP recommendation	Resulting CSNI action	Impact on SFEAR
Maintain the PANDA, PKL and SPES facilities in the thermal-hydraulic area (these facilities were in near-term danger of closure).	Initiated the SETH programme utilising the PANDA and PKL facilities (no host country support for SPES).	PANDA maintained through 2005. Was in near-term danger and addressed in the SFEAR study (new programme started after the publication of the SFEAR report). PKL active and not in near-term danger.
Monitor and maintain key thermal-hydraulic facilities in the long term. T/H facilities should be maintained in North America, Europe and Asia.	Facility status monitored. Initiated programme utilising the ROSA facility when it was in danger of being shut down.	ROSA is active and not in near-term danger. Other T/H facilities continue to be monitored (PACTEL, ATLAS).
Maintain the RASPLAV and MACE facilities in the severe accident area (these facilities were in near-term danger of closure	Initiated the MASCA programme as a follow-on to RASPLAV to maintain facilities. Initiated the MCCI programme utilising the MACE facility.	MASCA was active and not in near-term danger. MCCI is active and therefore the MACE facility is not in near-term danger.
Develop centre of excellence on fuel-coolant interaction (FCI) in consideration of potential loss of the FARO and KROTOS facilities.	Initiated the SERENA programme (group of experts to discuss status of FCI and future experimental needs). FARO shut down. KROTOS kept on standby.	SERENA programme has recommended an experimental programme be conducted in KROTOS and thus may impact preservation of KROTOS facility. CSNI expert group to review SERENA recommendation.
Develop centre of excellence (COE) on iodine chemistry and fission product behaviour.	Programme proposals have been evaluated by CSNI.	Behaviour of Iodine project started. ThAI project started.

Table 6: Ongoing and recently completed CSNI international research projects

Project	Scope	Facility involved	Host country
HALDEN	Fuel and materials I&C, human factors	Halden Reactor HAMLAB	Norway Norway
CABRI	Fuel in RIA transients	CABRI pulsed reactor (+NSRR tests)	France (Japan)
SCIP	Fuel integrity	Studsvik hot cells	Sweden
PRISME	Fire safety	DIVA facility, IRSN	France
MASCA*	Severe accident (in-vessel)	Kurchatov Institute	Russia
MCCI	Severe accident (ex-vessel)	Argonne National Lab.	USA
ROSA	System T/H	ROSA loop, JAEA	Japan
PKL-2	PWR boron dilution	PKL loop, AREVA	Germany
SETH	Containment (CFD)	PANDA, PSI MISTRA, CEA	Switzerland France
PSB-VVER	T-H for VVER 1000	PSB loop, EREC	Russia
ThAI	Containment T/H	ThAI	Germany
BIP	Behaviour of iodine	AECL Lab.	Canada
SERENA**	Steam explosion	KROTOS, CEA TROI, KAERI	France Korea

* MASCA followed the RASPLAV project and was completed in 2006.

** In preparation.

In addition, and as noted earlier and in particular in Section 3.7, the SFEAR report has been published (NEA, 2007d). This builds upon and updates the SESAR/FAP work, but also expands its scope to cover advanced LWR (ALWR), VVER and high temperature gas-cooled reactors (HTGR).

Given the focus of the current Expert Group activity reported here, it is of particular note that the SFEAR report comments that:

> *"Since publication of the SESAR/FAP report, research facilities have continued to be shut down world wide. In fact, of the facilities listed in the SESAR/FAP report in the areas of thermal-hydraulics, fuel, reactor physics, severe accidents and integrity of equipment and structures (i.e. those areas most unique to the nuclear power industry), approximately 35% have been shut down in the past five years. Accordingly, loss of critical research infrastructure (i.e. facilities, capabilities and expertise) remains a concern and is a major factor in conducting the current study. However, it should be recognised that the SESAR/FAP effort led to CSNI actions that preserved five key facilities during the 2000-2006 time period."*

The need to maintain databases of experimental data is also recognised as an important issue, but is not treated in the SFEAR report, as data preservation is being addressed separately by the NEA.

The focus of the SFEAR report is on the safety issues, research needs and supporting research facilities associated with currently operating water-cooled reactors in NEA member countries. These include PWR, BWR, PHWR and Russian-designed VVER. For these reactors, the main purpose of the SFEAR report is to:

i) summarise the currently identified safety issues, whose resolution depends upon additional research work;

ii) provide the current status of those research facilities unique to the nuclear industry that support resolution of the safety issues;

iii) where such facilities represent a substantial investment of resources and are in danger of premature closure, recommend actions CSNI could take in the short term to help maintain them;

iv) provide recommendations on long-term nuclear safety research facility infrastructure needs and preservation.

In addition, where research facilities do not exist, but may be useful to address currently identified safety issues, these areas are identified.

The SFEAR report also provides information on safety issues and research needs not unique to the nuclear industry and on safety issues and research needs associated with HTGR. This information is presented for completeness and for the use of designers, operators and researchers in planning and conducting future work.

The safety issues addressed in the SFEAR report are organised into the following technical areas:

- Those unique to the nuclear industry:
  - thermal-hydraulics;
  - fuel;
  - reactor physics;
  - severe accidents;
  - integrity of equipment and structures.
- Those not unique to the nuclear industry:
  - human and organisational factors;
  - plant control and monitoring;
  - seismic behaviour of structures;
  - fire assessment.
- Those unique to HTGR.

Following an introduction describing the scope, purpose and approach used in assessing the safety issues and facilities, the SFEAR report provides a short overview of each of the reactor types covered in the report. In complement, Chapter 5 of this report contains sections on each of the technical areas containing descriptions of the safety issues and research facilities associated with that technical area. This information is then used to calculate a numerical relative ranking for each facility, which is then used to derive conclusions and recommendations for CSNI consideration. The recommendations pertain only to those technical areas unique to the nuclear industry as listed above. Some of the recommended actions are directed toward measures that CSNI should take in the next 1-2 years to prevent the loss of key facilities in imminent danger of closure.

4.1.2 Nuclear safety and regulation – databases

As noted, the Nuclear Safety and Regulation Work Area also maintains a number of databases. Brief information is supplied on these in the following sections.

4.1.2.1 CCVM

The CSNI Code Validation Matrix (CCVM) contains two sections:

- *The Integral Test Data* (NEA, 2008e). Over the years the NEA Data Bank has collected a sizeable subset of separate effects test reactor transient and LOCA integral test data (ITD). These data with accompanying documentation are now available on DVDs.
- *The Separate Effects Test Data* (2008f). The construction of an internationally agreed upon separate effects test (SET) validation matrix for thermal-hydraulic system codes is an attempt to systematically collect the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties in the modelling of individual phenomena by the codes. These data, along with accompanying documentation, are now available on DVD.

In both cases the reports describing the experiments have been electronically scanned and transformed into PDF files. Restrictions apply to the distribution of the data; see the websites for more information (NEA, 20083, 2008f).

4.1.2.2 STRESA

This password-linked entrance provides access to the CSNI Code Validation Matrix online facilities, which are jointly operated with the European Commission JRC. The NEA STRESA database is a clone of one produced at JRC Ispra, but contains data provided from the CCVM, with some others included. STRESA has an interface permitting the graphical display of the data (STRESA, 2008).

4.2 NDC activities

The goal of the NEA in this area 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 examining the future role of nuclear energy in a sustainable development perspective and within the broad context of national and international energy policies (NEA, 2008x). In pursing this goal the NEA has a number of objectives:

- to analyse the impact of changes in electricity markets on nuclear power and assist member countries in assessing the role of nuclear energy in their energy policies;
- to promote international co-operation for the development of innovative nuclear energy systems;
- to review nuclear power in the broader perspective of sustainable development;
- to assess the availability of nuclear fuel and infrastructure required for the deployment of nuclear power and identify the eventual gaps;
- to establish a communication network within and outside the OECD framework aiming at providing factual information on nuclear issues.

Many of these objectives have no direct link to the need for facilities. However, the development of innovative and sustainable systems requires appropriate research and development and hence relevant facilities.

The programme of work in the nuclear development area covers a wide range of studies on the economics and technology of nuclear power and the organisation of relevant meetings for the exchange and consolidation of information. It maintains a close collaboration with other parts of the OECD, especially the International Energy Agency (IEA), as well as with the International Atomic Energy Agency (IAEA) and relevant sections of the European Commission. The Committee for Technical and Economic Studies on Nuclear Energy Development and Fuel Cycle (NDC) is the body that provides guidance on this work, ensuring that it reflects the priorities of member countries. It has one sub-group, the Joint NEA/IAEA Uranium Group (NEA, 2008t).

The bulk of the work is carried out by a number of *ad hoc* expert groups, each convened to perform a specific task. These studies are aimed at the publication of consensus reports based on the experts' experience in regard to the issues in question. The studies frequently establish a framework that is widely used by governments, NGOs and research organisations and that provides a basic corps of economic or technical data. Other topics are the subject of international workshops organised by the Secretariat. These meetings may lead to the publication of proceedings or to the commissioning of further analyses by expert groups.

The Secretariat also carries out analytical work directly for the NDC and ensures that the results of all the work are fed into activities of other parts of the NEA and the OECD, and international bodies such as the International Panel on Climate Change (IPCC, 2008).

For a list of *ad hoc* expert groups and ongoing studies see the Nuclear Development website (NEA, 2008x); however, see also the NEA activity on partitioning and transmutation, which is discussed in the *Joint Projects* section of this chapter (4.6.2).

The website also provides links to information on the following expert groups which have completed their mandates:

- The Impact of Nuclear Power Plant Life Extension study (report published at the end of 2006);
- Innovation in Nuclear Energy Technology (report published in February 2007);

- The Management of Recycled Fissile Materials (report for policy makers was published in 2007);
- The Timing of High-level Radioactive Waste Geological Disposal (report published in 2008).

4.3 Radiological protection

The goal of the NEA in this area is to assist member countries in the regulation and implementation of the system of radiological protection by identifying and addressing conceptual, scientific, policy, regulatory, operational and societal issues in a timely and prospective fashion, and clarifying their implications (NEA, 2008rr). As well as having objectives relating to regulation and operational capabilities, there are objectives relating to radiological protection science and the improvements in the implementation of scientific knowledge for practical usage. The Expert Group on the Implications of Radiological Protection Science (EGIS) (NEA, 2008n) has recently published its report (NEA, 2007b) which refers to facilities associated with the science of irradiation and dose measurement.

4.4 Nuclear Science

The goal of the NEA in this area is to help member countries identify, collate, develop and disseminate basic scientific and technical knowledge required to ensure safe, reliable and economic operation of current nuclear systems and to develop next-generation technologies (NEA, 2008z). The NEA nuclear science programme is developed and executed by the Nuclear Science Committee (NSC), comprising high-level scientific experts from all NEA member countries. Close co-operation with the Data Bank (NEA, 2008g) is maintained, primarily due to the mutual benefit that exists between the programme of work of the Data Bank and the type of activities pursued within the nuclear science programme.

The main areas of work are:

- reactor physics;
- fuel cycle physics and chemistry;
- criticality safety;
- material science;
- nuclear data.

Further details and links can be found on the Nuclear Science website (NEA, 2008z).

Because the work of the current Expert Group falls under the remit of the NSC and much of the content of this report relates to the NSC area of work, this section of Chapter 4 will not attempt to duplicate information already contained in Chapter 3 of this report. However, brief information on the NSC working parties is given below.

4.4.1 Working Party on Scientific Issues of Reactor Systems (WPRS)

This working party studies the reactor physics, fuel cycle, fuel behaviour, thermal-hydraulics and dynamics/safety of present and future nuclear power systems and performs uncertainty analyses of present and future nuclear power systems (NEA, 2008G). The working party's objective is to provide member countries with up-to-date information to preserve knowledge and develop consensus regarding:

- reactor physics, fuel behaviour, thermal-hydraulics and dynamics/safety issues associated with innovative fuels in present and future nuclear power systems;
- fuel cycle aspects considered will focus on fuel loading and discharge requirements, fission product and minor actinide inventories and radiotoxicity profiles versus time;
- fuel behaviour, thermal-hydraulics and kinetics/safety will be considered insofar as they impinge on the reactor performance;
- radiation transport and dosimetry will cover aspects relevant fro reactor vessels and internals, and irradiation facilities.

Reactor types considered include, but are not limited to the following:

- present-generation LWR with advanced and innovative fuels, evolutionary and innovative LWR and HWR;
- novel reactor systems (GNEP, Gen. IV systems);
- accelerator-driven (subcritical) and critical systems for waste transmutation.

The WPRS liaises closely with other NEA working groups, and particularly close working relationships are maintained with the Working Party on the Scientific Issues in Fuel Cycle (WPFC); see Section 4.4.2 below.

The WPRS provides advice to the nuclear community on the developments needed to meet the requirements (data and methods, validation experiments, scenario studies) for different reactor systems through a number of deliverables which include reports on benchmark exercises, irradiation exercises and uncertainty analyses. For a full list see the WPRS website (NEA, 2008G).

Also within the WPRS programme of work are the following:

- PBMR Coupled Neutronics/Thermal-hydraulics Transients Benchmark – The PBMR-400 Core Design (NEA, 2008mm).
- Expert Group on Reactor Stability and LWR Transient Benchmarks (NEA, 2008k).

4.4.1.1 Expert Group on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF)

This expert group deals with multiple aspects related to the modelling and design of accelerator shield systems including electron accelerators, proton accelerators, ion accelerators, spallation sources and the following types of facilities: synchrotron radiation facilities; very high-energy radiation facilities; accelerator production of tritium; and free electron lasers (NEA, 2008l). The expert group's objectives are to:

- promote the exchange of information among scientists in this field;
- identify areas in which international co-operation could be fruitful;
- carry out a programme of work in order to achieve progress in specific priority areas.

The ninth in the series of SATIF meetings was held in April 2008 (NEA, 2008v) while the eighth was in 2006 (NEA, 2008m). Links to a number of other related publications are also provided on the SATIF webpage.

4.4.2 Working Party on Scientific Issues of the Fuel Cycle (WPFC)

This working party deals with scientific issues in various existing and advanced nuclear fuel cycles, including fuel cycle physics, associated chemistry and flow sheets, the development and performance of fuels and materials, and accelerators and spallation targets (NEA, 2008H).

It has a structure based around expert groups on:

- heavy liquid metal (HLM) technology [including benchmarking of thermal-hydraulic loop models for lead-alloy-cooled advanced nuclear energy systems (LACANES)];
- chemical partitioning (which has now had programmes on flow sheet studies and separations criteria merged into its remit);
- fuel cycle transition scenarios studies.

Further expert groups on innovative fuels and innovative materials are in preparation.

The WPFC website provides links to various relevant meetings. These include the NEA Information Exchange Meetings on Actinide and Fission Product Partitioning and Transmutation (IEM P&T) organised in co-operation with the Nuclear Development Committee (NEA, 2008a, 2008b), the International Workshops on Utilisation and Reliability of High Power Proton Accelerators (HPPA) (NEA, 2008vv) and the Workshop on Structural Materials for Innovative Nuclear Systems (SMINS) (NEA, 2008I).

Other recent publications and reports not previously mentioned in this section include:

- *Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies – 2007 Edition* (NEA, 2007);
- *Physics and Safety of Transmutation Systems – A Status Report* (NEA, 2006a);
- *Fuels and Materials for Transmutation – A Status Report* (NEA, 2005);

The WPFC website also provides a link to a list of other NEA publications on the fuel cycle and P&T. In addition it links to the series of information exchange meetings on partitioning and transmutation of minor actinides and fission products, which is a joint activity with the Nuclear Development Division and is described below in Section 4.6.1.

4.4.3 Working Party on Nuclear Criticality Safety (WPNCS)

WPNCS deals with technical and scientific issues relevant to criticality safety. Specific areas of interest include (but are not limited to) investigations of static and transient configurations encountered in the nuclear fuel cycle (NEA, 2008F). These include fuel fabrication, transport and storage. The WPNCS's objectives are to:

- exchange information on national programmes in the area of criticality safety;
- guide, promote and co-ordinate high priority activities of common interest to the international criticality safety community, establish co-operation;
- monitor the progress of all activities and report to the NSC;
- publish databases, handbooks and reports;
- facilitate communications within the international criticality safety community through relevant websites;
- co-ordinate the ongoing series of International Conferences on Nuclear Criticality Safety (ICNC), to be held every four years;
- co-ordinate WPNCS activities with other working groups within the NEA and in other international frameworks to avoid duplication of activities;
- provide a technical basis for other international activities (*e.g.* ISO, IAEA).

[NB A Workshop on "Criticality Safety Research Needs for Future Nuclear Systems", is due to be held in September 2009, in Idaho, USA, while the International Conference on Nuclear Criticality Safety (ICNC'07) was held 28 May-1 June 2007, in St. Petersburg, Russia.]

WPNCS manages a number of expert groups and other activities. The expert groups' topics include:

- burn-up credit;
- uncertainty analysis for criticality safety assessment;
- source convergence for criticality analyses;
- criticality excursion analyses.

In addition, the Expert Group on Minimum Critical Values has completed its report and the EG has been closed (NEA, 2006b).

A further Expert Group on Assay Data for Spent Nuclear Fuel was mentioned earlier (NEA, 2008i) in Section 3.5.1.4 and is described in more detail in Section 4.4.3.1, followed by discussion of the Spent Fuel Isotopic Composition Database (SFCOMPO) (NEA, 2008tt).

The WPNCS also embraces the work on the International Criticality Safety Benchmark Evaluation Project (ICSBEP) described in Section 4.4.3.3.

Links to the activities of all these expert groups can be found via the WPNCS website (NEA, 2008F), all of which demonstrate the need for experimental data, databases and corresponding benchmark information, and are thus consistent with the requirements enumerated in more detail in Chapter 3.

4.4.3.1 Expert Group on Assay Data for Spent Nuclear Fuel

This NEA expert group was set up to pursue two activities in parallel: updating the SFCOMPO database (NEA, 2008tt), and writing a state-of-the-art report on the assay data of spent nuclear fuel. The expert group's report will cover the full process, ranging from the experimental methods used, to the best way to present and structure the data for its various applications. The applications of the data include: burn-up credit analysis, reactor system monitoring, reprocessing of nuclear fuel and radioactive waste management. Under the guidance of WPNCS, the major assignments of the expert group include:

- analysing the SFCOMPO database in order to assess the current situation and the need for new experimental data;
- collecting new isotopic composition data from post-irradiation examination (PIE) and incorporating them and their associated operating histories/data into the SFCOMPO database, and review of the SFCOMPO database's format;
- archiving original reports on any PIE data included in the SFCOMPO database, as well adding data references that were used in the original development of the data;
- providing technical advice in support of PIE activities in member countries and promoting international collaboration.

The SFCOMPO database is described separately in the next section.

4.4.3.2 Spent Fuel Isotopic Composition Database (SFCOMPO)

The Spent Fuel Isotopic Composition database (SFCOMPO) (NEA, 2008tt) was originally developed at the JAERI Department of Fuel Cycle Safety Research's Fuel Cycle Safety Evaluation Laboratory. It provides access to isotopic composition data via the Internet (Suyama, 1997; Mochizuki, 2001). In particular it archives measured isotopic composition data and the values of their ratios, which are required for the validation of burn-up codes.

Discussions at WPNCS in December 2001 led to the system for Internet dissemination of SFCOMPO being transferred from JAERI to the NEA Data Bank where it is now operated by NEA.

SFCOMPO uses javascript to search the data covering the following areas:

- reactor name;
- reactor type;
- active height [mm];
- assembly name;
- assembly location;
- fuel rod position;
- sampling position of fuel rod [mm];
- initial enrichment [wt.%];
- cooling time [year];
- laboratory;
- burn-up [GWd/tU];
- ^{148}Nd method;
- ^{137}Cs destructive method;
- ^{137}Cs non-destructive method;
- U, Pu isotope based method;
- theoretical;
- PIE data [kg/tU initial].

4.4.3.3 International Criticality Safety Benchmark Evaluation Project (ICSBEP)

The purpose of the "International Criticality Safety Benchmark Experiments Project" (ICSBEP) (INL, 2008) is to identify a comprehensive set of critical benchmark data and then to verify these data (to the extent possible) by reviewing the original documentation and any subsequent revised version. In addition, information is sought by talking with the original experimenters or other individuals who are familiar with the experimenters or the experimental facility.

Next, the data are evaluated and the overall uncertainties quantified through various types of sensitivity analysis. The data are then compiled into a standardised format. In addition, calculations are performed of each of the experiments included in the database with standard criticality safety codes.

Finally, the work has been collated into a single formal document, the ICSBEP Handbook, which can be used as a source of verified benchmark critical data.

As of May 2008, the ICSBEP website reports that the handbook spans over 42 000 pages and contains 464 evaluations representing 4 092 critical, near-critical, or subcritical configurations, 21 criticality alarm placement/shielding configurations with multiple dose points for each, and 46 configurations that have been categorised as fundamental physics measurements that are relevant to criticality safety applications. The handbook is intended for use by criticality safety analysts to perform necessary validations of their calculational techniques and is expected to be a valuable tool for decades to come. The handbook is currently in use in 60 countries.

The Database for the International Criticality Safety Benchmark Evaluation Project (DICE) is included on the DVD version of the ICSBEP Handbook. DICE is a tool intended to make more efficient use of the handbook and will enable users to more easily identify the information that meets their needs. DICE accomplishes two main objectives:

- It provides a summary description of each experimental configuration, where the main characteristics of the experiments are displayed in a uniform format.
- It allows users to search the handbook for experimental configurations that satisfy their unique input criteria (much more than a word search).

DICE is regularly improved following feedback from users.

While ICSBEP is primarily of interest to the Criticality Safety and Nuclear Data Communities, many of the benchmarks can be of significant value to the reactor physics community where the IRPhE Project (NEA, 2008r) is preserving integral data. Thus ICSBEP and the IRPhEP share a common purpose to identify, evaluate, verify, compile and document the information into resources of extensively peer reviewed benchmark data. For more details of the IRPhE project see Section 4.5.1 below.

Classification

Several groups of experimental facilities can be defined. These are split tables, either horizontal or vertical, simple cylindrical tanks containing solutions, water tanks containing arrays of commercial fuel rods and finally full reactor core systems.

The facilities belonging to the first group were created for determining critical masses of weapons-type devices at the early stages of nuclear technology development. The facilities of the second group were required for guaranteeing criticality safety in the chemical processes associated with Pu extraction, while others are associated with the demonstration of criticality safety in the storage of commercial fuel in water pools or in storage casks.

Experimental reactors have also been being used for performing critical experiments with the important contribution of ZPPR of ANL, BFS of IPPE and DIMPLE of UKAEA; the database contains results from many other reactors, however, including standard commercial training reactors to the specific-purpose devices such as space or naval propulsion simulators. Some are treated as critical assemblies for their start-up, while others are simple drivers of subcritical systems. The interest in these facilities is due to the large amount of associated operational information not yet properly recovered, evaluated or reported.

While ICSBEP is primarily of interest to the criticality-safety and nuclear-data communities, many of the benchmarks can be of significant value to the reactor physics community, for which the

IRPhE Project (NEA, 2008r) is also preserving integral data. Thus ICSBEP and IRPhEP share a common purpose to identify, evaluate, verify, compile and document the information into resources of extensively peer-reviewed benchmark data.

For more details concerning the IRPhE project see Section 4.5.1 below.

Experimental needs

Concerning the need for new experiments, a lack of intermediate enrichments and intermediate spectra evaluations can be deduced from analysis of the ICSBEP database. This has led to the use of heterogeneous mixtures of HEU and depleted uranium to obtain target average enrichments because of the lack of available suitably enriched materials.

An intermediate neutron energy spectrum has been a difficult target for experimenters with only a limited number of intermediate spectrum cases in ICSBEP. A collaborative effort in the use of the existing facilities could be very fruitful.

4.4.4 Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems (WPMM)

The WPMM will review and evaluate the multi-scale modelling and simulation techniques currently employed in the selection of materials used in nuclear systems. During the first meeting at the NEA on 15-16 January 2008, WPMM members shared past experiences in developing models for assessing nuclear materials and identified a number of scientific challenges that they plan to explore further (NEA, 2008E). See also Section 3.6.3.3.

4.4.5 Working Party on International Nuclear Data Evaluation Co-operation (WPEC)

The Working Party on International Nuclear Data Evaluation Co-operation (WPEC) was established to promote the exchange of information on nuclear data evaluations, measurements, nuclear model calculations, validation and related topics, and to provide a framework for co-operative activities between the participating projects. The working party assesses nuclear data improvement needs and addresses these needs by initiating joint evaluation and/or measurement efforts.

The evaluation projects involved in this co-operative effort are the Japanese Evaluated Nuclear Data Library (JENDL), the US Evaluated Nuclear Data File (ENDF) and the European Joint Evaluated Fission and Fusion (JEFF) project. (BROND, CENDL and FENDL). The participation of NEA non-member country activities, such as the Russian BROND and the Chinese CENDL projects, is supported by the Nuclear Data Section of the International Atomic Energy Agency (IAEA).

The work of the WPEC (NEA, 2008D) has been outlined in Section 3.1.

4.5 Nuclear Science and Data Bank activities on integral data preservation

As well as the work on ICSBEP mentioned above (Section 4.4.3.3), the closely related IRPhE Project is worthy of mention. The NEA is also preserving data in the fields of radiation shielding [Radiation Shielding Experiments Database (SINBAD)] and fuel performance [International Fuel Performance Experiments Database (IFPE)].

4.5.1 The International Reactor Physics Benchmark Experiments (IRPhE) Project

The aim of the IRPhE Project (NEA, 2008r) is to provide the nuclear community with qualified benchmark data sets by collecting reactor physics experimental data from nuclear facilities world wide. More specifically the objectives of the expert group are as follows:

- maintaining an inventory of the experiments that have been carried out and documented;
- archiving the primary documents and data released in computer-readable form;
- promoting the use of the format and methods developed and seeking to have them adopted as a standard.

Guidance or co-ordination is provided in: *i)* compiling experiments into a standard international agreed format; *ii)* verifying the data (to the extent possible) by reviewing original and subsequently revised documentation, and by consulting with the experimenters or individuals who are familiar with the experimenters or the experimental facility; *iii)* analysing and interpreting the experiments with current state-of-the-art methods; *iv)* publishing electronically the benchmark evaluations.

The expert group will:

- identify gaps in data and provide guidance on priorities for future experiments;
- involve the young generation (Masters and PhD students and young researchers) to find an effective way of transferring know-how in experimental techniques and analysis methods;
- provide a tool for improved exploitation of completed experiments for Gen. IV reactors;
- co-ordinate closely its work with other NSC experimental working groups, in particular the International Criticality Safety Benchmark Evaluation Project (ICSBEP) (INL, 2008), the Shielding Integral Benchmark Experiment Data Base (SINBAD) (NEA, 2008pp) and others, *e.g.* knowledge preservation in fast reactors of the IAEA, the ANS Joint Benchmark Activities;
- maintain a close link with the Expert Group on Reactor-based Plutonium Disposition (TFRPD) (NEA, 2008j) and the Working Party on International Evaluation Co-operation (WPEC) (NEA, 2008D).

Fundamental mode lattice experiments, heterogeneous core configurations, power reactor start-up data, core follow experiments, and experiments with specific applications such as fission product integral data and irradiation experiments are the types of experiment included in the database.

The benchmark specifications and experimental data are intended for use by nuclear reactor physicists and engineers to validate current and new calculational schemes including computer codes and nuclear data libraries, for assessing uncertainties, confidence bounds and safety margins, and to record measurement methods and techniques.

The third edition of the International Handbook of Evaluated Reactor Physics Benchmark Experiments was published in March 2008 and contains data from 25 different experimental series that were performed at 17 different reactor facilities. Twenty-one of the 25 evaluations are published as approved benchmarks. The remaining four evaluations are published as draft documents only. These draft documents have been reviewed by the IRPhEP Technical Review Group (TRG); however, all action items could not be completed or reviewed in time for the final publication or, in most cases, the TRG felt it necessary to review the revised evaluations before giving final approval. Additional evaluations are in progress and will be added to the handbook periodically. The handbook is published in electronic format (.pdf files) on DVD, and further details are given on the IRPhE Project website (NEA, 2008r).

Much of the work realised so far by the IRPhE project, in particular the evaluation and review of selected benchmark experiments, was possible thanks to substantial funding provided by the government of Japan. Belgium, Brazil, Canada, PR of China, Germany, Hungary, Japan, the Republic of Korea, the Russian Federation, Switzerland, the United Kingdom, and the United States of America have contributed evaluations, reviews and data at their own expense. Overall technical co-ordination of the IRPhEP is directly supported by the United States Department of Energy's Office of Nuclear Energy with significant in-kind contributions from the parallel OECD NEA International Criticality Safety Benchmark Evaluation Project (ICSBEP), supported in the United States by the Department of Energy's Office of Facility Management and ES&H Support.

4.5.2 The Radiation Shielding Experiments Database (SINBAD)

SINBAD (NEA, 2008pp) is a unique set of experiments in standard format for validation and benchmarking of computer codes and the nuclear data used for radiation transport, shielding and dosimetry problems.

A new release of the Radiation Shielding Experiments Database (SINBAD) was issued in 2007. As of October 2007 the SINBAD website records that the database contains compilations for 42 reactor shielding, 27 fusion neutronics and 15 accelerator shielding experiments. This work is jointly carried out by the Radiation Safety Information Computational Center (RSICC) (ORNL, 2006) and the NEA

Data Bank. Data for 84 experiments have been collected and the major emphasis has, so far, been on fission reactor shielding. More data sets are in the process of being identified for a future release. Emphasis will be placed on the quality of the experiments and new compilations will address cases not yet sufficiently covered by the present set.

A recent report provides further details about SINBAD (Kodeli, 2006).

4.5.3 The International Fuel Performance Experiments Database (IFPE)

The "International Fuel Performance Experiments" (IFPE) Database (NEA, 2008oo) is being compiled with information from integral experiments and as of May 2008 contains data sets concerning 1 436 rods/ samples from various sources encompassing BWR, CAGR, PHWR, PWR; VVER reactor systems have also been included.

The aim of the project is to provide a comprehensive and well-qualified database on Zr-clad UO_2 fuel for model development and code validation in the public domain. The data encompass both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in Material Testing Reactors. This work is carried out in close co-operation and co-ordination between the OECD/NEA, the IAEA and the IFE/OECD/Halden Reactor Project.

Activities associated with the database are:

- acquisition of data through discussion and negotiation with originators;
- compilation of the data into a standard form and content as agreed upon by an expert group set up to supervise the work;
- peer review of the data by independent experts;
- integration and indexing of the data into the IFPE database, inclusion of all used reports in electronic form;
- distribution to interested parties and assistance where necessary in use of data sets.

The database is restricted to thermal reactor fuel performance, principally with standard product Zircaloy-clad UO_2 fuel, although the addition of advanced products with fuel and clad variants is not ruled out. Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. Of particular interest to fuel modellers are data on: fuel temperatures, fission gas release (FGR), fuel swelling, clad deformation (*e.g.* creep-down, ridging) and mechanical interactions. Data on these issues are of great value if measured in-pile by dedicated instrumentation and in this respect, the IFPE Database is fortunate in having access to several diverse experiments. In addition to direct in-pile measurement, every effort is made to include PIE information on clad diameters, oxide thickness, hydrogen content, fuel grain size, porosity, electron probe micro analysis (EPMA) and X-ray fluorescence (XRF) measurements on caesium, xenon, other fission products and actinides.

4.6 Other projects

As well as the work areas identified above, the NEA is pursuing a number of other projects.

4.6.1 NEA Project on Partitioning and Transmutation of Minor Actinides and Fission Products

The NEA activity on partitioning and transmutation (P&T) (NEA, 2008nn), which is organised across various NEA work areas, specifically relates to the subjects discussed in Sections 3.4, 3.5 and 3.8.

As a result of the initiative taken by the Japanese government to launch a long-term research and development programme on the recycling and transmutation of actinides and long-lived fission products, known as the OMEGA programme (Minato, 2007; Mukaiyama, 1999), the NEA Committee for Technical and Economic Studies on Nuclear Energy Development and Fuel Cycle (NDC) was invited, in 1988, to conduct some form of international project related to actinide separation and use. The information exchange meetings are a key component of this international project, aiming to give experts a forum to present and discuss current developments in the field. Other activities involve status and assessment reports by expert groups on the broad field of P&T and its impact in the nuclear fuel cycle.

Another part of the project, carried out under the auspices of the Nuclear Science Committee (NSC), includes specific projects and specialists meetings on particular scientific aspects of P&T.

The P&T project is one example of horizontal activity within the NEA, involving several divisions and committees. Apart from the NDC and the NSC, the Radioactive Waste Management Committee (RWMC) (NEA, 2008qq) is involved in the project and informed of its outcome, as the potential application of P&T would have impacts on waste management and disposal not eliminating the need for geological disposal of high-level waste but reducing the radiotoxic inventory to be disposed of. Also, whenever relevant, the NEA co-operates with the International Atomic Energy Agency (IAEA) and the European Commission on specific topics of interest to both agencies.

4.6.2 List of NEA joint projects in the area of nuclear safety

The NEA participates in a number of joint projects (NEA, 2008u); the following were listed on the NEA website in February 2008:

- OECD/NEA Behaviour of Iodine (BIP) Project
- OECD/NEA CABRI Water Loop Project (NEA, 2008aa)
- OECD/NEA Computer-based Systems Important to Safety (COMPSIS) Project
- OECD/NEA Fire Incidents Records Exchange (FIRE) Project
- OECD/NEA Halden Reactor Project (IFE, 2005; NEA, 2008bb)
- OECD/NEA International Common-cause Failure Data Exchange (ICDE) Project
- OECD/NEA Melt Coolability and Concrete Interaction (MCCI) Project (NEA, 2008dd)
- OECD/NEA Piping Failure Data Exchange (OPDE) Project
- OECD/NEA PKL-2 Project (NEA, 2008ee)
- OECD/NEA Fire Propagation in Elementary, Multi-room Scenarios (PRISME) Project
- OECD/NEA PSB-VVER Project (EREC, 2008; NEA, 2008ff)
- OECD/NEA Rig of Safety Assessment (ROSA) Project (NEA, 2008hh)
- OECD/NEA Steam Explosion Resolution for Nuclear Applications (SERENA) Project (NEA, 2008kk)
- OECD/NEA SESAR Thermal-hydraulics (SETH-2) Project (NEA, 2008jj)
- OECD/NEA Stress Corrosion Cracking and Cable Ageing Project (SCAP)
- OECD/NEA Studsvik Cladding Integrity Project (SCIP)
- OECD/NEA Thermal-hydraulics, Hydrogen, Aerosols, Iodine (ThAI) Project (Becker, 2008; NEA, 2008ll)

The website also provides links to a number of completed projects:

- OECD/NEA Bubbler Condenser Project
- OECD/NEA Material Scaling (MASCA) Project
- OECD/NEA MASCA-2 Project (NEA, 2008cc)
- OECD/NEA PLASMA Project
- OECD/NEA RASPLAV Project (NEA, 2008gg)
- OECD/NEA Sandia Lower Head Failure Project
- OECD/NEA SCORPIO Project
- OECD/NEA SESAR Thermal-hydraulics (SETH) Project (NEA, 2008jj)

Projects involving radioactive waste management and radiological protection are also listed but fall outside the remit of the current Expert Group, except the Thermochemical Database (TDB) project (see below), which is a collaboration between the NEA Radioactive Waste Management Committee (RWMC) and the NEA Data Bank.

4.6.3 The Thermochemical Database (TDB) Project

Finally, it is worth mentioning the Data Bank's Thermochemical Database (TDB) Project (NEA, 2008uu), which in the field of radioactive waste management, although it is, perhaps, on the limit of our scope in this report.

The purpose of the TDB project is to make available a comprehensive, internally consistent, internationally recognised and quality-assured chemical thermodynamic database of selected chemical elements. This database should meet the specialised modelling requirements for safety assessments of radioactive waste disposal systems.

High priority was assigned to the critical review of relevant data for inorganic compounds and complexes containing the actinides uranium, neptunium, plutonium and americium, as well as the fission product technetium. Data on other elements present in radioactive waste (as fission or activation products) such as nickel, selenium, zirconium have also been critically evaluated, as well as data for compounds and complexes of the previously considered elements with selected organic ligands (oxalate, citrate, EDTA and isosaccharinic acid). Three new reviews of inorganic species and compounds of other elements (thorium, tin and iron) are currently under way.

The TDB project aims to produce a database that:

- contains data for all the elements of interest in radioactive waste disposal systems;
- documents why and how the data were selected;
- gives recommendations based on original experimental data, rather than compilations and estimates;
- documents the sources of experimental data used;
- is internally consistent;
- treats all solids and aqueous species of the elements of interest for nuclear waste storage performance assessment calculations.

The unique feature of the TDB database is that it contains data which have been evaluated directly from the original experimental data. Only basic formation and reaction thermodynamic data are compiled (Gibbs energies, enthalpies, entropies and heat capacities). No kinetic, diffusion or sorption data are included in the reviews.

A detailed description of the project is available through the website, where links to the TDB review reports can also be found.

Chapter 5: Conclusions and recommendations

The NSC tasked the Expert Group to seek to anticipate future needs for R&D facilities in nuclear science in close collaboration with other NEA standing technical committees with the specific aim that the study should contribute to promoting international collaboration for the development of new nuclear technologies.

While it is possible, from the analysis of the various work areas, to deduce conclusions which apply across the world, this does not necessarily mean that individual countries are willing or able to follow the implications. This dichotomy means that the following conclusions and recommendations should be read in the multi-national context in which they have been gathered, a context consistent with the aims and remit of the NEA. It is, however, encouraging to note the significant amount of existing multi-national activity in the nuclear science area and the fact that the building of new links continues. This multi-national feature is especially pleasing given the commercial and competitive world in which much development is actually undertaken.

While the review and consideration have naturally centred on larger facilities due to the funding and staffing issues implied, the Expert Group also recognised the importance of smaller, flexible facilities and of the necessary instrumentation.

While this report may not directly indicate for individual countries how early there will be a need to plan pilot plants for future facilities, or the consequences of not undertaking specific activities, the Expert Group believes that by demonstrating the breadth of current work and its distribution around the world some impression of the likely directions may be deduced.

Finally, while the problem of availability of a suitably qualified and experienced human resource has not been particularly stressed in our discussions (our remit was "R&D facilities") it has been noted in several components of this review. The issue of human resources is indeed an important factor and, while it is being addressed in various ways in a number of countries and regions (NEA, 2007a), the Expert Group believes there is an ongoing need to manage and plan for the provision of human as well as physical resources for the further development of nuclear energy.

The following sections summarise the conclusions and recommendations from the specific areas identified in Chapter 3.

5.1 Nuclear data

The review of the nuclear data field in Section 3.1 indicated that, while there may appear to be a rather large number of experimental facilities, few of those are actually performing measurements of interest for nuclear energy applications. When considering the availability of modern facilities able to provide results for materials of current technological interest and at the level of accuracy required for present and future applications, it becomes clear that the number of appropriate facilities is much more limited. In addition, major difficulties remain with the procurement of some target materials and the preparation of the corresponding samples.

As far as integral measurements are concerned, there is a similar shortage of facilities and there has been no new construction for quite some time. Only a small number of critical facilities with sufficiently diverse supplies of simulation materials are still operational today. These remaining facilities are essential to the validation of nuclear data and to neutron physics programmes.

We make recommendations relating to the maintenance and development of: *i)* expertise; *ii)* infrastructures and samples. In addition, Section 3.1.3 indicates areas where new facilities and expertise may become necessary in the future.

Expertise

We approve of the emphasis on the skills issue in the statement and recommendations of the NEA Steering Committee for Nuclear Energy regarding a government role in ensuring qualified human resources in the nuclear field (NEA, 2007a).

We regard better integration between academia and other user communities as being of great advantage in evolving evaluation and validation of databases to the appropriate standard required for nuclear energy applications. The challenges in modelling of nuclear reactions and the development of advanced measurement techniques form an excellent basis for scientific education, while the industry and other end users have definitive requirements for broad range and specific data plus reliable estimates of uncertainty. Better use should be made of this symbiosis.

Infrastructure and samples

The differential and integral measurement facilities which are still operating emerged from the efforts which lead to the earlier and current generations of reactors. There are few such facilities left; yet, they will be necessary for further progress in nuclear data, compatible with the users' needs. Therefore, they should be maintained, possibly even upgraded, and made accessible internationally to many users. New construction projects should be encouraged.

The availability of facilities able to supply the samples required for modern measurements is an important additional requirement alongside the availability of the actual measurement facilities themselves.

Equally, the well-qualified databases which provide a solid basis for calculations arose from co-ordinated efforts of the nuclear data community in the past decades. Requirements arising from: *i)* lifetime extension of presently operating reactors; *ii)* evolution of new reactors and their associated fuel cycle infrastructures; *iii)* the more stringent environmental and safety burdens of the current era put challenging demands on the accuracy of the models and the underlying nuclear database. They will only be met if the present level of expertise and facilities is maintained.

5.2 Reactor development

The Expert Group concludes that, generally speaking, obtaining a perspective on the future needs relating to specific research fields and types of reactors is extremely difficult; variations in importance of particular reactor designs over time and in particular countries influences the view. In addition, the current resurgence of interest in novel reactor concepts is widening the field of development and bringing, for example, gas-cooled reactors (GCR) to prominence once again.

As well as the direct requirements for specific research reactors and criticality assemblies related to particular reactor designs, the Expert Group concludes that there is also a need for zero (or low) power reactors and subcriticality assemblies for basic reactor physics experiments and educational purposes. This requirement to extend the knowledge of the skills base applies for any nuclear energy developments in the future regardless of reactor types adopted. On this basis, it is clear that criticality assemblies should be versatile (multi-purpose) in order to be able to react to changing requirements.

The Expert Group believes that this requirement will demand some new research reactors but can partly be met by continuing to operate existing research reactors – with the proviso that they meet best international safety standards. In particular, the need for criticality facilities should be underlined, as they are used for reactor physics and criticality safety studies. Enhancing the life of an existing reactor [*e.g.* the upgrade of JMTR (2008f)] is one possible reaction in the face of the shutdown of existing facilities as being experienced in some locations.

The Expert Group also stresses that there is a continuing requirement for research reactors as sources of neutrons, in particular for high intensity neutron sources as probes of materials. While large and/or multi-purpose accelerators such as IFMIF (ENEA, 2008) and JANNUS (Serruys, 2007) feature in current trends for such facilities, conventional research reactors have advantages such as the ability to provide continuous irradiation or more representative conditions. Thus both types of facility (accelerators and reactors) are required and are complementary in their abilities.

The Expert Group notes that several OECD countries have started construction or have announced their intention to build new nuclear reactors and other nuclear fuel cycle facilities. On the other hand, non-OECD countries like Russia, China and India have already launched active programmes leading to concrete implementation in this field. In particular, significant efforts have been deployed in order to build fast neutron reactors. However, no new fast spectrum power reactor is scheduled to come into operation before 2020 in OECD countries, which will likely hamper innovative R&D activities, especially in connection with fuel research.

The Expert Group believes that OECD countries could possibly provide a new impetus to the promotion of the relevant R&D in order to encourage innovation in the nuclear industry and to retain leadership with regard to nuclear technologies. Clearly needing to remain within the scope of available national budgets, this momentum could assist the further promotion of international collaboration. Possible areas are: the construction of a (perhaps even jointly-owned) fast neutron reactor and/or of a laboratory dedicated to grouped extraction and to the fabrication of minor actinides-loaded fuel (see *Enhancing international co-operation* on page 57).

The Expert Group concludes that the recent expansion of the GNEP partnership is an indication of the desire for increased collaboration within the international nuclear power community in order to address near-term developments. With a longer-term perspective, the Gen. IV initiative (GIF, 2008a) similarly brings together a number of countries through the Generation IV International Forum in developing reactor designs for future application.

The Expert Group recommends that, further federation of the financial, scientific and technical efforts of the OECD countries could optimise available resources. This would have the aim, for instance, of better usage of existing research reactors, or of building jointly-owned nuclear facilities, following an approach similar to that adopted by the Institut Laue-Langevin (ILL, 2008) [or, in the fusion field, ITER (2008)].

The Expert Group notes the key role played by international institutions in the promotion of such co-operation between countries and recommends that existing synergies between NEA and IAEA activities in this matter might be further explored.

The group also wishes to encourage the exchange of researchers, of research plans, and of results, such as the collaboration between facilities in France and Belgium [*e.g.* EOLE (Fougeras, 2007) and VENUS (Baeten, 2008)].

In relation to the skills base, the Expert Group notes that the present nuclear research-related facilities are operated by very competent and experienced scientists and technical staff. They recommend that this foundation of human resource and expertise be preserved, particularly in the current context of a “nuclear renaissance”, which implies the appropriate recruitment and retention of younger staff to replace those reaching retirement.

Equally the group concludes that there is a need, as far as is appropriate, to maintain existing operational facilities and to conserve the body of knowledge built up; this is because experience has shown that technologies that were abandoned may experience a revival (*e.g.* the HTR, now one of the six systems selected within the framework of Gen. IV).

Also in relation to knowledge retention, work on databases of older experiments such as IRPhE (NEA, 2008r) has led to the NSC recommendation that the methods and QA procedures used for those be adopted for documenting current and future experiments. The Expert Group reconfirms this recommendation with the proviso that the QA procedures should be consistent with the needs and not unduly cumbersome; clearly if the demands of the QA process are an undue burden, there is a risk of discouraging contributions.

The Expert Group notes that there is a wealth of information that exists in the results from past experiments on irradiated MOX, carbide, nitride and metal fuels, as well as operational reactor feedback, sodium technology, etc., and this resource should be retained.

Within the context of building up the knowledge base of younger researchers, the Expert Group commends the initiative of: *i)* France and Germany in creating and building up the Frédéric Joliot/Otto Hahn Summer School on Nuclear Reactors Physics, Fuels and Systems (CEA/FZK, 2008); *ii)* the formation of the World Nuclear University (WNU, 2008). Initiatives of this form are to be encouraged.

5.3 Neutron applications

Currently new neutron scattering sources appear to be coming on stream to keep pace with the projected losses due to source closures. However, there will be a concentration of resources in a fewer number of larger sources and, unless the ESS is built soon (ENP, 2003a), there will inevitably be a shift in the centre of gravity from Europe, to North America and Japan.

It is anticipated that neutron diffraction and small angle neutron scattering measurements of the structure and the defects therein will continue to play a role in the testing and development of new engineering materials for nuclear technology. It is also to be expected that the burgeoning fields of strain scanning and texture analysis will grow in importance as they find a wider audience. Inelastic neutron scattering measurements will probably constitute a lower level activity in this field, though should retain an important role in measuring scattering kernels which will be used in Monte Carlo studies to examine the behaviour of moderator performance.

An extensive use of neutron radiography techniques in fuel fabrication processes requires the adoption of standardised methods for non-destructive controls. Neutron radiography facilities should operate in a co-ordinated way in conjunction with standard authorities (ASTM, ISO, etc.) in order to adopt procedures for qualification of beams, technicians and image treatments.

Application of modern neutron radiography methods such as phase-contrast radiography could integrate neutron scattering and neutron radiography competences in a synergic action offering a wider applicability of neutron imaging techniques.

5.4 ADS and transmutation systems

Transmutation technologies using conventional systems or ADS are considered important for the sustainable development of nuclear energy all over the world. The technical challenges for ADS, however, spread over a wide range of fields. It is, therefore, highly desirable to share the experimental efforts in a systematic way. The MEGAPIE project (PSI, 2008) was a good precursor for international collaboration in this field.

An international roadmap for ADS is of importance.

It is considered necessary to build a dedicated accelerator in order to demonstrate its reliability, controllability, economy and safety for application to a nuclear energy system. Such a demonstration accelerator could be coupled with a subcritical reactor to form an experimental ADS. There is some feeling that a global programme (perhaps in a similar form to the ITER project in fusion energy development) is desirable.

Before proceeding to a demonstration plant, establishment of the technical basis from which to deal with MA in nuclear energy systems and to couple a proton accelerator with a fast spectrum reactor are extremely important for the purpose of ensuring reliable design of the system, safety assessment and training of young scientists and engineers. From this viewpoint, the Transmutation Experimental Facility (J-PARC, 2008) under the J-PARC project in Japan is expected to play an important role if it is adapted into an international framework.

A materials properties database for MA and LLFP is important in order to design the fuels for transmutation systems. Rarity of materials and licence restrictions on the amounts permitted in a facility make it difficult to measure the physical and chemical properties of these materials. It is, therefore, recommended that hot cell laboratories [like the Minor Actinide Laboratory (MA-Lab) at ITU Karlsruhe (ITU, 2008a)] be retained and that a sound way be developed to procure MA and LLFP samples for such materials property measurements, and also for nuclear data measurements and reactor physics experiments.

While recent MEGAPIE (PSI, 2008) work demonstrated the feasibility of a high power LBE target and its post-irradiation test will produce valuable knowledge, at present, the status of material irradiation data is too poor to produce a reliable design for a window-type target. It is recommended that a materials properties database be prepared covering a wide range of design conditions such as temperature, oxygen content and flow velocities of the LBE, beam density and irradiation period.

The alternative windowless design is being investigated, mainly in European countries, using a free surface of LBE by inertia. However, the stable control of such a free surface might be difficult when a high power proton beam is incident. A test using a real proton beam with megawatt class is considered necessary to demonstrate the engineering feasibility, before connecting it to a subcritical reactor, as well as mock-up experiments without beams.

It is recommended that an international benchmark of experiments be organised to establish a world standard of the materials properties for LBE, the most-studied candidate for the primary coolant of the subcritical core of an ADS. Moreover, an integral test to verify the feasibility of oxygen control in a realistic reactor vessel would be necessary before constructing a large-scale LBE-cooled nuclear system.

The thermal-hydraulics of the LBE coolant should also be verified by experimental work, *e.g.* local erosion of the materials in the core by the high speed LBE flow. Large-scale components such as heat exchangers and pumps also require development for LBE.

5.5 Fuel

For fuel development and testing, experimental irradiation facilities are essential. The Expert Group therefore recommends that the lifetime of existing key irradiation facilities, such as the Halden Reactor and the Advanced Test Reactor should be extended. Looking to the future, the development of the Jules Horowitz Reactor (JHR) Co-ordination Action (EC, 2008a) is of particular note.

The IFPE database (NEA, 2008oo) should be maintained and extended.

New facilities for Gen. IV conditions – high temperature, high fluxes, different coolant types, etc., require development. While it is noted that some specific loops are already under development, testing or construction, the need for fast spectrum irradiation facilities should be stressed, as the stakes are high with Gen. IV fast reactor fuel research and presently there is a shortage of facilities. R&D on fuel is a long-term process, a point that must be taken into consideration; many irradiation tests under representative conditions are necessary before a new fuel can be considered as being qualified for use.

Besides the fuel itself, structural materials ought to be tested too, in the proper regime of neutron spectrum, fluence, temperature, coolant environment, etc.; see Sections 3.6 and 5.6.

Pertaining to TRISO coated fuel particles (CFPs) for HTGRs, the Expert Group concurs with the recommendation in the SFEAR report (NEA, 2007d) that international collaboration should be strongly considered due to the importance of fuel performance to HTGR safety, the long lead time and the cost of fuel testing. In addition, the EG regards it as important to maintain existing test reactors, such as Cabri, NSRR and ATR, due to their ability to test HTGR fuels.

In relation to hot cells and post-irradiation examination the Expert Group recommends that the long-term availability of hot cells for fuel examination must be guaranteed. Equally, the conclusions of the HOTLAB project (EC, 2005) should be carefully considered, in particular that, while no substantial change is expected at least for the next five years, the longer term is harder to assess in terms of hot cell utilisation. Continuous observation of global developments will be required.

Finally, in relation to fuel cycle chemistry, the Expert Group concludes that:

- A considerable chemical engineering effort will be needed in a phase of scaling-up the proposed partitioning processes to pilot scale and subsequently to industrial prototype.
- Possessing R&D facilities which fulfil the MA handling requirements is becoming a determining factor among P&T oriented countries in view of: *i)* the strict regulations limiting the quantities of MA which can be handled in shielded facilities; *ii)* the construction costs.
- Irradiation facilities for studying the resistance of ligands towards radiolysis are essential for the development of new organic reagents used for an aqueous partitioning process (*e.g.* MARCEL at CEA Marcoule). An actinide laboratory capable of handling significant amounts of actinides is also needed to study the impact of alpha radiolysis.
- Organic synthesis laboratories, analytical chemistry laboratories and structural chemistry laboratories are also important for the development of partitioning processes.

5.6 Materials

The Expert Group concludes that facilities will continue to be required to cover the range of requirements for: *i)* materials irradiation; *ii)* modelling validation/materials characterisation; *iii)* materials testing.

The continuing availability of materials test reactors (MTRs) and the facilities that such reactors are able to provide is regarded as an essential feature of the study of material of interest to reactors and other branches of nuclear science. The scope of the irradiation capabilities will need to increase as the demands from work on new reactor types evolves.

Equally, the availability of large facilities such as spallation sources and reactors for analysis of materials is deemed essential. Spallation sources can also be valuable sources of neutrons for materials irradiation to complement the facilities available at MTRs.

5.7 Safety

The safety section of this report has largely been derived in collaboration with the CSNI effort that led to the SFEAR report (NEA, 2007d). Therefore the conclusions and recommendations that follow are essentially those in the SFEAR report and further details should be sought there. However, the following is a summary and pertains to both the short term and long term:

- CSNI efforts at facility preservation should focus on large facilities, whose loss would mean the loss of unique capability as well as the loss of substantial investment that, in the current climate of tight resources, would not likely be replaced. Preservation also includes maintaining expertise, knowledge, capabilities and personnel essential to infrastructure conservation. (Previous CSNI efforts have kept several large facilities active over the past five years, thus helping the current SFEAR effort. However, many large, expensive and unique facilities are projected to close over the next one to five years.)
- Both CSNI and CNRA should take steps to encourage industry co-operation by emphasising: *i)* the responsibility of industry to develop sufficient data to support their applications; *ii)* the benefits of co-operative research; *iii)* the value of preserving critical research infrastructure.
- Because of the large numbers of hot cells and autoclaves, each country is recommended to monitor the status of these essential facilities and bring to CSNI's attention any concerns regarding loss of critical infrastructure.
- Certain safety issues have no large scale facilities identified for the conduct of relevant research. The appropriate CSNI working groups should evaluate whether or not such facilities are needed to support resolution of these issues.

Short term

The following recommendations are directed toward those actions that CSNI could take in the short term to prevent the loss of key facilities in imminent danger of closure.

- In the thermal-hydraulics area, PANDA and PUMA are in danger of being closed in the next one to two years. These facilities are unique and expensive and at least one should be maintained. Further arguments and a preference are given in the SFEAR report (NEA, 2007d).
- In the severe accident area, most facilities supporting the resolution of the following safety issues for BWR, PWR, VVER and ALWR are in danger in the short term:
  - pre-core melt conditions;
  - combustible gas control;
  - coolability of over-heated cores.
- The SFEAR report recommends that three specific facilities should be preserved due to their replacement cost, high relative ranking and versatility.
- In the other technical areas (fuels, and integrity of equipment and structures) no short-term CSNI actions are recommended.

- The SFEAR report recognises that implementation of the above recommendations is dependent upon interest and commitment of the "host countries" to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries.

Long term

- In the longer term, it is recommended that CSNI adopt a strategy for the preservation of a research facility infrastructure, based upon preserving unique, versatile and hard to replace facilities. (Consistent with Expert Group's remit, this recommendation is based upon supporting currently operating LWRs and PHWRs and the licensing of future ALWRs and APHWRs.) The strategy should include consideration of short- and long-term priorities, cost of preservation (and would this detract substantially from other programmes/facilities) and contingency plans in case of facility loss.
- The factors used in the SFEAR report to arrive at conclusions and recommendations could be useful in developing a long-term strategy for assessing and initiating future co-operative research projects. These include:
  - facility operating and replacement cost;
  - the ability to define a useful experimental programme;
  - long-term resource implication and priorities;
  - industry participation;
  - host country long-term plans and commitment.
- A table of critical research facility infrastructure needs is given in the SFEAR report; those considered unique, hard to replace and having high relative importance in their technical area are identified. CSNI is recommended to monitor the status of these facilities in the longer term with a goal of taking action, as appropriate, to ensure that critical facilities are available for each reactor type to meet the critical research infrastructure needs. In addition, for new reactors and technologies, CSNI should take an active role in encouraging and organising co-operative research efforts, thus contributing to infrastructure preservation. Once again, host country interest will be an important factor in determining which facilities to preserve.

5.8 Nuclear and radiochemistry research

The Expert Group recommended that integrated hot cell laboratories [like the Minor Actinide Laboratory (MA-Lab) at ITU Karlsruhe] be retained to measure basic physical and chemical properties of actinide compounds.

The group also notes that hot cells and glove boxes owned by universities are important tools for education. It therefore recommends a network such as the pooled facilities in ACTINET as an important approach for the effective sharing of facilities and to promote international collaboration.

The group believes synchrotron radiation facilities capable of measuring radioactive samples should be retained, *e.g.* the SSRL for measurement of plutonium samples. In addition there are future requirements to measure properties of actinides and LLFP in spent fuel directly by means of X-ray absorption spectroscopy. Special beam lines like MARS at SOLEIL are needed for measuring highly radioactive samples.

5.9 Miscellaneous facilities

Relating to nuclear process heat for hydrogen production, the Expert Group concurs with the recommendation for further international collaboration in this field made during the 3rd Information Exchange Meeting on the Nuclear Production of Hydrogen in October 2005 (NEA, 2006).

It also endorses the need for consideration of safety issues to ensure that the chemical and nuclear facilities pose no risk to each other.

The Expert Group notes the need for co-operation on matters such as: *i)* safety; *ii)* materials and chemical property measurement and verification; *iii)* materials development, including structural materials, membranes and catalysts; *iv)* advanced fabrication techniques and the implied need for facilities to elucidate the corresponding information.

Relating to simulation and high performance computing infrastructure the Expert Group notes how high performance computers have become a significant part of the infrastructure for research and development relating to nuclear science application as well as in other spheres.

The group suggests that, beyond the advances in computational speed, further large gains should be achievable by investment in the development of novel algorithms, well adapted to the new multiprocessor computer platforms. In particular, new solvers of physics equations required for design, safety and operation of nuclear reactors need to be produced which will open up the possibility of using more refined models and hence the removal of approximations that are no longer justified. Areas of recent development in the nuclear science field are:

- coupling of neutronics and thermal-hydraulics;
- interpretation of in-reactor experiments;
- "mining" of large masses of data efficiently for a better use of the results, better determination of confidence bounds and for improving understanding through fast visualisation.

The Expert Group concludes that benefits can be obtained through innovation in computing methods in nuclear applications.

5.10 Other recommendations

The RTFDB database has been successfully created and published on the web. The Expert Group believes that it has been a most useful tool during the present review. The EG also believes it already forms a valuable resource for the scientific community world wide and therefore encourages the NSC to continue to update and expand RTFDB into the future. This would assist NEA in regularly reviewing the situation of nuclear facilities, especially in those fields for which the risk of losses are high, in order to monitor and comment upon undesirable trends.

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Appendix A: Mandate of the Expert Group

Scope

As a result of recent economic situations and declining budgets in nuclear development activities in many OECD/NEA member countries, numerous problems are arising in maintaining the present technical levels in the field of the nuclear technology, and in preserving existing integral data accumulated in the course of the nuclear development so far. In order to overcome these difficulties, it becomes more and more important to create and advance nuclear technology and infrastructure within an international framework. The NEA has promoted various activities as exemplified below.

The OECD/NEA Nuclear Science Committee (NSC) initiated a study on research and development (R&D) needs in nuclear science, held a workshop on R&D needs for Current and Future Nuclear Systems in November 2002 in Paris and then published a report on these topics. In parallel, NSC is reviewing existing integral reactor physics data within the IRPhE project. In addition, the Nuclear Development Committee (NDC) is documenting the status of research and test facilities to identify mechanisms and policies for promoting international collaboration in the area of nuclear education and R&D. The Committee on the Safety of Nuclear Installations (CSNI) has reviewed the continued need for experimental R&D facilities in the area of safety and has recently established an expert group to study "Support Facilities for Existing and Advanced Reactors (SFEAR)".

Further discussions in the NSC on R&D needs, using the outcome from other standing technical committees, are essential in order to benefit from possible synergies and to make recommendations concerning the future developments of nuclear energy, using new technology developed in member countries.

The NSC Expert Group will seek to anticipate the future real needs of research facilities for R&D needs in nuclear science based on the results of the above NEA activities and in close collaboration with other NEA standing technical committees. The results of the study will contribute to promoting international collaboration for the development of new nuclear technologies.

Objectives

The Expert Group will prepare a report on future research and test facilities needed in nuclear science. The study will mainly focus on:

- Reviewing the status of research and test facilities world wide and clarify future needs of research facilities corresponding to the R&D needs in nuclear science, collaborating with other technical standing committees, based on results from the NSC study on the R&D needs, as well as those of the NDC and CSNI on review of the status of research and test facilities.
- Monitoring the NSC IRPhE activity on existing integral data of reactor characteristics and fuel cycle in order to identify the future needs of research facilities.
- Establishing recommendations on future needs of research facilities in nuclear science for international collaboration.

Deliverables

- Organise Expert Group meetings to review and exchange information on status of integral data and needs for research and test facilities for future R&D in the field of nuclear science, such as nuclear data, reactor physics, fuel behaviour, material science, fuel cycle chemistry, nuclear production of hydrogen, high performance computing and thermal-hydraulics. The work will be undertaken in close collaboration with CSNI and NDC.
- Establish a database of research and test facilities for R&D in the field of nuclear science, to clarify the status and needs of these facilities.
- Produce a report on the status of integral data and the need of research and test facilities for future R&D in nuclear science.

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